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

October 5, 2005

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

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 525th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, SEPTEMBER 8-10, 2005 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 525th meeting, September 8-10, 2005, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letters, and memoranda:

REPORTS:

Reports to Nils J. Diaz, Chairman, NRC, from Graham B. Wallis, Chairman, ACRS:

- Report on Two Policy Issues Related to New Plant Licensing, dated September 21, 2005
- Report on the Safety Aspects of the License Renewal Applications for the Millstone Power Station, Units 2 and 3, dated September 22, 2005
- Draft Final Revisions to Generic License Renewal Guidance Documents, dated September 22, 2005

LETTERS:

Letters to Luis A. Reyes, Executive Director for Operations (EDO), NRC, from Graham B. Wallis, Chairman, ACRS:

- Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated September 20, 2005
- Interim Letter: Exelon Generation Company, LLC, Application for Early Site Permit and the Associated NRC Staff's Draft Safety Evaluation Report, dated September 22, 2005
- Report on a Proposed Technical Basis for Revision of the Embrittlement Criteria in 10 CFR 50.46, dated September 23, 2005

MEMORANDA:

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

- Draft Final Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities," dated September 14, 2005
- ACRS Review of the North Anna Early Site Permit Application - Final Safety Evaluation Report Changed Pages Prior to Publishing as NUREG, dated September 20, 2005
- Proposed Rule: Licenses, Certifications, and Approvals for Nuclear Power Plants, 10 CFR Part 52 and Conforming Amendments to Parts 1, 2, 10, 19, 20, 21, 25, 26, 50, 51, 54, 55, 72, 73, 95, 140, 170, and 171, dated September 20, 2005
- Proposed Revision to Standard Review Plan Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," dated September 22, 2005
- Proposed Draft Regulatory Guide DG-8028, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," dated September 23, 2005

HIGHLIGHTS OF KEY ISSUES

1. Final Review of the License Renewal Application for Millstone Power Station, Units 2 and 3

The Committee met with NRC staff and representatives of Dominion Nuclear Connecticut, Inc. (DNC) to review the license renewal applications for the Millstone Power Station (MPS), Units 2 and 3 and the associated NRC staff's final Safety Evaluation Report (SER). DNC requested approval for continued operation of each unit for 20 years beyond the current license expiration dates. The operating licenses for Units 2 and 3 expire on July 31, 2015, and November 25, 2025, respectively. Unit 2 is a four-loop Combustion Engineering pressurized water reactor (PWR) that is licensed to operate at 895 MWe. Unit 3 is a four-loop Westinghouse PWR that is licensed to operate at 1195 MWe. Unit 1 is permanently defueled but certain structures such as the turbine building and control room/radwaste treatment building are within the scope of license renewal for Units 2 and 3.

The draft SER was issued in February 2005 and contained six open items. All these open items have been resolved to the satisfaction of the staff.

In the draft SER, the staff granted an exception to the Fire Protection Program in that there are no aging effects requiring management for halon and carbon dioxide fire protection systems. Based on ACRS members' comments and further staff review, this exception was withdrawn. DNC has committed to manage aging of halon and carbon dioxide systems consistent with the Generic Aging Lessons Learned (GALL) Report.

The staff described the recent performance of MPS Units 2 and 3 as well as recent inspection findings. In the August 2005 final SER, the staff concluded that the applicant has satisfied the requirements of 10 CFR Part 54.

In a letter to the ACRS Chairman, dated September 7, 2005, a member of the public representing the Connecticut Coalition Against Millstone made several claims that were not discussed in the SER. These include equipment failures, alleged cases of cancer, and the recent (April 17, 2005) shutdown of Unit 3 due to a "tin whisker" causing a short on a circuit board. The Committee plans to discuss a proposed response to this letter at its October meeting.

Committee Action

The Committee issued a report to the NRC Chairman regarding this matter, dated September 22, 2005, concluding that the programs established and committed to by the applicant provide reasonable assurance that MPS, Units 2 and 3 can be operated in accordance with their current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The Committee recommended that the applications for renewal of the operating licenses for MPS, Units 2 and 3 be approved.

2. Interim Review of the Exelon/Clinton Early Site Permit Application

The Committee heard presentations by and held discussions with representatives of the NRC staff and Exelon Generation Company, LLC (the applicant) regarding the application for an early site permit (ESP) for the Clinton site, and the related staff's draft SER.

Exelon Generation Company, LLC, has applied for an ESP for locating nuclear power plants or modules having a total power generation rate of 2400 to 6800 MWt on the site where the Clinton plant, a boiling water reactor (BWR6) within a Mark III containment, is currently operating. The ESP application is based on the plant parameter envelope approach since the applicant has not identified the particular reactor technology that will be adopted. The applicant has chosen to characterize the seismic hazard using a methodology that differs from that utilized in previous ESPs.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations on this matter, dated September 22, 2005. The Committee recommended that a thorough, expeditious review of the applicant's performance-based seismic hazard analysis methodology be conducted, recognizing that this methodology may be used by applicants for purposes other than ESPs.

3. Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident"

During its consideration of the Proposed Revision 4 to Regulatory Guide (RG) 1.82, the Committee heard from the staff and from members of the public, including the State of Vermont. The Committee considered its previous position on granting containment overpressure credit, as stated in its December 12, 1997 letter (i.e., "selectively granting credit for small amounts of overpressure for a few cases may be justified") and more recently in its letter dated September 30, 2003. In that letter, the Committee recommended issuing Revision 3 to RG 1.82. That RG included a provision to grant, only where necessary, some containment accident pressure credit for some operating reactors with the caveat that "this should be minimized to the extent possible."

The Committee is concerned that the position that the overpressure should be conservatively calculated is the only explicit restriction on the use of overpressure credit given in the proposed revision of the RG. It believes that additional restrictive guidance should be placed on the granting of overpressure credit. Before such credit can be granted, licensees should demonstrate that there are no practical alternative approaches that can eliminate the need for such credit. Such credit should be granted only for robust containments for which there are positive means for indication of containment integrity such as inerted and sub-atmospheric containments. The time intervals for which such credit is needed should be limited to a few hours, commensurate with the demonstrated capability of all associated equipment to perform its intended functions during this time period.

Committee Action

The Committee issued a letter to the NRC EDO on this matter, dated September 20, 2005. The Committee recommended that the RG be revised to include restrictions as discussed above before it is released for public comment.

4. Possible Alternative Embrittlement Criteria to Those in 10 CFR 50.46

The Committee heard a presentation from the staff concerning a new technical basis for revision of the embrittlement criteria in 10 CFR 50.46. The NRC's Office of Nuclear Regulatory Research (RES) has undertaken, in cooperation with the nuclear industry, a confirmatory research program to understand the behavior of fuel cladding at the higher levels of fuel burnup that are becoming common within the nuclear power industry. This research has identified new mechanisms of cladding embrittlement and has improved the understanding of embrittlement mechanisms known at the time the current regulations were written. Based on these early research findings, the RES staff is proposing a revision to the embrittlement criteria that support the regulations that would eliminate reference to specific types of zirconium alloy cladding.

Committee Action

The Committee issued a letter to the EDO on this matter, dated September 23, 2005. The Committee recommended that the staff continue to move forward with the establishment of revised criteria for fuel performance during loss-of-coolant accidents (LOCAs) and write the requirements at a high level, with specific guidance provided in regulatory guides. The staff should also continue to perform the necessary research to validate its proposed process.

5. Draft Final Updates to License Renewal Guidance Documents

The Committee met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss draft final revisions to generic license renewal guidance documents. The staff revised NUREG-1800 (Standard Review Plan for License Renewal Applications for Nuclear Power Plants), NUREG-1801 (GALL Report), and RG 1.188 (Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses). The staff also drafted

NUREG-1832 (Analysis of Public Comments on the Revised License Renewal Guidance Documents) and NUREG-1833 (Technical Bases for Revision to the License Renewal Guidance Documents) to support these revisions. Comments made by ACRS members are addressed in Appendix B of NUREG-1832. New aging management programs (AMPs) that have been added to the GALL Report include the One-time Inspection of ASME Code Class 1 Small-Bore Piping Program, the External Surfaces Monitoring Program, the Metal-Enclosed Bus Program and Electrical Cable Connections not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. The draft final revision to RG 1.188 endorses NEI 95-10, Revision 6 (Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule). The updated license renewal guidance documents consider comments from stakeholders and reflect the staff's current position. The staff concluded that the new documents will increase the efficiency, effectiveness, and consistency of the license renewal review process.

NEI complemented the staff on its effort to update the license renewal guidance documents and provided additional comments on two new AMPs: Metal Enclosed Bus and Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements. In the Metal Enclosed Bus AMP the staff recommends testing of bolted connections in metal enclosed buses every ten years. Visual inspections may be performed every five years as an alternative to testing. NEI questioned the basis for the five year visual inspection interval. NEI also plans to discuss with the staff the aging effects addressed by the AMP on Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.

Committee Action

The Committee issued a report to the NRC Chairman regarding this matter, dated September 22, 2005, recommending that the draft final revisions to the generic license renewal guidance documents be approved for issuance and that the staff continue to evaluate the need for revisions to the guidance documents in order to maintain them current.

6. Meeting with the EDO, Deputy EDOs, and NRC Program Office Directors

The ACRS met with the EDO, Deputy EDOs, and Office Directors of Nuclear Reactor Regulation, Nuclear Regulatory Research (RES), and Nuclear Security and Incident Response to discuss items of mutual interest, including license renewal, new reactors, power uprate issues, fire protection, PWR sump performance issues and emergency preparedness. The EDO and the ACRS Chairman agree that the meeting was useful and that holding such meetings in the future would be mutually beneficial.

7. Interim Results of the Quality Assessment of Selected NRC Research Projects

The NRC Strategic Plan that was developed in accordance with the requirements of the Government Performance and Results Act (GPRA) requires that RES have an independent evaluation of the quality of its research programs. The Committee has agreed to assist RES in assessing the quality of selected research projects. During the September 8-10, 2005 ACRS meeting, the Committee discussed the interim results of the cognizant ACRS panel's quality assessment of the NRC research projects on: Standardized Plant Analysis Risk (SPAR) Models Development Program; Steam Generator Tube Integrity Program at the Argonne National Laboratory; and the Thermal Hydraulic Test Program at the Penn State University.

Committee Action

The Committee plans to discuss the draft report on ACRS assessment of the quality of the selected NRC research projects during its October 6-8, 2005 meeting.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of July 15, 2005, to comments and recommendations included in the ACRS interim letter dated June 9, 2005, concerning the safety aspects of the license renewal application for the Point Beach Nuclear Plant, Units 1 and 2. The Committee decided that it was satisfied with the EDO's response.

  The staff committed to performing a followup inspection under Inspection Procedure (IP) 71002. If the license is renewed, the staff will also conduct a post-approval site inspection for license renewal in accordance with IP 71003 before the period of extended operation begins.

- The Committee considered the EDO's response of July 8, 2005, to the ACRS memorandum dated June 3, 2005. The Committee received a letter from Roger Stoller, Chairman of the American Society for Testing and Materials Committee on Nuclear Technology, expressing his concerns about the lack of NRC staff participation in ASTM standards development activities. The Committee forwarded this letter to the EDO and requested a response regarding how these concerns would be addressed.

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

- The Committee considered the EDO's response of August 17, 2005, to comments and recommendations included in the ACRS letter dated May 13, 2005, concerning report on the safety aspects of the license renewal application for the Arkansas Nuclear One, Unit 2.

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

- The Committee considered the EDO's response of August 12, 2005, to comments and recommendations included in the ACRS interim letter dated June 14, 2005, concerning the staff's draft safety evaluation report on the Grand Gulf early site permit application.

  The Committee decided that it was not completely satisfied with the EDO's response regarding the exposition on threats posed by transportation accidents on the river adjacent to the proposed site, and the prognostication of the weather over the next 65 years based just on historical frequencies of severe weather events. The Committee plans to continue its discussion of this matter with the staff during its review of the final SER.

  The staff has committed to discuss the Committee's comments and concerns during the ACRS review of the final SER.

- The Committee considered the EDO's response of August 2, 2005, to comments and recommendations included in the ACRS letter dated June 14, 2005, concerning the risk-informed, performance-based, fire protection regulatory guide. The Committee decided that it was satisfied with the EDO's response.

  The staff committed to revise the RG to address majority of the ACRS comments and recommendations and provide the draft final RG and industry guidance document for ACRS review.

- The Committee considered the EDO's response of July 20, 2005, to comments and recommendations included in the ACRS letter dated June 10, 2005, concerning the draft final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities." The Committee decided that it was satisfied with the EDO's response.

  The staff has committed to further identify, quantify, and document the uncertainties associated with fire probabilistic risk analyses. Also, if new insights are gained, the staff committed to consider the need to revise NUREG/CR-6850.

- The Committee considered the EDO's response of July 8, 2005, to comments and recommendations included in the ACRS Report dated June 10, 2005, concerning a draft Commission Paper, "Risk-Informed Alternatives to the Single Failure Criterion."

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to discuss further developments in this area as part of its review of the program plan being developed for a risk-informed, performance-based revision to 10 CFR Part 50.

- The Committee considered the EDO's response of September 1, 2005, to comments and conclusions included in the ACRS report dated July 18, 2005, concerning the Dominion Nuclear North Anna, LLC, ESP application and the associated NRC final safety evaluation report. The Committee decided that it was satisfied with the EDO's response.

  The staff committed to interact with the ACRS starting in FY 2006 and to identify issues in the ESP Review Standard that should be modified for the planned revisions to the SRP. The staff committed to interact with the ACRS in developing lessons learned to improve and streamline the ESP process for future applications.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from July 5, 2005, through September 7, 2005, the following Subcommittee meetings were held:

- Planning and Procedures - July 5, 2005

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

- Thermal-Hydraulic Phenomena Subcommittee - July 19-20, 2005

The Subcommittee reviewed the latest proposed staff revision to Regulatory Guide 1.82 related to Emergency Core Cooling System Net Positive Suction Head. The staff described its plans to provide guidance related to containment overpressure credit. The staff presented the results of ongoing research concerning interactions of reactor coolant with debris in the reactor containment sump.

- Reactor Fuels Subcommittee - July 27-28, 2005

The Subcommittee continued its discussion on the proposed criteria for reactor fuel during LOCAs and reactivity insertion events.

- Plant Operations Subcommittee - August 24-25, 2005

The Subcommittee met with representatives of the NRC Region II Office in Atlanta, GA and discussed regional inspection, enforcement, and operational activities.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- In early 2006, the Committee plans to review the final SER associated with the Exelon Generation Company, LLC, application for an ESP.

- The Committee plans in early 2006 to review the draft final version of the proposed rule for licenses, Certifications, and approvals for nuclear power plants, 10 CFR Part 52 and conforming amendments to Parts 1, 2, 10, 19, 20, 21, 25, 26, 50, 51, 54, 55, 72, 73, 95, 140, 170, and 171, after reconciliation of public comments.

- The Committee plans to continue its discussion with the NRC staff regarding policy issues related to new plant licensing and the associated technology-neutral framework document.

- The Committee decided to review the proposed draft Generic Letter 2005-XX, "Steam Generator Tube Integrity and Associated Technical Specifications."

- The Committee plans to review the Browns Ferry Restart Panel's Report prior to restart of Browns Ferry Unit One.

- The Committee plans to review the program plan being developed for a risk-informed, performance-based revision to 10 CFR Part 50, as well as further developments associated with risk-informed alternatives to the single failure criterion.

- The Committee plans to forward the letter dated September 7, 2005 from the Connecticut Commission against Millstone to the EDO for possible action.
- The Committee plans to review licensee responses to Generic Letter 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors.
- The Committee plans to review the draft final version of Regulatory Guide, DG-8028, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

PROPOSED SCHEDULE FOR THE 526th ACRS MEETING

The Committee agreed to consider the following topics during the 526th ACRS meeting, to be held on October 6-8, 2005:

- Interim Review of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3
- Proposed Recommendations for Resolving Generic Safety Issue (GSI)-80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of Boiling Water Reactor Mark I and II Containments"
- Resolution of ACRS Comments on the Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants"
- Davis-Besse Reactor Pressure Vessel Head Integrity Calculations
- Quality Assessment of the Selected NRC Research Projects
- Licensees' Responses to the Bulletin on "Emergency Preparedness and Response Actions for Security-Based Events"
- NRC Staff's Response to the ACRS Letter on the Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident"
- Format and Content of the NRC Safety Research Program Report to the Commission

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

CERTIFIED

Date Issued: 11/10/2005
Date Certified: 11/21/2005

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SEPTEMBER 8-10, 2005

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MEMORANDA (continued)

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APPENDICES

CERTIFIED

MINUTES OF THE 525th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS SEPTEMBER 8-10, 2005 ROCKVILLE, MARYLAND

The 525th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on September 8-10, 2005. Notice of this meeting was published in the *Federal Register* on August 17, 2005 (65 FR 48445) (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. Graham B. Wallis (Chairman), Dr. William J. Shack (Vice Chairman), Mr. John D. Sieber, (Member-at-Large), Dr. George E. Apostolakis, Dr. Mario V. Bonaca, Dr. Richard S. Denning, Dr. Thomas S. Kress, Dr. Dana A. Powers, and Dr. Victor H. Ransom. 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. Graham B. Wallis, 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 Millstone Power Station, Units 2 and 3 (Open)

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

The Committee met with NRC staff, representatives of Dominion Nuclear Connecticut, Inc. (DNC), and a member of the Connecticut Coalition Against Millstone to review the license renewal applications for the Millstone Power Station (MPS), Units 2 and 3, and the associated NRC staff's final Safety Evaluation Report (SER). DNC requested approval for continued operation of each unit for 20 years beyond the current license expiration dates. The operating licenses for Units 2 and 3 expire on July 31, 2015, and November 25, 2025, respectively.

Unit 2 is a four-loop Combustion Engineering pressurized water reactor (PWR) that is licensed to operate at 895 MWe. The lower portions of its two steam generators were replaced with Alloy 690 materials. The Unit 2 reactor vessel head was replaced in the Spring of 2005, and the pressurizer is scheduled to be replaced in the Fall of 2006. DNC noted that the Unit 2 vessel does not have any bottom-mounted instrumentation nozzles.

Unit 3 is a four-loop Westinghouse PWR that is licensed to operate at 1195 MWe. The reactor vessel head is not scheduled to be replaced. It is in the lowest susceptibility category for vessel head penetration cracking and a bare metal visual inspection in 2002 found no evidence of material degradation or leakage. In 2004 a visual inspection of bottom-mounted instrumentation nozzles did not find evidence of material degradation of leakage.

Unit 1 is permanently defueled but certain structures such as the turbine building and control room/radwaste treatment building are within the scope of license renewal for Units 2 and 3.

DNC stated that each unit has 37 license renewal commitments and that implementation of these commitments has already begun. License renewal training is being provided and plant procedures are being changed. The individual tasks for each commitment will be loaded into the plant's Action Item Tracking and Trending System.

The draft safety evaluation report (SER) issued in February 2005 contained six open items. The open item associated with the scoping criteria in 10 CFR 54.4(a)(2) was resolved by the applicant revising its definition of "first equivalent anchor point." The first open item related to the Bolting Integrity Program was resolved by the applicant revising the program to manage loss of preload for all in-scope bolting. The second bolting integrity open item was resolved by the applicant demonstrating that the references for good bolting practices were consistent with the Generic Aging Lessons Learned (GALL) Report. The open item related to the reactor vessel flange leak detection lines were resolved when the applicant agreed to bring them within the scope of license renewal. For the open item associated with the reactor coolant pump Code Case N-481, staff performed an independent fracture mechanics evaluation of the Unit 2 reactor coolant pump casing and determined it to have adequate toughness for the period of extended operation. In addition, the applicant stated that this component would be managed through the "Inservice Inspection Program: Systems, Components, and Supports." The open item related to leak-before-break (LBB) analyses was resolved by the applicant identifying what sections of piping are covered by LBB and providing justification that the LBB analyses were valid for the period of extended operation.

In the draft SER the staff granted an exception to the Fire Protection Program in that there are no aging effects requiring management for halon and carbon dioxide fire protection systems. Based on ACRS comments at the Subcommittee meeting on April 6, 2005, and further staff review, this exception was withdrawn. The applicant has committed to manage aging of halon and carbon dioxide systems consistent with the GALL Report.

The staff described the recent performance of MPS Units 2 and 3 as well as recent inspection findings. In the final SER ,dated August 2005, the staff concluded that the applicant has satisfied the requirements of 10 CFR 54.

A member of the public, who represented the Connecticut Coalition Against Millstone, listed several issues that were not discussed in the SER. These include equipment failures, alleged cases of cancer, the shutdown of Unit 3 due to a "tin whisker" causing a short on a circuit board, and the possibility that Millstone will be required to convert to a closed cooling system.

Committee Action

The Committee issued a report to the NRC Chairman regarding this matter, dated September 22, 2005, concluding that the programs established and committed to by the applicant provide reasonable assurance that MPS Units 2 and 3 can be operated in accordance with their current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The Committee recommended that the applications for renewal of the operating licenses for MPS Units 2 and 3 be approved.

III. Interim Review of the Exelon/Clinton Early Site Permit Application (Open)

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

Dr. Dana A. Powers, Early Site Permits Subcommittee Chairman, stated that the purpose of this meeting is to review and discuss the NRC staff's draft safety evaluation report (DSER), including the supplement regarding the early site permit (ESP) and the application submitted by Exelon Generation Company, LLC (Exelon - the applicant) for the Clinton site. This matter was originally discussed at the Early Site Permits Subcommittee meeting held on September 7, 2005.

Ms. Marilyn Kray, Vice President Exelon, stated that the applicant is requesting an ESP with a duration of 20 years pursuant to Subpart A, "Early Site Permits," of 10 CFR Part 52. The Exelon ESP facility will be co-located on the property of the existing Clinton Power Station (CPS). The CPS site has a man-made cooling reservoir (Clinton Lake), an irregular U-shaped site in DeWitt County, 6 miles east of the city of Clinton. This site is located between the cities of Bloomington and Decatur which are north and south; and, Lincoln and Champaign-Urbana which are to the west and east.

Mr. Eddie Grant, Exelon, noted that the existing CPS Unit 1, is a boiling water reactor (BWR-6) with a rated core thermal power level of 3473 Mwt and a gross electrical output of 1138.5 Mwe. Exelon has not selected a specific reactor type for the ESP site. However, to support its ESP application, Exelon used available information from a range of possible facilities to characterize the proposed development. The proposed ESP facility would be located approximately 700 feet south of the current CPS facility. Depending on the reactor type selected, the Exelon ESP facility could have a total core thermal power rating between approximately 2400 and 6800 Mwt. The Exelon ESP facility would consist of a single reactor or multiple reactors (or modules) of the same reactor type. Unlike the existing plant, which uses the Clinton Lake for normal cooling processes, the ESP facility would use cooling towers. The Clinton Lake would be used as the source of makeup water for the ESP facility cooling water systems.

Mr. John Segala, Office of Nuclear Reactor Regulation (NRR), stated that the NRC staff received early site permit (ESP) applications in September and October 2003 from Dominion Nuclear North Anna, LLC (Dominion), for the North Anna site; Exelon Generation Company, LLC (Exelon), for the Clinton site; and System Energy Resources, Inc. (SERI), a subsidiary of Entergy Corporation, for the Grand Gulf site. All three applications were accepted and the staff's safety and environmental reviews of the applications were conducted.

The staff developed Review Standard (RS-002), "Processing Applications for Early Site Permits," to provide guidance to staff reviewers on the process for reviewing an ESP application. The ACRS has reviewed the RS-002 and stated that such a review standard "is appropriate for review of early site permit applications and will accommodate the industry's proposed use of the Plant Parameter Envelope (PPE)". On May 3, 2004, RS-002 was issued with the Commission's approval. It provides detailed direction for managing and conducting ESP reviews and expands upon existing regulatory guidance.

The regulations of 10 CFR Part 52 and 10 CFR Part 100, "Reactor Site Criteria", that apply to an ESP do not require that an applicant provide specific design information. However, some design information may be required to address 10 CFR 52.17(a)(1) which calls for "an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors."

The DSER summarizes the results of the staff's technical evaluation of the suitability of the proposed site for a nuclear power plant(s) falling within the plant parameter envelope (PPE) that Exelon specified in its application.

In the Site Safety Analysis Report (SSAR) of the ESP application, Exelon provided a list of postulated design parameters referred to as the PPE. The applicant stated that the PPE approach provides sufficient design details to support the NRC's review of the ESP application. Exelon states that the PPE is intended to bound multiple reactor designs. The actual reactor design selected would be reviewed at the combined license (COL) stage to ensure that the design fits within the PPE. The PPE references the following designs:

- ACR-700 (Atomic Energy of Canada, Ltd.)
- Advanced Boiling Water Reactor (General Electric)
- AP1000 (Westinghouse)
- Economic and Simplified Boiling Water Reactor (General Electric)
- Gas Turbine Modular Helium Reactor (General Atomics)
- International Reactor Innovative and Secure (IRIS) Project (Consortium led by Westinghouse)
- Pebble Bed Modular Reactor (PBMR (Pty) Ltd.)

The staff has reviewed the proposed PPE values and has found them to be acceptable. Should an ESP be issued for the Clinton ESP site, an entity might wish to reference that ESP, as well as a certified design, in a COL or construction permit (CP) application. Such a COL or CP applicant would need to demonstrate that the site characteristics established in the ESP bound the postulated site parameters established for the chosen design, and that the design characteristics of the chosen design fall within the PPE values specified in the ESP application.

The NRC staff developed a draft safety evaluation report (DSER) that summarizes the staff's technical evaluation of the Clinton ESP site. The DSER focused on the following matters:

- Population density and land use characteristics of the site environs including seismology, meteorology, geology, and hydrology.
- Potential hazards to a nuclear power plant(s) that might be constructed on the ESP site posed by manmade facilities and activities, transportation accidents, and the existing nuclear power plants.
- Potential capability of the site to support the construction and operation of a nuclear power plant(s) with design parameters falling within those specified in the applicant's PPE.
- Suitability of the site for development of adequate physical security plans and measures.
- Proposed major features for an emergency plan.
- Quality assurance measures applied to the information submitted by the applicant.
- The acceptability of the applicant's proposed exclusion area and low population zone (LPZ) under the dose consequence evaluation factors of 10 CFR 50.34(a)(1).

In developing the DSER, the staff identified certain issues that require additional information. The staff referred to these issues as "Open Items." In addition, the staff has identified items (verification that any ESP application revision is consistent with request for additional information-RAI responses) as resolved, the staff needs confirmation that the applicant has taken the planned action. In addition, the staff has identified permit conditions and site-related COL action items that it will recommend the Commission impose should an ESP be issued to the applicant.

The applicant in its SSAR analyzed and provided the radiological consequences of design-basis accidents (DBAs) to demonstrate that new nuclear units could be sited at the proposed ESP site without undue risk to the health and safety of the public. The applicant, however, did not identify a particular reactor design to be considered for the proposed ESP site. Instead, the applicant developed a set of reactor DBA source term parameters using surrogate reactor characteristics.

In selecting DBAs for dose consequence analyses, the applicant focused on two light-water reactors, the certified ABWR, and the AP1000 designs to serve as surrogates. Using source terms developed from these two designs, the applicant performed radiological consequence analyses for the following DBAs:

- PWR main steamline break
- PWR feedwater system pipe break
- locked rotor accident
- reactor coolant pump shaft break
- PWR rod ejection accident
- BWR control rod drop accident
- failure of small lines carrying primary coolant outside containment
- PWR steam generator tube failure
- BWR main steamline break
- PWR and BWR LOCAs
- fuel handling accident

The applicant calculated site-specific DBA doses by first obtaining information from the ABWR and AP1000 design control documents (DCDs), then calculated site-specific χ/Q values using onsite meteorological information. The applicant then multiplied the doses from the two designs by the ratio of the site-specific χ/Q values, to the assumed χ/Q values from the DCDs. The applicant cited Regulatory Guide (RG 1.183), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000 as the applicable NRC regulations. The NRC staff finds the applicant's site-specific χ/Q values and dose consequence evaluation methodology to be acceptable. In addition, the staff concludes that the proposed distances to the exclusion area boundary and the LPZ outer boundary of the proposed ESP site, in conjunction with the fission product release rates to the environment provided by the applicant as PPE values to be adequate.

The staff developed a supplemental DSER that summarizes the results of the NRC staff's technical evaluation of the suitability of the proposed Exelon ESP site in terms of the site's seismology and geology. The original DSER did not include the review of seismology and geology because the Exelon ESP application included a previously unreviewed performance-based methodology for determining the safe shutdown earthquake (SSE) for the Clinton site.

Exelon, in its application, used the seismic source and ground motion models published by the Electric Power Research Institute (EPRI) for the central and eastern United States (CEUS), "Seismic Hazard Methodology for the Central and Eastern United States," which were issued in 1986. RG 1.165 indicates that the applicant may use the seismic source interpretations developed by Lawrence Livermore National Laboratory (LLNL) in the "Eastern Seismic Hazard Characterization Update," or the EPRI document as inputs for a site-specific analysis.

The NRC staff plans to provide the draft final SER to the Committee by February 8, 2006, and discuss it during the March 2006 meeting. The staff will incorporate the ACRS comments and recommendations and issue the final SER as NUREG by May 1, 2006.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations on this matter, dated September 22, 2005. The Committee recommended that a thorough, expeditious review of the applicant's performance-based seismic hazard analysis methodology should be conducted recognizing that this methodology may be used by applicants for purposes other than ESPs.

IV. Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Open)

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

During its consideration of the Proposed Revision 4 to Regulatory Guide (RG) 1.82, the Committee heard from the staff and from members of the public, including the State of Vermont. The Committee considered its previous position on granting containment overpressure credit, as stated in its December 12, 1997 letter (i.e., "selectively granting credit for small amounts of overpressure for a few cases may be justified") and more recently in its letter dated September 30, 2003. In that letter, the Committee recommended issuing Revision 3 to RG 1.82. That RG included a provision to grant, only where necessary, some containment accident pressure credit for some operating reactors with the caveat that "this should be minimized to the extent possible."

The Committee is concerned about the position that the overpressure should be conservatively calculated is the only explicit restriction on the use of overpressure credit given in the proposed revision of the RG. It believes that additional restrictive guidance should be placed on the granting of overpressure credit. Before such credit can be granted, licensees should demonstrate that there are no practical alternative approaches that can eliminate the need for such credit. Such credit should be granted only for robust containments for which there are positive means for indication of containment integrity such as inerted and sub-atmospheric containments. The time intervals for which such credit is needed should be limited to a few hours, commensurate with the demonstrated capability of all associated equipment to perform its intended functions during this time period.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations on this matter, dated September 20, 2005. The Committee recommended that the RG be revised to include restrictions as discussed above before it is released for public comment.

V Possible Alternative Embrittlement Criteria to Those in 10 CFR 50.46 (Open)

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

The Committee heard a presentation from the staff concerning a new technical basis for revision of the embrittlement criteria in 10 CFR 50.46. The Committee recommended that the requirements of 10 CFR 50.46(b) concerning the coolability and geometric integrity of a reactor core during a design-basis loss-of-coolant accident (LOCA) and the aftermath of such an accident, be updated to facilitate the use of better reactor materials and improved understanding of phenomena and processes that affect core integrity and core coolability.

It also recommended that the updated requirements be written at a high level so that they are as technology-neutral and materials-neutral as practicable. Methods acceptable to the staff for demonstrating that specific cladding materials meet the high-level requirements of the regulations should be described in regulatory guides.

Finally, the process developed by the staff for the qualification of zirconium alloy cladding provides a basis for a regulatory guide for such materials. The research needed to validate this process should be completed.

Committee Action

The Committee recommended that the staff continue to move forward with the establishment of revised criteria for fuel performance during LOCAs, and should write the requirements at a high level, with specific guidance provided in regulatory guides. It should also continue to perform the necessary research to validate its proposed process.

VI. Draft Final Updates to License Renewal Guidance Documents (Open)

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

The Committee met with representatives of the NRC staff and NEI to discuss the draft final revisions to generic license renewal guidance documents. The staff revised NUREG-1800 (Standard Review Plan for License Renewal Applications for Nuclear Power Plants), NUREG-1801 (Generic Aging Lessons Learned (GALL) Report), and RG 1.188 (Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses). These documents will be placed in ADAMS (the NRC's electronic filing system) and on the NRC webpage by September 30, 2005. The staff has also drafted NUREG-1832 (Analysis of Public Comments on the Revised License Renewal Guidance Documents) and NUREG-1833 (Technical Bases for Revision to the License Renewal Guidance Documents) to support these revisions. These NUREGs will be available by October 31, 2005. NUREG-1832 contains an appendix that compares aging management review (AMR) line items from the January 2005 and September 2005 versions of the GALL Report. Comments made by ACRS Members are addressed in Appendix B of NUREG-1832. New aging management programs include the one-time inspection of ASME Code Class 1 Small-Bore Piping Program, the External Surfaces Monitoring Program, the Flux Thimble Tube Inspection Program, the Metal-Enclosed Bus Program, and the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. The proposed revision to RG 1.188 in January 2005 endorsed NEI 95-10 Revision 5 (Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule) with two exceptions. These exceptions are related to alternative scoping of non-safety-related piping and supports, and the use of short term exposure to leakage in determining the need for aging management. The draft final revision to RG 1.188 fully endorses NEI 95-10, Revision 6. The license renewal guidance documents consider comments from stakeholders and reflect the staff's current position. The staff concluded that the new documents will increase the efficiency, effectiveness, and consistence of the license renewal review process.

NEI complemented the staff on its effort to update the license renewal guidance documents and provided additional comments on two of the new AMPs. XI.E4 (Metal Enclosed Bus) recommends testing of bolted connections in metal enclosed buses every 10 years. As an alternative to testing visual inspections may be performed every five years. NEI questioned the basis for the five-year visual inspection interval. NEI also requested an opportunity to discuss, with the staff, the aging effects addressed by XI.E6 (Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements).

Committee Action

The Committee issued a report to the NRC Chairman regarding this matter, dated September 22, 2005, recommending that the draft final revisions to the generic license renewal guidance documents be approved and that the staff should continue to evaluate the need for revisions to the guidance documents in order to maintain them current.

VII. Meeting with the EDO, Deputy EDOs, and NRC Program Office Directors (Open)

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

The Committee met with the EDO, Deputy EDOs, and Office Directors of Nuclear Reactor Regulation (NRR), Nuclear Regulatory Research (RES), and Nuclear Security and Incident Response (NSIR) to discuss items of mutual interest which included license renewal, new reactors, power uprate issues, fire protection, PWR sump performance issues and emergency preparedness. The EDO and the ACRS Chairman agreed that the meeting was useful and that holding such meetings in the future would be mutually beneficial.

VIII. Interim Results of the Quality Assessment of Selected NRC Research Projects (Open)

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

The NRC Strategic Plan that was developed in accordance with the requirements of the Government Performance and Results Act (GPRA) requires that RES have an independent evaluation of the quality of its research programs. The Committee has agreed to assist RES in assessing the quality of selected research projects. During the September 8-10, 2005 ACRS meeting, the Committee discussed the interim results of the cognizant ACRS panel's quality assessment of the NRC research projects on Standardized Plant Analysis Risk (SPAR) Models Development Program; Steam Generator Tube Integrity Program at the Argonne National Laboratory; and the Thermal Hydraulic Test Program at the Penn State University.

Committee Action

The Committee plans to discuss the draft report on ACRS assessment of the quality of the selected NRC research projects during its October 6-8, 2005 meeting.

X. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

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

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

- The Committee considered the EDO's response of July 15, 2005, to comments and recommendations included in the ACRS interim letter dated June 9, 2005, concerning the safety aspects of the license renewal application for the Point Beach Nuclear Plant, Units 1 and 2. The Committee decided that it was satisfied with the EDO's response.

  The staff committed to performing a followup inspection under Inspection Procedure (IP) 71002. If the license is renewed, the staff will also conduct a post-approval site inspection for license renewal in accordance with IP 71003 before the period of extended operation begins.

- The Committee considered the EDO's response of July 8, 2005, to the ACRS memorandum dated June 3, 2005. The Committee received a letter from Roger Stoller, Chairman of the American Society for Testing and Materials Committee on Nuclear Technology, expressing his concerns about the lack of NRC staff participation in ASTM standards development activities. The Committee forwarded this letter to the EDO and requested a response regarding how these concerns would be addressed.

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

- The Committee considered the EDO's response of August 17, 2005, to comments and recommendations included in the ACRS letter dated May 13, 2005, concerning report on the safety aspects of the license renewal application for the Arkansas Nuclear One, Unit 2.

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

- The Committee considered the EDO's response of August 12, 2005, to comments and recommendations included in the ACRS interim letter dated June 14, 2005, concerning the staff's draft safety evaluation report on the Grand Gulf early site permit application.

  The Committee decided that it was not completely satisfied with the EDO's response regarding the exposition on threats posed by transportation accidents on the river adjacent to the proposed site, and the prognostication of the weather over the next 65 years based just on historical frequencies of severe weather events. The Committee plans to continue its discussion of this matter with the staff during its review of the final SER.

  The staff has committed to discuss the Committee's comments and concerns during the ACRS review of the final SER.

- The Committee considered the EDO's response of August 2, 2005, to comments and recommendations included in the ACRS letter dated June 14, 2005, concerning the risk-informed, performance-based, fire protection regulatory guide. The Committee decided that it was satisfied with the EDO's response.

  The staff committed to revise the RG to address majority of the ACRS comments and recommendations and provide the draft final RG and industry guidance document for ACRS review.

- The Committee considered the EDO's response of July 20, 2005, to comments and recommendations included in the ACRS letter dated June 10, 2005, concerning the draft final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities." The Committee decided that it was satisfied with the EDO's response.

  The staff has committed to further identify, quantify, and document the uncertainties associated with fire probabilistic risk analyses. Also, if new insights are gained, the staff committed to consider the need to revise NUREG/CR-6850.

- The Committee considered the EDO's response of July 8, 2005, to comments and recommendations included in the ACRS Report dated June 10, 2005, concerning a draft Commission Paper, "Risk-Informed Alternatives to the Single Failure Criterion."

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to discuss further developments in this area as part of its review of the program plan being developed for a risk-informed, performance-based revision to 10 CFR Part 50.

- The Committee considered the EDO's response of September 1, 2005, to comments and conclusions included in the ACRS report dated July 18, 2005, concerning the Dominion Nuclear North Anna, LLC, ESP application and the associated NRC final safety evaluation report. The Committee decided that it was satisfied with the EDO's response.

The staff committed to interact with the ACRS starting in FY 2006 and to identify issues in the ESP Review Standard that should be modified for the planned revisions to the SRP. The staff committed to interact with the ACRS in developing lessons learned to improve and streamline the ESP process for future applications.

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 7, 2005. 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 2005 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

During this session, the Subcommittee also discussed and developed recommendations on items requiring future Committee action.

Meeting with the NRC Commissioners

The ACRS is scheduled to meet with the NRC Commissioners between 1:30 and 3:30 p.m. on Thursday, December 8, 2005 to discuss items of mutual interest. The Committee approved the list of topics as follows:

- Policy Issues Related to New Plant Licensing (TSK)
- Risk-Informed Alternatives to the Single Failure Criterion (WJS)
- Early Site Permits (DAP)
- License Renewal/Power Uprates (MVB)
- Proposed Alternative Embrittlement Criteria (DAP)
- Fire Protection Matters (GEA/RSD)

Proposed ACRS Meeting Dates fo CY 2006

Proposed ACRS meeting dates for CY 2006 were discussed. The members should review these dates and provide comments. The Planning and Procedures Subcommittee will resolve any comments received from the members during its October 5, 2005 meeting and submit a revised set of dates for approval by the Committee during this meeting.

ACRS Retreat in 2006

The Committee will hold a retreat in January 2006. The Committee should select the dates and location for the retreat. The members should propose topics for the retreat by September 23, 2005. The Subcommittee will discuss the proposed topics during its October 5, 2005 meeting and provide a list to the full Committee for consideration during this meeting.

Quadripartite Meeting Status

The Quadripartite Meeting is scheduled to be held on October 18-20, 2006 at the Jurys Washington Hotel in Washington, D.C. The meeting agenda is being finalized. An agreement from the Quadripartite Member Countries has been met. In addition, responses from colleagues in Switzerland and Sweden to Chair a Breakout Session is anticipated. Specific topics have been assigned to ACRS members and an abstract of the presentation is due in February 2006.

Member Countries will be sharing their country-specific experience and perspectives with the entire audience on three main topics:

- Technical Topics of Interest to Members in the Past Four Years
- Technology Advances and Changes to Regulatory Approach
- Response to Significant Operations Events

Additionally, several specific technical topics are planned for the three Discussion Sessions and the eight Breakout Sessions (50 minutes each). A specific Member Country is assigned as the Chair for each session. The Chair will open the session with brief remarks. All other Member Countries will be provided an opportunity to make a brief presentation on the same topic. For Breakout Sessions the Chair may have 10 minutes, while the other Countries each are limited to 5 minutes for their presentations. This approach leaves approximately 25 minutes for group discussion and questions/answers following the presentations.

Next steps include inviting key note speakers, identifying guests for the evening events, and selecting translators for the Japanese and the French.

Follow-Up Items Resulting from Browns Ferry Site Visit and Meeting with Region II Personnel

Several ACRS members and staff visited the Browns Ferry Nuclear Plant, Units 1, 2, and 3, on August 23 and held a meeting with Region II personnel on August 24 and 25, 2005. Additionally, Drs. Wallis, Powers, and Kress met with representatives of INPO in Atlanta to discuss various INPO programs, particularly their Safety Culture assessment activities. Any follow-up items resulting from this plant visit and meeting should be discussed and a course of action should be developed for addressing them.

Candidates for Potential Membership on the ACRS

On June 28, 2005, the Screening Panel met to discuss 42 applications received in response to the solicitation for the current vacancies on the ACRS. The Panel selected six applicants in the areas of plant operations and materials and metallurgy to interview during the September 2005 full Committee meeting. The Screening Panel will continue to look for qualified candidates to fill the vacancy on the Committee in the thermal-hydraulics area. Current planning is to submit a slate of candidates in the areas of plan operations and materials and metallurgy to the Commission in September 2005 for consideration.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 526th ACRS Meeting, October 6-8, 2005.

The 525th ACRS meeting was adjourned at 1:00 p.m. on September 10, 2005.

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

November 21, 2005

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

FROM: Graham B. Wallis
ACRS Chairman

Graham B. Wallis

SUBJECT: CERTIFIED MINUTES OF THE 525th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), SEPTEMBER 8-10, 2005

I certify that based on my review of the minutes from the 525TH 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.

prior to the meeting to be advised of any potential changes to the agenda.

Dated: August 10, 2005.

Michael L. Scott,

Branch Chief, ACRS/ACNW.

[FR Doc. E5–4485 Filed 8–16–05; 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 8–10, 2005, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Wednesday, November 24, 2004 (69 FR 68412).

Thursday, September 8, 2005, 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.–9:45 a.m.: Final Review of the License Renewal Application for Millstone Power Station, Units 2 and 3 (Open)—The Committee will hear presentations by and hold discussions with representatives of the Dominion Nuclear Connecticut, Inc. and the NRC staff regarding the license renewal application for Millstone Power Station, Units 2 and 3 and the associated Final Safety Evaluation Report prepared by the NRC staff.

10 a.m.–12 Noon: Interim Review of the Exelon/Clinton Early Site Permit Application (Open)—The Committee will hear presentations by and hold discussions with representatives of the Exelon Generation Company, LLC and the NRC staff regarding the Clinton early site permit application and the associated Draft Safety Evaluation Report prepared by the NRC staff.

1:30 p.m.–3:30 p.m.: Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding proposed Revision 4 to Regulatory Guide 1.82 and the supporting Standard Review Plan, Section 6.2.2, "Containment Heat Removal Systems," related to emergency core cooling system net positive suction head (NPSH) and the use of containment overpressure credit in calculating NPSH.

3:45 p.m.–5:45 p.m.: Possible Alternative Embrittlement Criteria to Those in 10 CFR 50.46 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff, Electric Power Research Institute, and Framatome regarding possible alternative embrittlement criteria to those in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and related 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 as well as a proposed report on policy issues related to new plant licensing.

Friday, September 9, 2005, 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.–9:45 a.m.: Draft Final Updates to License Renewal Guidance Documents (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding draft final updates to NUREG–1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," NUREG–1801, Revision 1, "Generic Aging Lessons Learned (GALL) Report," Regulatory Guide 1.188, Revision 1, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," and NEI 95–10, Revision 6, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54—The License Renewal Rule," which is endorsed by Regulatory Guide 1.188.

10 a.m.–12 Noon: Meeting with the EDO, Deputy EDOs, and NRC Program Office Directors (Open)—The Committee will hear presentations by and hold discussions with the NRC Executive Director for Operations (EDO), Deputy EDOs, Office Directors of Nuclear Reactor Regulation, Nuclear Regulatory Research, and Nuclear Material Safety and Safeguards regarding items of mutual interest.

1:30 p.m.–2:30 p.m.: Interim Results of the Quality Assessment of Selected NRC Research Projects (Open)—The Committee will discuss the interim results of the cognizant ACRS panel's quality assessment of the NRC research projects on: Standardized Plant Analysis Risk (SPAR) Models Development Program; Steam Generator Tube Integrity Program at the Argonne National Laboratory; and the Thermal-Hydraulic Test Program at the Penn State University.

2:30 p.m.–3:15 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.

3:15 p.m.–3:30 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the EDO to comments and recommendations included in recent ACRS reports and letters.

3:45 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, September 10, 2005, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

3 p.m.—3:30 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 5, 2004 (69 FR 59620). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Cognizant ACRS staff (301–415–7364), between 7:30 a.m. and 4:15 p.m., e.t.

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., e.t., 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 11, 2005

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E5–4486 Filed 8–16–05; 8:45 am]

BILLING CODE 7590–01–P

OVERSEAS PRIVATE INVESTMENT CORPORATION

September 8, 2005 Public Hearing

Time and Date: 2 p.m., Thursday, September 8, 2005.

Place: Offices of the Corporation, Twelfth Floor Board Room, 1100 New York Avenue, NW., Washington, DC.

Status: Hearing Open to the Public at 2 p.m.

Purpose: Public Hearing in conjunction with each meeting of OPIC's Board of Directors, to afford an opportunity for any person to present views regarding the activities of the Corporation.

Procedures:

Individuals wishing to address the hearing orally must provide advance notice to OPIC's Corporate Secretary no later than 5 p.m., Wednesday, August 31, 2005. The notice must include the individual's name, title, organization, address, and telephone number, and a concise summary of the subject matter to be presented.

Oral presentations may not exceed ten (10) minutes. The time for individual presentations may be reduced proportionately, if necessary, to afford all participants who have submitted a timely request to participate in an opportunity to be heard.

Participants wishing to submit a written statement for the record must submit a copy of such statement to OPIC's Corporate Secretary no later than 5 p.m. Wednesday, August 31, 2005. Such statements must be typewritten, double-spaced, and may not exceed twenty-five (25) pages.

Upon receipt of the required notice, OPIC will prepare an agenda for the hearing identifying speakers, setting forth the subject on which each participant will speak, and the time allotted for each presentation. The agenda will be available at the hearing.

A written summary of the hearing will be compiled, and such summary will be made available, upon written request to OPIC's Corporate Secretary, at the cost of reproduction.

Contact Person For Information: Information on the hearing may be obtained from Connie M. Downs at (202) 336–8438, via facsimile at (202) 218–0136, or via e-mail at *cdown@opic.gov*.

Dated: August 15, 2005.

Connie M. Downs,

OPIC Corporate Secretary.

[FR Doc. 05–16380 Filed 8–15–05; 12:07 pm]

BILLING CODE 3210–01–M

SECURITIES AND EXCHANGE COMMISSION

[Investment Company Act Release No. 27027; 812–13026]

AXP California Tax-Exempt Trust, et al.; Notice of Application

August 11, 2005.

AGENCY: Securities and Exchange Commission ("Commission").

ACTION: Notice of application for an order under section 12(d)(1)(J) of the Investment Company Act of 1940 ("Act") for an exemption from sections 12(d)(1)(A) and (B) of the Act, under sections 6(c) and 17(b) of the Act for an exemption from section 17(a) of the Act, and under section 17(d) of the Act and rule 17d–1 under the Act to permit certain joint transactions.

Applicants: AXP California Tax-Exempt Trust, AXP Dimensions Series, Inc., AXP Discovery Series, Inc., AXP Equity Series, Inc., AXP Fixed Income Series, Inc., AXP Global Series, Inc., AXP Government Income Series, Inc., AXP Growth Series, Inc., AXP High Yield Income Series, Inc., AXP High Yield Tax-Exempt Series, Inc., AXP Income Series, Inc., AXP International Series, Inc., AXP Investment Series, Inc., AXP Managed Series, Inc., AXP Market Advantage Series, Inc., AXP Money Market Series, Inc., AXP Partners International Series, Inc., AXP Partners Series, Inc., AXP Sector Series, Inc., AXP Selected Series, Inc., AXP Special Tax-Exempt Series Trust, AXP Stock Series, Inc., AXP Strategy Series, Inc., AXP Tax-Exempt Series, Inc., AXP Tax-Free Money Series, Inc. (together, the "AXP Funds"), AXP Variable Portfolio-Income Series, Inc., AXP Variable Portfolio-Investment Series, Inc., AXP Variable Portfolio-Managed Series, Inc., AXP Variable Portfolio-Money Market Series, Inc., AXP Variable Portfolio-Partners Series, Inc., AXP Variable Portfolio-Select Series, Inc. (these six entities together, the "Variable Portfolio Funds"), Growth Trust, Growth and Income Trust, Income Trust, Tax-Free Income Trust, World Trust (these five entities together, the "Master Trusts") and Ameriprise Financial, Inc., formerly known as American Express Financial Corporation ("AFI", and together with the AXP Funds, the Variable Portfolio Funds and the Master Trusts, the "Applicants").[1]

[1] Applicants request that any relief granted also apply to (i) any existing or future registered management investment companies and their series that are part of the same "group of investment companies" as defined in section 12(d)(1)(G) of the Act and for which AFI or a person controlling, controlled by or under common control (within the meaning of section 2(a)(9) of the Act) with AFI (each, an "Adviser") serves as investment adviser ("Registered Funds") and (ii) any existing or future unregistered entities for which an Adviser serves as investment adviser, trustee, managing member or general partner exercising investment discretion, and which are excepted from the definition of investment company pursuant to section 3(c)(1) or section 3(c)(7) of the Act ("Unregistered Funds"), qualified employee benefit plans, trusts, institutional accounts, bank common funds and bank collective trusts (within the meaning of section 3(c)(11) of the Act) that are not investment companies as defined in the Act ("Other Institutional Clients", and together with the

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

August 2, 2005

SCHEDULE AND OUTLINE FOR DISCUSSION
525th ACRS MEETING
SEPTEMBER 8-10, 2005

THURSDAY, SEPTEMBER 8, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - ~~9:45~~ 10:00 A.M. Final Review of the License Renewal Application for Millstone Power Station, Units 2 and 3 (Open) (JDS/JGL/CS)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the Dominion Nuclear Connecticut, Inc. and the NRC staff regarding the license renewal application for Millstone Power Station, Units 2 and 3 and the associated Final Safety Evaluation Report prepared by the NRC staff.

10:00-10:15
~~9:45 - 10:00~~ A.M. ***BREAK***

3) 10:00 - 12:00 Noon Interim Review of the Exelon/Clinton Early Site Permit Application (Open) (DAP/MME)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the Exelon Generation Company, LLC and the NRC staff regarding the Clinton early site permit application and the associated Draft Safety Evaluation Report prepared by the NRC staff.

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

4) 1:30 - 3:30 P.M. Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Open) (VHR/GBW/RC)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the NRC staff regarding proposed Revision 4 to Regulatory Guide 1.82 and the supporting Standard Review Plan, Section 6.2.2, "Containment Heat Removal Systems," related to emergency core cooling system net positive suction head (NPSH) and the use of containment overpressure credit in calculating NPSH.

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

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

5) 3:45 - 5:45 P.M. Possible Alternative Embrittlement Criteria to Those in 10 CFR 50.46 (Open) (DAP/RC)

- 5.1) Remarks by the Subcommittee Chairman
- 5.2) Briefing by and discussions with representatives of the NRC staff, Electric Power Research Institute, and Framatome regarding possible alternative embrittlement criteria to those in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and related matters.

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

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

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

Discussion of proposed ACRS reports on:

- 6.1) Final Review of the License Renewal Application for Millstone Power Station, Units 2 and 3 (JDS/JGL/CS)
- 6.2) Interim Review of the Exelon/Clinton Early Site Permit Application (DAP/MME)
- 6.3) Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (VHR/GBW/RC)
- 6.4) Possible Alternative Embrittlement Criteria in 10 CFR 50.46 (DAP/RC)
- 6.5) Policy Issues Related to New Plant Licensing (TSK/MME)

FRIDAY, SEPTEMBER 9, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - ~~9:45~~ 9:30 A.M. Draft Final Updates to License Renewal Guidance Documents (Open) (MVB/CS)

- 8.1) Remarks by the Subcommittee Chairman
- 8.2) Briefing by and discussions with representatives of the NRC staff regarding draft final updates to NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," NUREG-1801, Revision 1, "Generic Aging Lessons Learned (GALL) Report," Regulatory Guide 1.188, Revision 1 "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," and NEI 95-10,

Revision 6, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," which is endorsed by Regulatory Guide 1.188, Revision 1.

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

9:30
~~9:45~~ - 10:00 A.M. ***BREAK***

11:45
9) 10:00 - ~~12:00~~ Noon — Meeting with the EDO, Deputy EDOs, and NRC Program Office Directors (Open) (GBW/JTL/MLS)

- 9.1) Remarks by the ACRS Chairman
- 9.2) Briefing by and discussions with the NRC Executive Director for Operations (EDO), Deputy EDOs, and Office Directors of Nuclear Reactor Regulation, Nuclear Regulatory Research, and Nuclear Material Safety and Safeguards regarding items of mutual interest.

11:45
~~12:00~~ - 1:30 P.M. ***LUNCH***

2:40
10) 1:30 - ~~2:30~~ P.M. — Interim Results of the Quality Assessment of Selected NRC Research Projects (Open) (DAP/GEA/JDS/GBW/EAT/CS/RC)
Discussion of the interim results of the cognizant ACRS panel's quality assessment of the NRC research projects on: Standardized Plant Analysis Risk (SPAR) Models Development Program; Steam Generator Tube Integrity Program at the Argonne National Laboratory; and the Thermal- Hydraulic Test Program at the Penn State University.

3:00- 3:50
11) ~~2:30~~ - ~~3:15~~ P.M. — Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/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.

2:40 - 3:00
12) ~~3:15~~ - ~~3:30~~ P.M. — Reconciliation of ACRS Comments and Recommendations (Open) (GBW, 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.

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

13) 3:45 - ~~7:00~~ 7:30 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
13.1) Final Review of the License Renewal Application for Millstone Power Station, Units 2 and 3 (JDS/JGL/CS)
13.2) Interim Review of the Exelon/Clinton Early Site Permit Application (DAP/MME)
13.3) Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (VHR/GBW/RC)
13.4) Possible Alternative Embrittlement Criteria in 10 CFR 50.46 (DAP/RC)
13.5) Policy Issues Related to New Plant Licensing (TSK/MME)
13.6) Draft Final Updates to License Renewal Guidance Documents (MVB/CS)

SATURDAY, SEPTEMBER 10, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 3:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 13.

15) 3:00 - 3:30 P.M. Miscellaneous (Open) (GBW/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

525TH ACRS MEETING
SEPTEMBER 8-10, 2005

NRC STAFF (September 8, 2005)

L. Dudes, NRR	S. Ng, NRR	R. McNally, NRR
Y. Li, NRR	R. Li, NRR	T. Ford, NRR
T. Cheng, NRR	O. Yee, NRR	S. Lee, NRR
S. Turk, OGC	J. Eads, NRR	B. Elliot, NRR
P. Clifford, NRR	Y. Diaz, NRR	F. Saba, NRR
M. Stutzke, NRR	K. Hsu, NRR	J. Segal, NRR
R. Emch, NRR	K. Alm-Lytz, NRR	R. Moody, NSIR
M. Kotzalas, NRR	K. Weaver, NRR	B. Harvey, NRR
D. Allison, NRR	J. Zimmerman, NRR	Q. Gan, NRR
T. Hafer, NRR	V. Rodriguez, NRR	R. Pettis, NRR
L. Marsh, NRR	R. Aluck, NRR	R. Anand, NRR
M. Kowal, NRR	J. Honcharik, NRR	N. Patel, NRR
T. Alexion, NRR	A. Keim, NRR	W. Beckman, NRR
C. Holden, NRR	T. Le, NRR	G. Werner, OCM
S. Lu, NRR	J. Medoff, NRR	J. Voglewede, RES
C. Jackson, OCM,	S. Hoffman, NRR	M. Snell, RES
R. Architzel, NRR	J. Ma, NRR	
D. Roberts, NRR	R. Subbaratman, NRR	
M. Rubin, NRR	J. Hylach, NRR	

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

D. Lochbaum, UCS	M. Kray, Exelon	M. Nissley, W
P. Aitken, Dominion	A. Cornell, CACCO	R. Schoff, W
C. Sorrell, Dominion	T. Mundy, Exelon	G. Cliefton, NEI
G. Komosky, Dominion	M. Cambrie, Warley Parsons	J. Traifors, Link
T. Hendy, Dominion	E. Grant, Exelon	R. Yang, EPRI
J. Knorr, NMC	S. Frantz, Morgan Lewis	M. Billone, ANL
D. Mitchell, W	B. Maher, Exelon	G. Swindlehurst, Duke Power
M. Hotchkiss, Dominion	R. Wells, Parallex	J. Potts, GNF
B. Watson, Dominion	B. Hoffman, Public Citizen	R. Reynolds, AREVA
R. Rucker, First Energy	T. Yamada, JNES	J. Holm, Framatome
D. Kunsemiller, FENOC	R. Bell, NEI	S. Dolloy, Inside NRC/PLATTS
K. Hanson, Geomtry Consultants	J. Weil, McGraw-Hill	
R. Youngs, Geomtry Consultants	V. Nicekani, IAC	
R. Kennedy, RPK Struct. Mectories	S. Hoffman, VT DPS	
D. Anderson, Chzm Hill	D. Rosinski, PWSP	
J. Ioannidi, Worky Parsons	W. Sherman, VY DPS	
B. Holcomb, Chzm Hill	R. Heck, W	
C. Stepp, EHS		

APPENDIX III: MEETING ATTENDEES

525TH ACRS MEETING
SEPTEMBER 8-10, 2005

NRC STAFF (September 9, 2005)

M. Drouin, RES
R. Sulbandue, NRR
S. Lee, NRR
D. Barss, NSIR
R. Blount, NSIR
S. Weerakkod, NRR
G. Rhee, RES
D. Marksberry, RES
D. Rasmuson, RES
F. Eltawila, RES
J. Dozier, NRR
A. Hull, NRR
D. Nguyen, NRR
A. Pal, NRR
K. Alm-Lytz, NRR
J. Zimmerman, NRR
R. Dipert, NRR
G. Georgiev, NRR
P. Kang, RES
A. Wilson, RES
K. Chang, NRR
K. Hsu, NRR
L. Tran, NRR
H. Asher, NRR
R. Aluck, NRR
B. Elliot, NRR
L. Lund, NRR
A. Keim, NRR
S. Lee, NRR
R. Jenkins, NRR
G. Gallati, NRR
D. Terao, NRR
J. Wermiel, NRR
C. Paperiello, RES
T. Alexion, NRR
B. Borchardt, NRR
J. Hannon, NRR
J. Hopkins, NRR
E. McKenna, NRR
F. Gillespie, NRR
T. Liu, NRR
T. Scarbrough, NRR
C. Grimes, NRR
G. Suber, NRR
J. Wiggins, RES
B. Boger, NRR
V. Rodriguez, NRR
Y. Diaz, NRR
M. Lintz, NRR
J. Dixon-Herrity, OEDO
A. Gody, OEDO

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
R. Rucker, First Energy
E. Patel, Parallax
C. Willbanks, ATL Intl
M. Detamore, PPL
A. Baione, Parallax
D. Kunsemitter, FENOC
M. Heath, Progress Energy
J. Hinze, Intl Access Corp.
V. Nilekani, IAC
C. Myer, Southern Nuclear
A. Marion, NEI
J. Ross, NEI
K. Nakamote, JNES
D. Kosloff, FENOC
S. Traiforos, LINK
W. Sherman, State of VT

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC (September 10, 2005)
S. Hoffman, VT DPS
W. Sherman, VT DPS
J. Silberg, Pittsburgh Winthrop Shaw Pittman

APPENDIX IV: FUTURE AGENDA

September 15, 2005

SCHEDULE AND OUTLINE FOR DISCUSSION
526th ACRS MEETING
OCTOBER 6-8, 2005

THURSDAY, OCTOBER 6, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:00 A.M. Interim Review of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Open) (MVB/CS)
 2.1) Remarks by the Subcommittee Chairman
 2.2) Briefing by and discussions with representatives of the Tennessee Valley Authority and the NRC staff regarding the license renewal application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 and the NRC staff's Safety Evaluation Report with Open Items.

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

3) 10:15 - 11:45 A.M. Proposed Recommendations for Resolving Generic Safety Issue (GSI)-80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of Boiling Water Reactor Mark I and II Containments" (Open) (JDS/JGL)
 3.1) Remarks by the Subcommittee Chairman
 3.2) Briefing by and discussions with representatives of the NRC staff regarding the recommendations proposed by the NRC Office of Nuclear Regulatory Research for resolving GSI-80.

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

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

4) 12:45 - 2:15 P.M. Resolution of ACRS Comments on the Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants" (Open) (GEA/JGL)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding the changes made to this Guide and to NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)," in response to the ACRS comments and recommendations included in its June 14, 2005 letter.

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

5) 2:30 - 4:00 P.M. Davis-Besse Reactor Pressure Vessel Head Integrity Calculations (Open) (JDS/EAT)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the expert elicitation and calculations performed for the reactor pressure vessel head integrity of the Davis-Besse Nuclear Power Plant.

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

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

6) 4:15 -5:15 P.M. Quality Assessment of the Selected NRC Research Program (Open) (DAP/GEA/JDS/GBW/HPN/EAT/CS/RC)

6.1) Remarks by the Subcommittee Chairman

6.2) Discussion of the results of the cognizant ACRS panel's assessment of the quality of the NRC research projects on: Standardized Plant Analysis Risk (SPAR) Models Development Program; Steam Generator Tube Integrity Program at the Argonne National Laboratory; and the Thermal-Hydraulic Test Program at the Penn State University.

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

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

7.1) Interim Review of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (MVB/CS)

7.2) Proposed Recommendations for Resolving GSI-80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of Boiling Water Reactor Mark I and II Containments" (JDS/JGL)
7.3) Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants" (GEA/JGL)

FRIDAY, OCTOBER 7, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 - 10:00 A.M. Licensees' Responses to the Bulletin on, "Emergency Preparedness and Response Actions for Security-Based Events" (Open/Closed) (MVB/EAT)
9.1) Remarks by the Subcommittee Chairman
9.2) Briefing by and discussions with representatives of the NRC staff regarding licensees' responses to the Bulletin related to emergency Preparedness and Response Actions for Security-Based Events.

[NOTE: A portion of this session may be closed to protect information classified as national security and safeguards information pursuant to 5 U.S.C. 552b(c) (1) and (3)].

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

10) 10:15 - 11:15 A.M. NRC Staff's Response to the ACRS Letter on the Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Open) (VHR/GBW/RC)
10.1) Remarks by the Subcommittee Chairman
10.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's response to the ACRS letter on the Proposed Revision 4 to Regulatory Guide 1.82.

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

11) 11:15 - 12:15 P.M. <u>Format and Content of the NRC Safety Research Program Report to the Commission</u> (Open) (DAP/HPN/SD)
Report by and discussions with the Chairman of the ACRS Subcommittee on Safety Research Program regarding format and content of the ACRS report to the Commission on the NRC Safety Research Program as well as assignments for the ACRS members.

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

12) 1:15 - 2:15 P.M. <u>Future ACRS Activities/Report of the Planning and Procedures Subcommittee</u> (Open) (GBW/JTL/SD)

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

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

13) 2:15 - 2:30 P.M. <u>Reconciliation of ACRS Comments and Recommendations</u> (Open) (GBW, 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.

14) 2:30 - 3:00 P.M. <u>Subcommittee Report</u> (Open) (JDS/MVB/JGL)
Report by and discussions with the Chairmen of the ACRS Subcommittees on Plant Operations and Plant License Renewal regarding matters discussed at the September 21, 2005 Subcommittee meeting.

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

15) 3:15 - 7:00 P.M. <u>Preparation of ACRS Reports</u> (Open)
Discussion of proposed ACRS reports on:

15.1) Interim Review of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (MVB/CS)

15.2) Proposed Recommendations for Resolving GSI-80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of Boiling Water Reactor Mark I and II Containments" (JDS/JGL)

15.3) Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants" (GEA/JGL)
15.4) Quality Assessment of Selected NRC Research Projects (DAP/GEA/JDS/GBW/HPN/CS/RC)

SATURDAY, OCTOBER 8, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

16) 8:30 - 12:00 Noon Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 15.

17) 12:00 - 12:30 P.M. Miscellaneous (Open) (GBW/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
525th ACRS MEETING
SEPTEMBER 8-10, 2005

[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

- Opening Remarks by the ACRS Chairman

  1. Items of Interest, dated September 8-10, 2005

2. Final Review of the License Renewal Application for Millstone Power Station, Units 2 and 3
   2. Millstone Units 2 and 3 License Renewal Presentation to the ACRS presentation by B. Watson, MPS LR Supervisor, Dominion Nuclear Connecticut [Viewgraphs]
   3. Millstone Power Station Units 2 and 3 License Renewal Safety Evaluation Report, presentation by J. Eads, Sr. Project Manager, NRR [Viewgraphs]
   4. Millstone 2, 2Q/2005 Plant Inspection Findings, Initiating Events

3. Interim Review of the Exelon/Clinton Early Site Permit Application
   5. Early Site Permit Application, Clinton Power Station Site, Draft Safety Evaluation Report presentation by Exelon [Viewgraphs]
   6. Exelon Early Site Permit Safety Review Status presentation by J. Segala, Sr. Project Manager, NRR

4. Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident"

5. Possible Alternative Embrittlement Criteria to Those in 10 CFR 50.46
   7. Proposed Revisions to Generic License Renewal Guidance Documents presentation by J. Dozier and A. Hull, NRR [Viewgraphs]

8. Draft Final Updates to License Renewal Guidance Documents

9. Meeting with the EDO, Deputy EDOs, and NRC Program Office Directors
   8. ACRS Meeting with EDO and Office Directors presentation by NRR, RES, and NSIR [Viewgraphs]

10. Interim Results of the Quality Assessment of Selected NRC Research Projects

11 Future ACRS Activities/Report of the Planning and Procedures Subcommittee
   XX. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - September 7, 2005 [Handout #11.1]

12 Reconciliation of ACRS Comments and Recommendations
   XX. Reconciliation of ACRS Comments and Recommendations [Handout #12]

Appendix IV
525th ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB DOCUMENTS

2 Review of the License Renewal Application and Final Safety Evaluation Report for the Millstone Power Station, Units 2 and 3

1. Proposed Schedule
2. Letter to the EDO dated June 14, 2005, (ADAMS Accession No. ML051650432) "Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants"
3. Letter to the EDO dated July 18, 2005 (ADAMS Accession No. ML052000086) "Draft Final Regulatory Guide DG-1137, "Guidelines for Lightning Protection of Nuclear Power Plants"
4. Letter to the EDO dated June 10, 2005 (ADAMS Accession No. ML051650244) "Draft Final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities"
5. Letter to the EDO dated July 25, 2005 (ADAMS Accession No. ML052080386) "Proposed Generic Letter 2005-XX, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations"
6. Letter to the EDO dated July 7, 2005 (ADAMS Accession No. ML051940496) "Proposed Generic Letter 2005-XX, "Impact of Potentially Degraded HEMYC/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs"
7. Letter to the EDO dated July 7, 2005 (ADAMS Accession No. ML051950536) "Proposed Generic Letter 2005-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems"
8. Letter to the EDO dated April 8, 2005 (ADAMS Accession No. ML051030389) "Draft NUREG, "Verification & Validation of Selected Fire Models for Nuclear Power Plant Applications"
9. Letter to the EDO dated March 8, 2005 (ADAMS Accession No. ML050680282) "Proposed Generic Letter 2005-XX, "Grid Reliability & the Impact on Plant Risk & the Operability of Offsite Power"
10. Letter to the EDO dated February 11, 2005 (ADAMS Accession No. ML050460166) "Draft Regulatory Guide, DG-1137, "Guidelines for Lightning Protection for Nuclear Power Plants"

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4. Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident"

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21. Memorandum from R. Caruso to G. Wallis, "Status Report for the Meeting of the ACRS Subcommittee on Thermal Hydraulics," July 19-20, 2005, in Rockville, MD, July 12, 2005
22. "Draft Meeting Summary, T/H Subcommittee Meeting July 19, 2005."

5. <u>Possible Alternative Embrittlement Criteria to Those in 10 CFR 50.46</u>

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27. "Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46," R. Meyer, July 27, 2005
28. "ANL LOCA-Relevant Research Results," M. Billone, et al., July 27-29, 2005
29. "LOCA Discussion," M. Billone, et al. July 27-29, 2005
30. "LOCA Embrittlement Correlation," M. Billone, Y. Yan, July 27-29, 2005
31. "ECCS Acceptance Criteria - Overview of Industry Concerns," O. Ozer, July 27, 2005
32. "Impact of the LOCA Cooling Rates and Pre-Corrosion Layer on the Cladding Residual Ductility: Main Learnings [sic] from the Recent CEA Experiments," J-P, Mardon, July 27, 2005
33. "Evaluation of ANL Experimental Results and Summary of Key Questions," R. Montgomery, July 27-28, 2005
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40. Generic Aging Lessons Learned (GALL) Report Tabulation of Results, NUREG-1801, Vol. 2, Rev. 1, September 2005
41. Regulatory Guide 1.188, Rev. 1, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," August 2005
42. Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, NUREG-1800, Rev. 1, September 2005
43. Technical Bases for Revision to the License Renewal Guidance Documents, Draft NUREG-1833, September 2005
44. Analysis of Public Documents on the Revised License Renewal Guidance Documents, Draft NUREG-1832, September 2005

45. Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, NEI, 95-10, Rev. 6, June 2005

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On Yee	
~~Jim Knorr~~	~~NMC LLC Point~~
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Tom Hendy	Dominion
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JUAN AYALA	" " "
Richard McNally	NRR/DE/EMEB
Tanya Ford	NRR/DSSA/SRXB
SAMSON LEE	NRR/DRIP/RLEP
BARRY ELLIOT	NRR/DE/EMCB
Farideh Saba	NRR/DLPM/DP2-1
~~John [illegible]~~	
John Segala	NRR/DRIP/RNRP
Robert Moody	NSIR/DPR/EPD
Brad Harvey	NRR/DSSA/SPSB-C
Qi Gan	NRR/DRIP/RLEP
RPETTIS	NRR
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Bill Watson	Dominion
Roger Rucker	First Energy
David Kunsemiller	FENOC Beaver Valley
[illegible]	
~~John M~~	
Kathryn L Hanson	Geomatrix Consultants
Robert Youngs	Geomatrix Consultants
Robert Kennedy	RPK Struct. Mechanics
Don Anderson	CH2M HILL
John Ioannidi	Worley Parsons
Bernie Holcomb	CH2M HILL
Carl C Stepp	EHS
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Allin Cornell	CACCo consultant to Exelon
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~~[illegible]~~	
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Russ Bell	NEI
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VIJAY M NILEKANI	IAC.
~~[illegible]~~	
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Doug Rosinski	PWSP
William Sherman	VT Dept of Pub Service
[illegible]	Westinghouse
Mitch Nissley	Westinghouse
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AMAR PAL	NRR/DE/EEIB
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Jake Zimmerman	NRR/DRIP/RLEP
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[illegible] J. Kang	NRR/MEB/ATS.
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525th ACRS MEETING

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No. S-05-010

"Stacked Up Like LaGuardia"

The Honorable Jeffrey S. Merrifield
Commissioner
U.S. Nuclear Regulatory Commission
at
American Nuclear Society
2005 Utility Working Conference
Amelia Island, Florida
August 8, 2005

Introduction

Good morning ladies and gentlemen. We begin this conference at a time of great expectation: a time when the President and Congress have taken a significant step to jump start new reactor orders in the U.S., and a time when the industry is poised to enter what I have called "the emerging second great bandwagon effect." While this presages a significant chapter in the history of nuclear power, I would like to share with you this morning what I believe are some significant challenges that lie ahead for the U. S. as we enter this new phase in our nation's energy history.

As the title of my speech indicates, I am looking toward the future with some trepidation as to how the NRC and our future applicants will handle the many challenges we may face in the coming years. Today, I would like to discuss my personal views on how the NRC's licensing and budgeting process will play a critical role in the development of new reactor projects. I will also spend some of my time discussing how the anticipated turnover in the upper levels of utility management could impact the future safe operation of the industry we oversee.

Budgeting for Future Challenges

My fellow Commissioners and I have spent the last few weeks focused on reviewing and finalizing the agency's budget for Fiscal Year 2007. As with any budgeting process, the NRC must attempt an educated guess as to its resource needs almost two years into the future. Normally, we do a pretty good job playing this guessing game, but for a variety of factors, this year required more than the usual amount of guesswork. Among all the usual factors, our job was made even more difficult by the rapidly changing environment for new plant orders. This prospect was underscored by President

Bush's recent visit to Calvert Cliffs Nuclear Plant, where he outlined his strong support for the rapid growth and advancement of nuclear power. With the recent passage of the president's energy legislation, which the president will sign into law today, Congress has declared that it, too, shares his vision.

For their part, a number of utilities, either through announced or unannounced efforts, are moving toward submission of combined license applications for new reactors. Additionally, the NRC has received a growing number of inquiries from vendors that may apply for design certification in the next few years. Add to this the potential for early site permit applications, including the recent notification from Southern Company that it intends to submit an early site permit application in the summer of 2006, and it is obvious the NRC will have an extensive workload. That being said, it is incredibly challenging to translate these "ifs" and "maybes" into budgetary dollars and FTE when we have no firm application dates from which to work.

Yet, without concrete information that all this work will materialize, the Commission is in a particularly tough situation. We need to plan for the possibilities I mentioned, but cannot justify huge budgetary increases based on mere hearsay or splashy PowerPoint presentations. As a fiscal conservative who spent my early years on the Commission figuring out how to downsize our agency during a period of perceived nuclear decline, I don't want to overshoot the mark to meet what I would call the "maximum credible order scenario." Yet, in a tight budget environment also, I don't want to undershoot our request given the difficulties associated with obtaining supplemental appropriations from Congress. We will obviously be prepared to handle the few applications that we have been made aware of to date, but beyond that, I think there is some uncertainty as to how the agency would handle an unexpected bow wave of "surprise" applications for combined licenses, design certifications, or early site permits.

The "Stacking Up" Phenomenon

This unpleasant conundrum reminds me of what I like to call the "stacked up like LaGuardia" phenomenon. Air traffic controllers are responsible for ensuring the safe operation of flights while the planes are in the air, as well as during take off and landing. They know they have a limited number of gates with which to accommodate arriving and departing flights and a limited number of people who can arrive at these gates. But they also know that sometimes there are far more planes trying to land than there are available gates and personnel to handle them. Success of flight operations is highly dependent on maintaining the proper timing between arrivals and departures and ensuring that the airlines have all necessary personnel in the right place at the right time. Clearly, it is much easier for them to achieve their safety mission of ensuring safe flight operation if they only have one or two planes in the air. Yet, as we all painfully know, this isn't how the system works. Even the most carefully orchestrated schedules can go out the window when the number of planes increases and unexpected flights enter the picture. What typically results in this situation is a "stacking up" of planes waiting to land and trying to take off. At busy airports, like LaGuardia, this occurs all too frequently and causes delay and frustration for travelers and airport personnel alike.

I want the NRC to avoid a worst case scenario, like those faced by air traffic controllers, with incoming applications "stacked up like LaGuardia." The likelihood of this situation occurring increases dramatically as the number of possible applications for combined licenses, design certifications, and early site permits increases beyond planning assumptions. It is absolutely imperative that we know well in advance if an interested party intends to submit a license application. As with air traffic controllers, we know we have a limited number of resources to draw on to review these applications and must exercise impeccable timing and planning to ensure that we do not create unnecessary delay while moving forward with these actions.

I can also tell you that the risk of "stacking up" will increase dramatically if the industry uses the same application pattern it followed with license renewal. Initially, the industry as a whole was skeptical that the NRC could successfully review and approve an application for renewal of an operating license. Consequently, there were only one or two licensees bold enough to test the NRC's renewal process. After the agency demonstrated its review process was reasonably efficient and effective, however, we experienced a dramatic influx of renewal requests from a number of licensees. We were forced to do two things in order to manage the situation. First, while maintaining our safety focus, we took numerous steps to streamline our process and reduced our review time to 22 months from docketing to approval. Second, we mandated a policy of "first in-first out" for the handling of applications, while simultaneously limiting the number of license renewal applications we are working on in-house. I would fully expect that if faced with a similar situation in the future, the Commission would mandate the same policy for applications for combined licenses, design certifications, and early site permits. To return to my flight analogy, we only have so many "gates" and so many people who can manage these gates.

Of course, as a fee-based agency, any over-budgeting is passed on to our licensees. This is not an ideal situation. For example, when we plan for review of a design certification or for pre-application activities, we allocate personnel and funds sufficient to ensure a timely review. If these applications are subsequently delayed or withdrawn, as was the case with the ACR-700 and the PBMR, our staffing level remains the same and the cost for it is borne by all of our licensees. Not only is this wasteful of our human resources, but it is also unfair to our licensees. Clearly, the NRC is not the only party that could be dubbed "unpredictable" when it comes to the nuclear arena.

At the end of the day, it is in the best interest of all, if those who are intending to submit future applications to the NRC understand our budgetary and resource constraints and use that understanding to establish and follow realistic timetables. In my opinion, if it is reasonable for Congress and the industry to expect timeliness on the part of the NRC, it is also reasonable for the NRC to expect the same of vendors and licensees. Additionally, at a time when we are faced with a multiplicity of vendors competing for our review time, I believe the NRC must focus its efforts on designs that already have licensee interest rather than on designs that vendors wish to certify in hopes of leveraging reactor orders. We are well beyond the time when the agency can waste resources certifying designs that will never be ordered in the United States. We must focus on those designs that have a realistic possibility of being ordered and built.

That having been said, we need to do a better job communicating these expectations to our licensees, and particularly to the vendors. If we are alerted to an incoming application three to five years in advance, we can make the necessary adjustments to our budget proposal and staffing plans. If we have a letter of intent in hand, we are far more likely to receive the necessary funding increases from Congress. For us to meet our safety mission, ensuring the safe operation of both the currently operating fleet of reactors and those that may be built in the future, we need utilities and vendors to be candid with us about their realistic intentions.

The Pendulum Swings

Next, I would like to switch gears and discuss the potential impact of recent and future changes in utility management personnel. In the past few decades, we have observed a swinging of the pendulum a number of times between utility managers who come from a nuclear background and those who are more grounded in finance or law. As this pendulum appears to be on the move once again, I would like to share a few thoughts with you regarding this shift.

Prior to the accident at Three Mile Island in 1979, it was rare for a utility Chief Executive Officer to have risen from a nuclear operations background. More often than not, CEOs were of a more general technical background or had business or legal expertise. This generation of managers seemed to believe that nuclear plants were "just another way to boil water." The accident at TMI and the difficult period that followed had a significant impact on this mindset. In large part, it was TMI that caused the management pendulum to swing in the opposite direction. From this mishap emerged a new breed of utility leaders that were both "battle hardened" from dealing with the TMI action items and sensitive to the critical importance of nuclear safety.

One CEO in particular deserves an honorable mention for the key role he played during this time period. Chairman of the Board and CEO of Duke Power, Bill Lee, was a symbol of the reform the nuclear industry accomplished in the 1980s and 90s. Following the accident at TMI, Mr. Lee, in his role as Duke's president, led the creation of the Institute of Nuclear Power Operations, which strengthened and standardized the industry's nuclear safety and training programs. Later, as Duke's Chairman of the Board, he again took the lead for the industry, organizing the World Association of Nuclear Operators after the 1986 Chernobyl disaster in the Soviet Union. Both of these organizations did much to restore public confidence in nuclear power as a safe source of energy. This was by no means an easy task, and efforts to maintain public confidence continue even today.

Lee and his contemporaries learned the hard way what a delicate undertaking it can be to maintain a fair balance between ensuring financial profit of a nuclear operation while at the same time ensuring that its safety is preserved. These CEOs understood that a nuclear plant required special "care and feeding," and that operation of these plants could not be approached with the fossil plant mentality of "operate it until it breaks and then fix it."

They also struggled through the period of fear and suspicion that followed the TMI accident and saw firsthand how important candid communication with the public can be.

Today, we are in the midst of a transitional time when many of these leaders have left the industry, and it appears that the pendulum will once again swing back to an era where a large number of these senior managers may not hail from a nuclear background. A new generation of CEOs will be managing the future of the industry, and will be facing an entirely different backdrop as they enter their new positions. These men and women will run their organizations during a period of time when the nation's nuclear fleet is operating near peak performance. Reactors today are running at a much higher capacity than in the past, and with the notable exception of Davis-Besse, we have enjoyed a period of increasingly safe operations when compared to the 80s and 90s. This situation, although preferable from my perspective as a regulator, can be deceptive to the unwary.

It is essential that new industry leaders understand the inherent pitfalls of running a nuclear power plant. As I mentioned before, these plants must have safety infused into their systems and procedures, as well as into the minds of the employees that operate them. Our licensees must be proactive about discovering potential equipment problems early on to prevent equipment failures. Short term goals based on the bottom line cannot be allowed to overtake safety goals. As I have said on many occasions, good performers save money. I don't have to highlight how expensive poor performance has been for certain utilities.

These new utility leaders will also have to familiarize themselves with the way in which the NRC interacts with its licensees. I know full well that much of the industry believes that we are far too intrusive in our regulation of their nuclear facilities. Compared with other federal and state agencies, the NRC is probably a greater presence in the daily workings of our licensees than other regulators may be. But it is precisely this strong presence that enables us to fulfill our safety mission of ensuring protection of public health and safety.

Please do not get me wrong. I know that these executives with their extraordinary credentials can live up to my expectations. Indeed, one would expect me of all people to say that, given the fact that I came into my current position seven years ago as an attorney with a limited understanding of nuclear technologies. It would certainly be the "pot calling the kettle black" if I implied this would or could not be done. I know better than most the steep learning curve necessary to understand this technology, its promise, and its pitfalls. A strong commitment to learning about nuclear safety must certainly have been in the mind of Bill Lee when he helped establish INPO. Programs offered by INPO, like its "Reactor Technology Course for Utility Executives," offer an invaluable forum for learning and discussion.

In the end, this industry has had and will continue to have a number of key participants who do not come from a nuclear background. The clear need, however, is to ensure that this diversity of backgrounds enhances safe operation, rather than degrades it. Dedication to understanding nuclear safety and nuclear technology must be foremost in the mind of all future leaders in this arena. We cannot demand otherwise.

Conclusion

In summary, I can confidently say that the Commission is working hard to prepare for the future, whatever it may entail. We are planning to the best of our ability, but in my opinion, it will be the new generation of industry management that will help us avoid the "stacking up" phenomenon of which I spoke. I am positive that continued productive relationships between the NRC and industry leadership will help prevent us from becoming the "LaGuardia" we all hope the NRC will never be. Thank you very much.

August 3, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - COMJSM-05-0001 - STAFF FORMAL MEMBERSHIP IN COMMITTEES AND OUTSIDE GROUPS

The staff should develop a centralized list of all external organizations, both domestic and international, for which there is either a formal NRC representative or for which the NRC pays for travel for a staff member to attend the organization's meetings, functions, or sponsored meetings. This list should contain both the organization and the NRC staff member associated with that organization, and should be maintained current by staff, be readily accessible to senior management, and be reviewed as part of the annual budget process for appropriate accountability and decision making.

For the purpose of this list, the charter and NRC objectives for participation in committees and organizations requiring recurrent or large commitments should be clearly defined. The list should include standing committees for which we have a commitment to send representatives, such as the American Society of Mechanical Engineers code committees, the International Atomic Energy Agency, and the Nuclear Energy Agency standing committees. The list would not include committees or organizations contacted as part of public outreach activities for a specific rulemaking effort, industry groups which staff meets with to resolve technical issues at existing facilities, or a one time (or very infrequent) request to brief an organization on NRC efforts in some area (again as part of public outreach).

The EDO should review existing practices to ensure staff is provided with adequate and clear guidance when they are speaking on behalf of the Commission.

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

July 29, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-05-0052 - PROPOSED RULEMAKING FOR "RISK-INFORMED CHANGES TO LOSS-OF-COOLANT ACCIDENT TECHNICAL REQUIREMENTS"

The Commission has approved publication of the proposed rulemaking for risk-informed changes to loss-of-coolant accident technical requirements, subject to the comments noted below and the specific changes provided in the attachment.

(EDO) (SECY Suspense: 10/28/05)

General Comments

1. The requirements of the proposed 10 CFR 50.46a should be edited to remove the overly prescriptive regulatory treatment of beyond design basis LOCAs to be consistent with the low frequency of these events. (These changes, as well as those of other comments, are reflected in the attachment. The staff should make conforming changes, as needed, throughout the notice.)

2. The rule language should be simplified so that the change processes can be implemented in a straight-forward manner. The risk-informed change process in this rule should be based on the key principles of RG1.174. The NRC change processes in 10 CFR 50.59 and 50.90 are well understood and tested, and the proposed rule should rely on them as much as possible. For some changes, it may be difficult to distinguish between changes permitted under 50.46a and changes permitted under other sections. As a result, for licensees that use 50.46a, the integrated, risk-informed change process should be used for *all* changes made under 50.59 or 50.90. The proposed rule should be revised to address these points regarding the change process.

3. The Advisory Committee on Reactor Safeguards (ACRS) should review any final rule and Regulatory Guide proposal, following changes proposed as a result of the public comment period.

4. The staff should examine the other regulations and guidance to be sure there are no conflicts inadvertently introduced by the proposed rule, and if any are found should propose a resolution to the Commission.

5. The staff should update the Statements of Consideration to appropriately address the

issue of seismic loading of degraded piping and should solicit public comments on the subject. The staff should plan for a 90 day public comment period and make appropriate documents available to the public to inform the rulemaking effort.

6. The staff should include the following questions or comments in the *Federal Register* notice and specifically seek public comments on these issues. These items should be listed together.

A. The Commission instructed the staff not to make 50.46a available to future reactors. However, future light water reactors may benefit from 50.46a. As a result, comments should be solicited in the *Federal Register* regarding whether 50.46a should be made available to future light water reactors.

B. The proposed 50.46a includes an integrated, risk-informed change process to allow for changes to the facility following reanalysis of the beyond design basis LOCAs. However, the current regulations already have requirements addressing changes to the facility (10 CFR 50.59 and 50.90). It may be more efficient to include the integrated, risk-informed change requirements, for plants that use 50.46a, under our existing change processes. As a result, the staff should solicit comments on whether to revise 50.59 and 50.90 to accommodate changes enabled by 50.46a.

C. This rule will rely on risk information and the staff has included PRA requirements in the rule. However, there are other regulations that also rely on risk information (e.g. maintenance rule and alternative special treatment requirements). It may be more effective to describe the PRA requirements, consistent with the Commission policy on a phased approach to PRA quality, in one location in the regulations so that the PRA requirements are consistent among all regulations. As a result, the staff should solicit comments on the most effective way to include PRA requirements (e.g., contents, reporting, and changes) in the regulations.

D. The staff proposal includes specific "Operational Requirements" for operating configurations included in the analysis of beyond design basis LOCAs. Historically, operational restrictions have not been contained in 50.46 but were controlled through other requirements (e.g., technical specifications and maintenance requirements). It may be more practical to control equipment credited in the beyond design basis LOCA analysis in a more consistent manner with other operational restrictions. As a result, the staff should solicit comments on the most effective means and location for controlling appropriate operational restrictions for beyond design basis LOCAs.

E. The ACRS noted that "a better quantitative understanding of the possible benefits of a smaller break size is needed before finalizing the selection of the transition break size." The break size to be included in the final rule should be selected to maximize the potential safety improvements. The staff should specifically solicit comments on the relationship between the maximum design basis break size and potential safety improvements in the *Federal Register* notice.

F. Given the Commission's intent (ref: SRM for SECY-04-0037) that plant changes made possible by this rule should be constrained in areas where the current design requirements "contribute significantly to the 'built-in capability' of the plant to resist security threats," the Commission seeks examples on either side of this threshold (changes allowed vs. changes prohibited), and additionally any examples of changes that could enhance plant security and defense against radiological sabotage or attack. The Commission also solicits comments on whether the rule should explicitly include this requirement or otherwise rely on separate rulemaking being considered to more globally address this issue (e.g., changes to 50.59 and 50.90). Any examples that involve Safeguards Information should be marked and submitted using the appropriate procedures.

G. Given the potential impact to the licensee (i.e. the backfit rule not applicable) of the staff's periodic potential for re-evaluation of estimated LOCA frequencies, should the rule require licensees to maintain the capability to bring the plant into compliance, with an increased transition break size (TBS), within a reasonable period of time?

H. Is the rule sufficiently clear as to be "inspectable?" That is, does the rule language lend itself to timely and objective NRC conclusions regarding whether or not a licensee is in compliance with the rule, given all the facts? In particular, are the proposed requirements for PRA quality sufficient in this regard?

The following questions or comments are already included in the *Federal Register* but are listed or paraphrased here to ensure the list is complete and that it accurately captures the staff's intended solicitations.

I. The acceptability of combining 50.46a related and unrelated changes to meet 50.46a risk acceptance criteria (a.k.a. "bundling"). (I through M from pages 45-46 of FRN)

J. Whether 50.46a(f)(2)(iv) should allow unrelated changes to be bundled, or whether the rule should limit the consideration of risk impacts to only those changes related to the proposed rule.

K. Whether changes unrelated to 50.46a proposed by a licensee that meet the proposed high-level criteria for preventing creation of risk outliers should be included in determining the 50.46a change in risk estimate regardless of whether they are risk decreases or increases.

L. If bundling should be allowed, are the proposed high-level criteria for preventing creation of risk outliers adequate or should additional high-level criteria be imposed on what can and cannot be bundled, and if so, what specific high-level criteria should be utilized and incorporated into the final rule?

M. Whether there are circumstances that would favor bundling of changes that have already been implemented or the risk impacts of existing plant features when calculating the 50.46a change in risk estimates, in order to facilitate or enable safety improvements.

N. Whether there is an alternative to tracking the cumulative risk increases that is sufficient to provide reasonable assurance of protection to public health and safety and common defense and security. (pg 48 of FRN)

O. Whether the rule itself should include high-level criteria and requirements for the risk evaluation process and acceptance criteria described in Reg Guide 1.174, as currently proposed. (pg 51 of FRN)

P. Whether there are less burdensome, or more effective, ways of ensuring that the cumulative impact of an unbounded number of "minimal" changes remains inconsequential. (pg 71 of FRN)

Attachment: Changes to the *Federal Register* Notice in SECY-05-0052

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

Attachment

Changes to the Federal Register Notice in SECY-05-0052

1. On page 102, paragraph (a)(1), revise line 2 to read ' ... postulated design basis accident loss-of-coolant ....'

2. On page 103, paragraph (2), add the following at the end of the paragraph: "LOCAs involving breaks at or below the Transition Break Size (TBS) (see definition below) are considered design basis accidents. LOCAs involving breaks above the TBS are considered beyond design basis accidents."

3. On page 104, paragraph (c), revise line 4 to read ' ... analysis methods for LOCAs involving breaks at or below the TBS must meet ....' Revise line 6 to read ' ... for evaluation models ~~and analysis methods~~ for LOCAs involving breaks at or below the TBS. The analysis methods for LOCAs involving breaks above the TBS must be maintained, available for inspection, and include the analytical approaches, equations, approximations, and assumptions.

4. On pages 104-105, paragraph (2), revise line 1 to read ' ... *ECCS* analyses ~~*evaluation*~~ *for LOCAs* ....' Revise lines 3 and 4 to read ' ... satisfied. The ~~evaluation model or~~ analysis method ....' Revise lines 8 and 9 to read ' ... supporting justification, including the methodology used, must be available ~~provided~~ to show that ....' Delete the last sentence (When the calculated ... be exceeded.)

5. On pages 107-112, Paragraph (f) "Changes to the facility, technical specifications, and procedures," replace paragraphs (f)(1), (2), and (6) with the following:

(1) *Submission and approval process.* A licensee may request to make changes to its facility, technical specifications or procedures by submitting an application for a license amendment under 10 CFR 50.90. The application must contain the following information:
(i) The information required under 10 CFR 50.90 and;
(ii) A discussion of the method and a demonstration that the criteria in paragraph (c) and (f)(2) of this section have been met,

(2) *Risk-informed Integrated Safety Performance (RISP).* A licensee who wishes to make changes to its facility, technical specifications or procedures must perform a risk-informed integrated safety performance assessment which demonstrates that the following criteria associated with the change are met.
(i) For changes reviewed and approved by the NRC under 10 CFR 50.90, the total increases in core damage frequency and large early release frequency are small and the overall risk remains small. For changes that do not require prior NRC approval under 10 CFR 50.59, any increases in the estimated risk are minimal compared to the overall plant risk profile.
(ii) Defense-in-depth is maintained, in part by, assuring that:

reasonable balance is provided among prevention of core damage, prevention of containment failure (early or late), and consequence mitigation;

system redundancy, independence, and diversity are provided commensurate with the expected frequency of postulated accidents, the consequences of those accidents, and uncertainties; and

independence of barriers is not degraded.

(iii) Adequate safety margins are retained to account for uncertainties.

(iv) Adequate performance-measurement programs are implemented to ensure the RISP assessment reflects actual plant design and operation. These programs shall be designed to:

detect degradation of the system, structure or component before plant safety is compromised;

provide feedback of information and timely corrective actions;

monitor systems, structures or components at a level commensurate with their safety significance.

(6) *Facility and procedures changes not requiring NRC review and approval.* A licensee may make changes to its facility or procedures under § 50.59 without prior NRC review and approval and, provided the requirements below are met.

(i) *Submission and approval process.* A licensee who wishes to make changes to its facility or procedures without prior NRC review and approval must submit an application under § 50.90 to request NRC approval of a process for evaluating the acceptability of such changes. The application must contain the following information:

(A) A description of the licensee's PRA model and risk assessment methods for demonstrating compliance with paragraphs (f)(3) and (f)(4) of this section;

(B) A description of the methods and decisionmaking process for evaluating compliance with the risk criteria, defense-in-depth criteria, safety margin criteria and performance measurement criteria in paragraph (f)(2) of this section; and

(C) A description of the analysis to be performed for demonstrating compliance with paragraph (c) of this section.

(ii) *Acceptance criteria.* The NRC may approve a licensee's process for making changes to its facility and procedures without prior NRC review and approval, and a licensee may make such changes following such NRC approval if the process ensures that:

(A) The acceptance criteria in paragraphs (d) and (f)(2) of this section will be met; and

(B) The change is permitted under 10 CFR 50.59.

The Statements of Consideration should reflect the Commission's continuing support of the RG 1.174 guidelines as an acceptable approach for evaluating proposed changes. The Statements of Consideration should reflect consideration of other elements of defense-in-depth if and when they are relevant, as indicated by the words "in part by" in section (f)(2)(ii). The Statements of Consideration also should provide a discussion of what is meant by the "overall risk remains small."

6. On page 108, the requirements for maintaining containment integrity for realistically calculated pressures and temperatures for beyond design basis LOCAs for plants that adopt 10 CFR 50.46a should be moved from 50.46a(f)(2)(i)(B) and incorporated into GDC 50.

7. On page 110, paragraph (4), revise line 4 to read ' ... used produce ~~realistically conservative~~ realistic results.'

8. On page 111, paragraph (5), revise line 10 to read ' ... that ~~all changes accomplished under this section continue~~ facility design and operation continue to be consistent with the PRA assumptions used to meet ....'

9. On page 113, paragraph (h)(1), revise line 3 to read ' ... significant. For LOCAs involving pipe breaks at or below the TBS, f~~F~~or each change ....' Insert the following after the period in line 7: 'For LOCAs involving pipe breaks above the TBS, for each change to or

error discovered in an ECCS evaluation model or analysis method or in the application of such a model or method that affects the result, the licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in § 50.4.'

10. On page 114, revise paragraph (ii) to read: For LOCAs involving pipe breaks larger than the TBS, one which results in a significant reduction in the capability to meet the requirements of (d)(2) of this section ~~calculated peak fuel cladding temperature different by more than 300°F from the temperature calculated for the limiting transient using the last acceptable analysis method, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 300°F~~.

11. On page 114, paragraph (2), revise lines 1 through 9 to read ' ... licensee shall ~~compare the revised values of baseline CDF and LERF to those calculated under the last PRA model required by paragraph (f)(5) of this section; determine the cumulative changes in CDF and LERF for changes in the facility, technical specifications and procedures implemented under this section using the updated PRA model; and compare the revised values to the CDF and LERF values calculated under the previous PRA model required by paragraph (f)(5) of this section. If the baseline CDF or LERF increases by 20 percent or more, the cumulative change in CDF increases by 1x10⁻⁶ per year or more, or the cumulative change in LERF increases by 1x10⁻⁷ per year or more, the licensee shall~~ report the change to the NRC if the change results in a significant reduction in the capability to meet the requirements in (f)(2) of this section.

12. On page 120, delete the last sentence (For analysis methods ... be exceeded.)

July 28, 2005

The Honorable Tom Davis, Chairman
Committee on Government Reform
United States House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

On behalf of the U.S. Nuclear Regulatory Commission (NRC) and in accordance with 31 U.S.C. 720, I hereby submit our responses to the recommendations made by the U.S. Govemment Accountability Office (GAO) in its report entitled "Intemet Protocol Version 6: Federal Agencies Need to Plan for Transition and Manage Security Risks" (GAO-05-0471). Specific responses to the GAO recommendations are enclosed.

Sincerely,

/RA/

Nils J. Diaz

Enclosure:
NRC Responses to GAO Recommendations

cc: Representative Henry Waxman

Identical letter sent to:

The Honorable Tom Davis, Chairman
Committee on Government Reform
United States House of Representatives
Washington, D.C. 20515
cc: Representative Henry Waxman

The Honorable Susan Collins, Chairman
Committee on Homeland Security
and Governmental Affairs
United States Senate
Washington, D.C. 20510
cc: Senator Joseph I. Lieberman

The Honorable George V. Voinovich, Chairman
Subcommittee on Clean Air, Climate Change,
and Nuclear Safety
Committee on Environment and Public Works
United States Senate
Washington, D.C. 20510
cc: Senator Thomas Carper

The Honorable Ralph M. Hall, Chairman
Subcommittee on Energy and Air Quality
Committee on Energy and Commerce
United States House of Representatives
Washington, D.C. 20515
cc: Representative Rick Boucher

The Honorable Joe Barton, Chairman
Committee on Energy and Commerce
United States House of Representatives
Washington, D.C. 20515
cc: Representative John D. Dingell

The Honorable James M. Inhofe, Chairman
Committee on Environment and Public Works
United States Senate
Washington, D.C. 20510
cc: Senator James M. Jeffords

The Honorable David M. Walker
Comptroller General of the United States
Government Accountability Office
Washington, D.C. 20548

The Honorable Joshua B. Bolten, Director
Office of Management and Budget
Washington, D.C. 20503

NRC Responses to GAO Recommendations, GAO-05-0471,
Internet Protocol Version 6: Federal Agencies Need to Plan
for Transition and Manage Security Risks

Background

On May 24, 2005, the Government Accountability Office (GAO) issued a report on Internet Protocol Version 6: Federal Agencies Need to Plan for Transition and Manage Security Risks (GAO-05-0471). This report discusses the key issues surrounding Internet Protocol Version 6 (IPv6) and the security considerations for the transition to IPv6 for federal agencies. GAO expressed concern about poorly configured and unmanaged IPv6 capabilities within federal agencies. GAO recommends in the report that agency heads take action to address near term security risks, including determining what IPv6 capabilities they may have and initiate steps to ensure they can control and monitor IPv6 traffic.

Recommendations and Responses

Nuclear Regulatory Commission's (NRC's) responses to GAO's recommendations to Office of Management and Budget (OMB) appear below.

Recommendation 1 to OMB

Instruct Federal agencies to begin addressing key IPv6 planning considerations, including:

- developing inventories and assessing risks,
- creating business cases for the IPv6 transition,
- establishing policies and enforcement mechanisms,
- determining costs, and
- identifying time lines and methods for transition, as appropriate.

Response

NRC has a three phase approach to planning for and implementing IPv6. Phase one, which has already begun, emphasizes the agency's business drivers and goals for the IPv6 transition. This phase will identify the risks and benefits as a result of the transition to IPv6. From this development process, the agency's strategy will be refined to determine the alignment of the technology with the NRC business goals. The expected completion of phase one is September 2005.

Phase two of the agency's strategy will include a readiness assessment to identify the existing NRC technology baseline for IPv6. This baseline assessment will include a careful review of all planned and scheduled information technology (IT) acquisitions. The baseline will also be used to determine the IPv6 transition prioritization criteria for existing investments at various stages of their life cycle. Phase two of NRC's strategy is expected to start in September 2005 and be completed in November 2005.

Phase three is the implementation phase of the IPv6 strategy. It will provide a transition plan that will take into account specific agency enterprise-wide and individual IT investments. This phase is expected to start in October 2007 and be completed in September 2009.

NRC's three-phased strategy for the implementation, management, and investment in IPv6 technologies will address the development of inventories for current IT investments and planned acquisitions as part of the second phase. Based upon the inventory baseline, the risks associated with the implementation of the technology will be assessed.

With respect to the creation of business cases for the IPv6 transition, the agency's strategy is to identify the business drivers for migration from the current Internet Protocol Version 4 (IPv4) to IPv6, to ensure the transition provides value in achieving the agency's business goals, and to ensure that the technology will integrate with the agency's EA.

In phase three of the agency's strategy, the NRC will identify any specific IT, procurement, or other policy documents requiring modification to provide the necessary IPv6 transition support to effectively promulgate the OMB IPv6 guidance within the agency. The agency will include an appropriate clause enforcing IPv6 compatibility in agency IT acquisitions and has factored into agency resource planning the need to transition to IPv6.

IPv6 Framework Phase	Estimated Start Date	Estimated Completion Date
I - Strategic Planning	1/05	9/05
II - IPv6 Readiness Assessment	9/05	11/05
III - Implementation	10/07	9/09

Recommendation 2 to OMB

Agency heads take immediate actions to address the near-term security risks, including determining what IPv6 capabilities they may have, and initiate steps to ensure that they can control and monitor IPv6 traffic.

Response

The agency's current IT infrastructure is only configured to support IPv4-formatted traffic. Consequently, there are no current IPv6 capabilities and no near term security risks. The agency will address the management and monitoring of IPv6 traffic in the development of agency policy and governance as IPv6 is incorporated into NRC's infrastructure.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED 08/04/05

SERVED 08/04/05

COMMISSIONERS

Nils J. Diaz, Chairman
Jeffrey S. Merrifield
Gregory B. Jaczko
Peter B. Lyons

In the Matter of DOMINION NUCLEAR CONNECTICUT, INC. (Millstone Nuclear Power Station, Units 2 and 3)	Docket Nos. 50-336-LR & 50-423-LR

CLI-05-18

MEMORANDUM AND ORDER

On July 20, 2005, the Licensing Board in this case issued a Memorandum and Order. LBP-05-16, 62 NRC ___. The Board order concluded that Suffolk County's tardiness in submitting its petition to intervene was excusable under the late-filing standards of 10 C.F.R. § 2.309(c). Additionally, the Board found no basis to exclude the County from participation under the contention requirements of 10 C.F.R. § 2.309(f). The Board also certified to the Commission the question whether to grant Suffolk County's request for an exemption from (or waiver of) 10 C.F.R. § 50.47(a)(1) (which provides that emergency planning issues are not germane to license renewal determinations). Today we grant review of that certified question. In doing so, we follow our "customary practice" of accepting Board-certified questions.[1]

[1] *See, e.g., Exelon Generation Co.* (Early Site Permit for Clinton ESP Site), CLI-05-9, 61 NRC 235, 236 (2005); *Duke Energy Corp.* (Catawba Nuclear Station, Units 1 and 2), CLI-04-11, 59 NRC 203, 209 (2004); *Private Fuel Storage, L.L.C.* (ISFSI), CLI-01-12, 53 NRC 459, 461 (2001).

We also intend to consider, *sua sponte*, three other questions -- (1) whether Suffolk County's late-filed contention was admissible under the criteria for considering late-filed pleadings and contentions set out in 10 C.F.R. § 2.309(c); (2) whether Suffolk County's contention regarding "emergency planning" satisfied the contention requirements in 10 C.F.R. § 2.309(f); and (3) whether, under the circumstances of this case, the Board properly postponed its contention-admissibility decision pending settlement talks.

We solicit the views of the adjudication's participants on these three questions, plus the certified question. To this end, we establish the following filing schedule. No later than August 18, 2005, the Staff, licensee and petitioner may file initial briefs, each of which may not exceed 25 pages, exclusive of the tables of contents and authorities (both of which we require). No later than August 25, 2005, the Staff, licensee and petitioner may file response briefs, each of which may not exceed 10 pages and need not include tables of contents and authorities. Each participant should ensure that we *receive* each of its briefs no later than 4:15 p.m. on the due date.

IT IS SO ORDERED.

For the Commission[2]

/RA/

Annette L. Vietti-Cook
Secretary of the Commission

Dated at Rockville, Maryland,
this 4th day of August, 2005.

[2] Chairman Diaz was not present when this item was affirmed. Accordingly the formal vote of the Commission was 3-0 in favor of the decision. Chairman Diaz, however, had previously voted to approve this Memorandum and Order and had he been present he would have affirmed his prior vote.

EA-05-102 - Indian Point 2 (Entergy Nuclear Operations, Inc.)

EA-05-102

August 1, 2005

Mr. Fred Dacimo
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Station
295 Broadway, Suite 1
Post Office Box 249
Buchanan, NY 10511-0249

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING (NRC Engineering Team Inspection Report 05000247/2005006) Indian Point Nuclear Generating Unit 2

Dear Mr. Dacimo:

This letter provides the final results of our significance determination for the preliminary White finding identified at Indian Point 2 during an engineering team inspection completed on April 27, 2005. The results of the inspection were discussed with Mr. C. Schwarz and other members of your staff via telephone during an exit meeting on May 18, 2005. The inspection finding was assessed using the significance determination process and was preliminarily characterized as White, a finding with low to moderate importance to safety that may require additional NRC inspections. The basis for this preliminary White finding was explained in our letter dated June 17, 2005, which transmitted the subject inspection report.

This preliminary White finding involved inadequate evaluation and corrective actions for a degraded condition that existed in the safety injection system. The degraded condition involved water from a safety injection accumulator leaking past several closed valves, allowing water containing absorbed nitrogen to reach other portions of the emergency core cooling system, including the common suction supply piping for the safety injection pumps and the #23 safety injection pump casing. As the water moved from a higher to lower system pressure, the nitrogen gas was released from the water, thereby challenging the performance of the safety injection pumps. As a result, the NRC concluded that the #23 safety injection pump was not functional because the pump casing was filled with gas, and the capability of the #21 and #22 safety injection pumps was challenged because the common suction header contained a significant accumulation of gas.

Although Entergy entered the degraded accumulator condition issue into its corrective action system on November 21, 2004, prioritization, evaluation and correction of this condition adverse to quality was inadequate. In particular, Entergy did not recognize the potential for gas intrusion into the safety injection system and the resultant potential challenge to safety injection pump operation. In addition, Entergy did not adequately assess industry Operating Experience in accordance with its procedures in that such operating experience (e.g. - NRC Information Notice 97-40) related to safety injection accumulator backleakage, was not assimilated and acted upon in a timely manner.

In our letter dated June 17, 2005, the NRC provided you an opportunity to either request a Regulatory Conference to discuss this finding, or to explain your position in a written response. On June 23, 2005, Mr. J. Comiotes of your staff informed Mr. B. McDermott of my staff, that Entergy declined the offer to have a Regulatory Conference. However, you did provide a written response dated July 12, 2005, in which you stated Entergy's belief that the #21 and #22 safety injection pumps remained operable. However, you noted that there are uncertainties involved with predicting the performance of the safety injection pumps with nitrogen gas in the system, and did not want to expend resources to prove past pump op[illegible]ity. Therefore, after noting that this issue did not represent a significant increase in risk, you stated that Entergy ac[illegible] the results of the inspection finding as presently characterized.

After considering the information developed during the inspection, and the information presented in your response, the NRC has concluded that the inspection finding is appropriately characterized as White, an issue with low to moderate increased importance to safety that may require additional inspections. You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC has also determined that this finding is a violation of 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Action). The circumstances surrounding the violation are described in detail in the enclosed Notice of Violation as well as the subject inspection report. In accordance with the NRC Enforcement Policy, this Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket as summarized in the NRC Engineering Team inspection report dated June 17, 2005. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for p[illegible] inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's A[illegible]wide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Samuel J. Collins
Regional Administrator

Docket No. 50-247
License No. DPR-26

Enclosure: Notice of Violation

cc w/encl:
G. J. Taylor, Chief Executive Officer, Entergy Operations, Inc.
M. R. Kansler, President - Entergy Nuclear Operations, Inc.
J. T. Herron, Senior Vice President and Chief Operating Officer
P. Rubin, General Manager - Plant Operations
O. Limpias, Vice President, Engineering
C. Schwarz, Vice President, Operations Support
J. McCann, Director, Licensing
C. D. Faison, Manager, Licensing, Entergy Nuclear Operations, Inc.
P[illegible]oy, Manager, Licensing, Entergy Nuclear Operations, Inc.
M[illegible]mb, Director of Oversight, Entergy Nuclear Operations, Inc.
J. Comiotes, Director, Nuclear Safety Assurance
J. M. Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc.
P. R. Smith, President, New York State Energy, Research and Development Authority

Spath, Program Director, New York State Energy Research and Development Authority
. Eddy, Electric Division, New York State Department of Public Service
. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
. O'Neill, Mayor, Village of Buchanan
G. Testa, Mayor, City of Peekskill
. Albanese, Executive Chair, Four County Nuclear Safety Committee
. Lousteau, Treasury Department, Entergy Services, Inc.
hairman, Standing Committee on Energy, NYS Assembly
hairman, Standing Committee on Environmental Conservation, NYS Assembly
hairman, Committee on Corporations, Authorities, and Commissions
1. Slobodien, Director, Emergency Planning
. Brandenburg, Assistant General Counsel
. Rubin, Manager of Planning, Scheduling & Outage Services
ssemblywoman Sandra Galef, NYS Assembly
ounty Clerk, Westchester County Legislature
. Spano, Westchester County Executive
. Bondi, Putnam County Executive
. Vanderhoef, Rockland County Executive
. A. Diana, Orange County Executive
. Judson, Central NY Citizens Awareness Network
. Elie, Citizens Awareness Network
. Lochbaum, Nuclear Safety Engineer, Union of Concerned Scientists
ublic Citizen's Critical Mass Energy Project
. Mariotte, Nuclear Information & Resources Service
. Zalcman, Pace Law School, Energy Project
L. Puglisi, Supervisor, Town of Cortlandt
Congresswoman Sue W. Kelly
Congresswoman Nita Lowey
Senator Hillary Rodham Clinton
Senator Charles Schumer
J. Riccio, Greenpeace
A. Matthiessen, Executive Director, Riverkeeper, Inc.
M. Kapolwitz, Chairman of County Environment & Health Committee
A. Reynolds, Environmental Advocates
M. Jacobs, Director, Longview School
D. Katz, Executive Director, Citizens Awareness Network
P. Gunter, Nuclear Information & Resource Service
P. Leventhal, The Nuclear Control Institute
K. Coplan, Pace Environmental Litigation Clinic
W. DiProfio, PWR SRC Consultant
D. C. Poole, PWR SRC Consultant
W. Russell, PWR SRC Consultant
W. Little, Associate Attorney, NYSDEC
R. Christman, Supt. Operations Training
L. Cortopassi, Manager Training and Development
S. Glenn, INPO

NOTICE OF VIOLATION

Entergy Nuclear Operations, Inc.
Indian Point Unit 2

Docket No.: 50-247
License No.: DPR-26
EA-05-102

During an NRC inspection completed April 27, 2005, for which an exit meeting was held on May 18, 2005, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the viola is listed below:

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that conditions adverse to quality

be promptly identified and corrected.

Entergy's corrective action system procedure ENN-LI-102, "Corrective Action Process," requires, in part, the identification, evaluation and correction of a broad range of problems, and documentation of both previous site (in-house) and industry Operating Experience Reviews, when appropriate (commensurate with issue significance).

Contrary to the above, between November 21, 2004 and January 27, 2005, Entergy did not promptly identify and correct a condition adverse to quality regarding the potential for gas intrusion into the safety injection system discharge and suction piping from known leakage from the #24 safety injection accumulator. Although Entergy entered the degraded accumulator condition issue into its corrective action system on November 21, 2004, prioritization, evaluation and correction of this condition adverse to quality was inadequate since Entergy did not:

1. recognize the potential for gas intrusion into the safety injection system and the resultant potential challenge to safety injection pump operation; and

2. adequately assess industry Operating Experience in accordance with ENN-LI-102 in that such operating experience (e.g. - NRC Information Notice 97-40) related to safety injection accumulator backleakage, was not assimilated and acted upon in a timely manner.

As a result, a significant amount of gas accumulated within the safety injection discharge piping, the #23 safety injection pump casing, and the common safety injection pump suction header. This resulted in the #23 safety injection pump not being functional, and the capability of the #21 and #22 safety injection pumps being challenged.

This violation is associated with a WHITE significance determination process finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket as summarized in NRC Engineering Team Inspection Report 05000247/2005006 dated June 17, [illegible] However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description t[illegible]n does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation; EA-05-102," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

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

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

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

[illegible] this 1st day of August 2005

EA-05-100 - Three Mile Island 1 (AmerGen Energy Co., LLC)

July 29, 2005

EA-05-100

Mr. Christopher M. Crane
President and CEO
AmerGen Energy Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING
(NRC Emergency Preparedness Program Inspection Report 05000289/2005006)
Three Mile Island Station, Unit 1

Dear Mr. Crane:

This letter provides the final results of our significance determination for the preliminary White finding identified at Three Mile Island, Unit 1, during an emergency preparedness program inspection completed on May 19, 2005. The results of the inspection were discussed with Mr. G. Chick, Plant Manager, and other members of your staff during an exit meeting on May 19, 2005. The inspection finding was assessed using the significance determination process and was preliminarily characterized as White, a finding with low to moderate importance to safety that may require additional NRC inspections. The basis for this preliminary White finding was explained in our letter dated June 30, 2005, which transmitted the subject inspection report.

This preliminary White finding involved approximately 50 percent of the Emergency Response Organization (ERO), including key responders, not receiving the required annual radiological response classroom retraining necessary to maintain familiarity with their specific emergency response duties. Although these individuals had received such training in March 2003, they were not provided the retraining until November 2004. The TMI Annex Emergency Plan defined annual as every 12 months ± 3 months, and therefore, the annual retraining of these individuals should have been completed by June 2004. As a consequence of these individuals not completing the annual retraining until November 2004, the individuals would not have been considered qualified to respond to a radiological emergency for an approximate five-month period (June - November 2004). This resulted in some key ERO positions not being filled by qualified ERO members in accordance with AmerGen's Three Mile Island Emergency Plan requirements.

In our letter dated June 30, 2005, the NRC provided you an opportunity to either request a Regulatory Conference to discuss this finding, or to explain your position in a written response. In a telephone call between Mr. C. Smith and Mr. R. Lorson of my staff on July 8, 2005, as well as in a letter on that date from Mr. R. G. West, Vice President, Three Mile Island Unit 1, the NRC was informed that AmerGen did not contest the preliminary White finding, declined a Regulatory Conference, and would not be providing a written response.

After considering the information developed during the inspection, the NRC has concluded that the inspection finding is appropriately characterized as White, an issue with low to moderate increased importance to safety that may require additional inspections.

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual

Chapter 0609, Attachment 2.

The NRC has also determined that this finding is a violation of 10 CFR 50.47(b)(15). The circumstances surrounding the violation are described in detail in the enclosed Notice of Violation (Notice) as well as the subject inspection report. In accordance with the NRC Enforcement Policy, this Notice is considered escalated enforcement action because it is associated with a White finding. You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response.

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Samuel J. Collins
Regional Administrator

Docket No. 50-289
License No. DPR-50

Enclosure: Notice of Violation

cc w/Attachment: **(VIA E-MAIL)**
Chief Operating Officer, AmerGen
Site Vice President - TMI Unit 1, AmerGen
Plant Manager - TMI, Unit 1, AmerGen
Regulatory Assurance Manager - TMI, Unit 1, AmerGen
Senior Vice President - Nuclear Services, AmerGen
Vice President - Mid-Atlantic Operations, AmerGen
Vice President - Operations Support, AmerGen
Vice President - Licensing and Regulatory Affairs, AmerGen
Director Licensing - AmerGen
Manager Licensing - TMI, AmerGen
Vice President - General Counsel and Secretary, AmerGen
T. O'Neill, Associate General Counsel, Exelon Generation Company
J. Fewell, Esq., Assistant General Counsel, Exelon Nuclear
Correspondence Control Desk - AmerGen
Chairman, Board of County Commissioners of Dauphin County
Chairman, Board of Supervisors of Londonderry Township
R. Janati, Chief, Division of Nuclear Safety, State of PA
J. Johnsrud, National Energy Committee
E. Epstein, TMI-Alert (TMIA)
D. Allard, PADER

NOTICE OF VIOLATION

AmerGen Energy Company
Three Mile Island Unit 1

Docket No. 50-289
License No. DPR-50
EA-05-100

During an NRC inspection completed on May 19, 2005, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR 50.54(q) specifies that a licensee authorized to possess and operate a nuclear power reactor shall follow and maintain in effect emergency plans which meet the standards in 10 CFR 50.47(b).
>
> 10 CFR 50.47(b)(15), states that radiological emergency response training is provided to those who may be called on to assist in an emergency.
>
> 10 CFR 50, Appendix E, Section F.a.1, states, in part, that the licensee will provide training of employees, as well as exercising by periodic drills of radiation emergency plans, to ensure that employees of the licensee are familiar with the specific emergency response duties.
>
> TMI Annex Emergency Plan, Section 2.3, states in part, that retraining is performed on an annual basis which is defined as every 12 months ± 3 months.
>
> Contrary to the above, as of November 2004, approximately 50% of the Emergency Response Organization had not received the annual required radiological response classroom retraining since March 2003, a period of approximately 20 months which was in excess of the required 12 months ± 3 months. As a result, from June 2004 until November 2004, these individuals would not have been considered qualified to respond to a radiological emergency and fill their assigned Emergency Response Organization positions.
>
> This violation is associated with a WHITE significance determination process finding.

Pursuant to the provisions of 10 CFR 2.201, Amergen Energy Company, LLC, is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-05-100" and should include for the violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

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

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

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

Dated this 29th day of July 2005

EA-05-039 - Pilgrim (Entergy Nuclear Operations, Inc)

July 14, 2005

EA 05-039

Mr. Michael Balduzzi
Site Vice President
Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, Massachusetts 02360

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY - $60,000 (NRC Office of Investigations Report No. 1-2004-040) Pilgrim Nuclear Power Station

Dear Mr. Balduzzi:

This letter refers to an investigation initiated by the NRC Office of Investigations (OI), Region I, on August 27, 2004, at Entergy Nuclear Operation's Pilgrim Nuclear Power Station (Pilgrim). This investigation was initiated to determine if a Control Room Supervisor (CRS) at Pilgrim was sleeping/inattentive to duty on June 29, 2004, and whether others were aware that the CRS was inattentive and failed to correct the condition. As described in our letter to you dated March 23, 2005, the NRC concluded, based on the results of the OI investigation, that (1) the CRS was sleeping and inattentive to duty; (2) a Reactor Operator (RO) observed the CRS sleeping but deliberately failed to take immediate action to awaken the CRS, notify the Shift Manager (SM), and/or write a condition report (CR); and (3) a SM subsequently observed that the CRS was inattentive and failed to fully follow procedures in careless disregard of requirements.

In our March 23, 2005, letter, we also informed you that the failure of the CRS, RO and SM to follow procedures caused you to violate Technical Specification 5.4.1 and 10 CFR 26.20. Specifically, (1) the CRS was asleep for approximately four minutes on June 29, 2004, and was, therefore, neither alert nor attentive to his duties in violation of the "Conduct of Operations" procedure, and the CRS did not ask the SM for relief before or after he was awakened by the SM; (2) the RO observed the CRS asleep, but deliberately failed to take immediate actions to awaken the CRS, failed to inform appropriate site personnel, and failed to initiate a CR in violation of the "Corrective Action Process" procedure; and (3) after observing the inattentive CRS and taking some immediate actions to correct the situation, the SM, in careless disregard of requirements, failed to immediately relieve the CRS of his duties, did not have the CRS for-cause fitness for duty (FFD) tested, did not inform appropriate site personnel, and did not initiate a CR in violation of the "Fitness For Duty Program" and the "Corrective Action Process" procedures.

The NRC considered the RO's actions to be deliberate because he knew that procedures required him to take action, yet he did not awaken the CRS, or notify management, or write a CR when he observed that the CRS was sleeping. In addition, the NRC considered the SM's actions to be in careless disregard of requirements because, even though he took immediate actions to end the CRS's inattentive behavior by slamming a desk, the SM should have known of the requirement to relieve the CRS of duty and have him for-cause FFD tested. He also should have been aware of the requirement to inform appropriate site personnel and write a CR, given his position as the senior person on duty at the time, as well as his prior knowledge of a previous similar event. It is important to note that the SM was aware of a sleeping incident in a break room two months earlier that led to all SMs being required to discuss the event with their crews to emphasize peer checking, alertness, attentiveness, and FFD-related matters. In addition, the SM had an opportunity to inform appropriate site personnel of his observation because he met with operations management approximately two hours after he had observed the inattentive CRS, yet he did not inform them of his observation. These conclusions were noted in a factual summary of the OI report that was sent to you in our March 23, 2005, letter.

Our March 23, 2005, letter scheduled a predecisional enforcement conference and also offered you the opportunity to request alternative dispute resolution (ADR) instead. At your request, a predecisional enforcement conference was conducted on April 8, 2005, with members of your staff in the Region I Office in King of Prussia, PA. During the conference, you discussed the apparent violations, their significance, their root causes, and your corrective actions. You also disagreed with some of the NRC conclusions discussed in the March 23, 2005, letter.

With respect to the CRS, although you agreed that he was inattentive to duty, you stated that the CRS had not felt unfit before or after the event, and therefore, would not have been expected to ask for relief. This was consistent with the testimony provided by the CRS. After further consideration of this matter, the NRC has concluded that you were not in violation regarding this aspect of the event.

With respect to the SM, you agreed that he should have informed management of his observation that the CRS was inattentive to duty. You also agreed that the SM's failure to notify management was in violation of your procedures. However, you maintained that the SM's failure to inform management was a mistake in judgement rather than careless disregard of requirements. In addition, you disagreed that the SM needed to immediately relieve the CRS from duty, have him for-cause FFD tested and write a CR. You noted that since the SM observed the CRS "head-bobbing" rather than sleeping as the RO had observed, and since the SM did not believe (based on a counseling session with the CRS shortly after the incident) that drugs or alcohol were involved, immediate relief and for-cause FFD testing were not required. You also noted that since the SM had observed the CRS "head-bobbing" rather than sleeping, and took prompt action to correct the inattentive condition by counseling the CRS, preparation of a CR was not required.

After further consideration of this matter, the NRC has concluded that the SM observed the CRS (who was responsible for supervising the manipulation of the controls of a nuclear facility) to be inattentive to his duties, and that this observation constituted an adverse condition that should have been communicated to management and warranted initiation of a CR per your procedure. Furthermore, the NRC maintains that the SM, by not informing management of his observation and writing a CR, acted in careless disregard of requirements, after considering (1) his position as the senior member of the operating crew; (2) his prior discussions with the crew just two months earlier regarding the importance of being alert and attentive and (3) his failure to inform management despite being admittedly very upset with what he had observed and despite meeting with operations management approximately two hours later. Although the SM may not have observed the CRS sleeping, credible information was available to substantiate that the CRS was asleep; therefore, in accordance with your procedures, the NRC maintains that the CRS should have been relieved of duty and for-cause FFD tested.

Accordingly, after considering the information developed during the OI investigation and the information that you provided during the enforcement conference, the NRC has determined that four violations of NRC requirements occurred. The violations, which are cited in the enclosed Notice of Violation (Notice), involve: (1) the CRS being asleep, and therefore, neither alert nor attentive to his duties (Violation A); (2) the RO observing the CRS asleep, but deliberately failing to take immediate actions to awaken the CRS, inform appropriate site personnel, and initiate a CR (Violation B.1); (3) the SM, in careless disregard of requirements, failing to inform appropriate site personnel and initiate a CR (Violation B.2); and (4) the CRS not being relieved of duty and for-cause FFD tested (Violation C).

Although there was no actual safety consequence resulting from this event because there were no plant conditions that warranted immediate action, it is important for licensed operators to be alert and attentive to their control room duties at all times so that they can adequately monitor the reactor, manipulate reactor controls, and react to any plant transients. It is also important that when licensed operators are not alert or attentive to their duties, appropriate action must be taken to immediately correct the situation and inform management. After considering these facts, as well as the willful acts of the RO and SM, the NRC has concluded that these four violations should be categorized as a Severity Level III problem in accordance with the NRC Enforcement Policy.

In accordance with the Enforcement Policy that was in effect in June 2004, a base civil penalty in the amount of $60,000 is considered for a Severity Level III problem. Because the Severity Level III problem included violations that were willful, the NRC considered whether credit was warranted for *Identification and Corrective Action* in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. Because you did not identify the violations, credit is not warranted for the *Identification* factor. Because your corrective actions taken or planned were considered comprehensive credit for corrective action is warranted. Your actions included: (1) initial actions by the SM to assure the CRS became attentive; (2) senior managers meeting with all operations personnel, security personnel and other plant personnel to discuss the event, FFD obligations, and safety conscious work environment requirements; (3) enhancement of back-shift monitoring; (4) enhancement of FFD, Continuing Behavior Observation Program and safety conscious work environment training; and (5) initiation of plans to conduct lifestyle training and operating crew teamwork training.

After considering the available information, I have been authorized, after consultation with the Director, Office of Enforcement, to issue the enclosed Notice of Violation and Proposed Imposition of Civil Penalty (Notice) in the base amount of $[illegible],000 for the Severity Level III problem. This action is being taken to emphasize that all licensed operators must remain alert and attentive to duty at all times, and must act appropriately when conditions adverse to quality are identified. You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

As an option, you may request alternative dispute resolution (ADR) with the NRC in an attempt to resolve this issue. ADR is a general term encompassing various techniques for resolving conflict outside of court using a neutral third party. The technique that the NRC has decided to employ during a pilot program which is now in effect is mediation. Additional information concerning the NRC's pilot program is described in the enclosed brochure (NUREG/BR-0317) and can be obtained at http://www.nrc.gov/what-we-do/regulatory/enforcement/adr.html. The Institute on Conflict Resolution (ICR) at Cornell University has agreed to facilitate the NRC's program as an intake neutral. Please contact ICR at 877-733-9415 within 10 days of the date of this letter if you are interested in pursing resolution of this issue through ADR.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select **What We Do, Enforcement,** then **Significant Enforcement Actions.**

Should you have any questions regarding this letter, please feel free to contact Mr. Clifford Anderson at 610-337-5227.

Sincerely,

/RA/

Samuel J. Collins
Regional Administrator

Docket No. 50-293
License No. DPR-35

Enclosures: 1. Notice of Violation and Proposed Imposition of Civil Penalty
2. Brochure NUREG/BR-0317

cc w encl:
G. J. Taylor, Chief Executive Officer, Entergy Operations
J. T. Herron, Senior Vice President and Chief Operating Officer
M. Kansler, President, Entergy Nuclear Operations, Inc.
S. J. Bethay, Director, Nuclear Assessment
D. L. Pace, Vice President, Engineering
C. Schwarz, Vice President, Operations Support
J. F. McCann, Director, Nuclear Safety Assurance
C. D. Faison, Manager, Licensing
M. J. Colomb, Director of Oversight, Entergy Nuclear Operations, Inc.
D. Tarantino, Nuclear Information Manager
B. S. Ford, Manager, Licensing, Entergy Nuclear Operations, Inc.
J. M. Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc.
S. Lousteau, Treasury Department, Entergy Services, Inc.
R. Walker, Department of Public Health, Commonwealth of Massachusetts
The Honorable Therese Murray
The Honorable Vincent deMacedo
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
Chairman, Nuclear Matters Committee

Plymouth Civil Defense Director
D. O'Connor, Massachusetts Secretary of Energy Resources
J. Miller, Senior Issues Manager
Office of the Commissioner, Massachusetts Department of Environmental Protection
Office of the Attorney General, Commonwealth of Massachusetts
Electric Power Division, Commonwealth of Massachusetts
R. Shadis, New England Coalition Staff
D. Katz, Citizens Awareness Network
Chairman, Citizens Urging Responsible Energy
J. Sniezek, PWR SRC Consultant
R. Toole, PWR SRC Consultant
C. McCombs, Acting Director, MEMA and Commonwealth of Massachusettts, SLO Designee
Commonwealth of Massachusetts, Secretary of Public Safety

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTY

Entergy Nuclear Operations, Inc
Pilgrim Nuclear Power Station

Docket No. 50-293
License No. DPR-35
EA-05-039

Based on an NRC investigation conducted by the Office of Investigations, Region I Field Office, the report of which was issued on February 4, 2005, four violations of NRC requirements were identified. In accordance with the NRC Enforcement Policy, the NRC proposes to impose a civil penalty pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The particular violations and associated civil penalty are set forth below:

Technical Specification 5.4.1 of Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station (PNPS) requires the establishment and implementation of procedures covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, February 1978.

Regulatory Guide 1.33 recommends, in part, administrative procedures covering authorities and responsibilities for safe operation and shutdown, and shift and relief turnover.

10 CFR Part 50, Appendix B, requires a quality assurance program for nuclear power plants to assure that conditions adverse to quality are promptly identified and corrected, and includes, in part, written policies, procedures or instructions.

10 CFR Part 26, "Fitness For Duty Programs," prescribes requirements and standards for the establishment and maintenance of certain aspects of fitness for duty programs and procedures by licensees, and each licensee subject to this part shall establish and implement written policies and procedures to meet these objectives.

A. PNPS Procedure Number 1.3.34, "Conduct of Operations", Section 5.15, requires, in part, that Operations personnel on duty will remain alert and awake so that they may respond to plant conditions or emergencies. PNPS Procedure Number 1.3.34 is required by Technical Specification 5.4.1.

Contrary to the above, for approximately four minutes on June 29, 2004, the on duty Control Room Supervisor (CRS) was not alert to his duties in the control room in that he was asleep in a chair, and therefore, not in a condition to respond to plant conditions or emergencies.

B. PNPS Procedure ENN-LI-102, "Corrective Action Process", Section 4.1, requires, in part, that all personnel working at Entergy Nuclear Northeast (ENN) facilities are responsible for identifying and reporting problems. Section 5.1.1.4, requires, in part, that any individual who discovers an adverse condition is expected to ensure that: immediate actions are taken as necessary to minimize the consequence of the condition; appropriate site personnel are notified of the identified condition; and the condition is promptly documented in a Condition Report (CR). ENN-LI-102 is required by 10 CFR Part 50, Appendix B.

Contrary to the above, on June 29, 2004:

1. a Reactor Operator (RO), at approximately 4:40 a.m., became aware of an adverse condition (the RO observed the CRS to be asleep in a chair), and the RO did not take immediate actions to awaken the CRS and minimize the consequence of the condition, did not inform appropriate site personnel that he had observed the CRS to be asleep, and did not document the condition in a CR.

2. a Shift Manager (SM), at approximately 4:45 a.m., became aware of an adverse condition (the SM observed the CRS "head-bobbing" in a chair, and was therefore, inattentive to his duties and not fully alert), and the SM did not inform appropriate site personnel of the condition and did not document the condition in a CR.

C. PNPS Procedure ENN-NS-102, "Fitness For Duty Program," Section 3.0, defines, in part, for-cause testing as testing that is conducted as soon as possible following an observed behavior that indicates questionable fitness for duty. Section 5.3 states, in part, that factors such as fatigue, mental stress and illness may affect an individual's fitness for duty. Section 5.7 further states, in part, that testing for-cause shall be based on observation or information received from a credible source that indicates possible impairment of an individual's ability to work safely. ENN-NS-102 is required by 10 CFR Part 26.

Contrary to the above, for approximately four minutes on June 29, 2004, the on duty CRS was asleep in a chair in the control room and not fit for duty, and appropriate measures were not taken to relieve the CRS from duty and have him for-cause FFD tested.

These violations constitute a Severity Level III problem.
Civil Penalty - $60,000. (EA-05-039)

Pursuant to the provisions of 10 CFR 2.201, Entergy Nuclear Operations, Inc., is hereby required to submit a written [illegible]ent or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the [illegible]f this Notice of Violation and Proposed Imposition of Civil Penalty(ies) (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-05-039" and should include for each alleged violation: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, and if denied, the reasons why, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps that will be taken to avoid further violations, and (5) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown.

Within the same time as provided for the response required above under 10 CFR 2.201, Entergy Nuclear Operations, Inc. may pay the civil penalty proposed above or the cumulative amount of the civil penalties if more than one civil penalty is proposed, in accordance with NUREG/BR-0254 and by submitting to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, a statement indicating when and by what method payment was made, or may protest imposition of the civil penalty in whole or in part, by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the Licensee fail to answer within 30 days of the date of this Notice of Violation and Proposed Imposition of Civil Penalty, an order imposing the civil penalty will be issued. Should the Licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violation(s) listed in this Notice, in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty in whole or in part, such answer may request remission or mitigation of the penalty.

In requesting mitigation of the proposed penalty, the factors addressed in Section VI.C.2 of the Enforcement Policy should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or [illegible]ation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference [illegible]citing page and paragraph numbers) to avoid repetition. The attention of the Entergy Nuclear Operations, Inc. is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing a civil penalty.

Upon failure to pay any civil penalty due which subsequently has been determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282c.

The response noted above (Reply to Notice of Violation, statement as to payment of civil penalty, and Answer to a Notice of Violation) should be addressed to: Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region I, and a copy to the NRC Resident Inspector at Pilgrim Nuclear Power Station.

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

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

Dated this 14th day of July 2005

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

August 25, 2005

NRC INFORMATION NOTICE 2005-25: INADVERTENT REACTOR TRIP AND PARTIAL SAFETY INJECTION ACTUATION DUE TO TIN WHISKER

ADDRESSEES

All holders of operating licenses for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees about recent operating experience related to the growth of "tin whiskers" in electronic circuits at nuclear power stations. Recipients are expected to review the information for applicability to their facilities and consider appropriate actions to avoid similar problems. However, the measures suggested in this information notice are not NRC requirements and no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

On April 17, 2005, Millstone Nuclear Generating Station, Unit 3, experienced an unexpected safety injection actuation and reactor trip caused by a fault on a solid state protection system (SSPS) circuit card. The fault generated a false low steamline pressure signal, bypassing the 2-out-of-3 SSPS logic and causing the A safety train actuation and reactor trip. The licensee examined the failed circuit card using a magnifying glass and found a microscopic tin filament (approximately 2 mm long). The filament created a bridge between the affected diode and the output trace on the card. This microscopic filament of tin called "tin whisker," had grown out of the tin coating covering the leads of the diode.

The licensee inspected all circuit cards in the SSPS and discovered tin whiskers on other circuit cards. In each case, the whisker appeared to originate at the tin coating on diode leads. Suspect cards were either replaced or cleaned before being placed back in service. The licensee sampled additional circuit cards from other important plant systems but found no other evidence of tin whiskers.

BACKGROUND

Tin whiskers are electrically conductive crystalline structures of tin that sometimes grow from surfaces where pure tin (especially electroplated tin) is used as a final finish. Tin whiskers have

ML052150404

been observed to grow to lengths of several millimeters (mm) and in rare instances up to 10 mm. Electronic system failures have been attributed to short circuits caused by tin whiskers that bridge closely spaced circuit elements maintained at different electrical potentials.

Tin whiskers appear to have increased following international efforts to remove alloying metals such as lead from solder and other circuit card manufacturing materials to reduce environmental and health hazards. With the move toward lead-free electronics, tin has become a drop-in replacement for the tin-lead finish currently used for electrical component terminations. The move to lead-free electronics means that failures of some high-reliability components may continue to increase until a solution to the tin whiskers problem is found. Tin whiskers have been cited as the cause for various minor component failures in the nuclear industry and significant failures in the aerospace industry.

DISCUSSION

Some of the failures due to whiskers are documented in licensee event reports (LERs):

Plant	LER No.	NUDOCS Accession No.
Dresden Unit 2	50-237/1987-22	8709230145
Duane Arnold	50-331/1990-04	9005010072
Dresden Unit 2	50-237/1997-19	9801270112
South Texas Unit	50-499/1999-06	9910080186

In most of the events, metallic whiskers caused a short of the local power range monitors (LPRM) detectors resulting in a momentary spike on the average power range monitors (APRMs). In other cases, whiskers resulted in a failure of a channel input relay to the engineered safety features (ESF) actuation logic. In most cases, failure of the channel inputs in to the reactor protection system (RPS) or the ESF actuation did not result in a full RPS or ESF actuation. Only half of the RPS or ESF logic was met.

The incident at Millstone Unit 3 demonstrates that a single tin whisker can cause a protective feature to actuate. It is reasonable to assume that the same phenomenon could also prevent a protective system actuation. The extent-of-condition review performed at Millstone also showed that circuit cards need not be in service to be susceptible to whiskering. Research available from NASA's Goddard Space Center (http://nepp.nasa.gov/whisker) and Computer Aided Life Cycle Engineering (CALCE) at the University of Maryland supports this discovery and provides other valuable information on prevention techniques and growth mechanisms. While the information provided directly states that the exact mechanism for growth is unknown, common growth conditions and theories are discussed.

The data from the extent-of-condition review at Millstone Unit 3, NASA and CALCE information indicate that more than one manufacturer makes high-reliability circuit cards susceptible to tin whiskering. The data also indicates that tin whiskering is not significantly influenced by the environment in which the cards are used. Therefore, if one card procured from a specific vendor shows evidence of whiskering, all cards of that type from the same manufacturer can be expected to show signs of whiskering. In general, components containing 3% or greater lead concentration in the solder and/or manufactured with conformal coatings appear to be less susceptible to tin whiskering.

CONTACTS

This information notice requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/
Patrick L. Hiland, Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contacts: Paul Rebstock, NRR
301-415-3295
E-mail: pjr1@nrc.gov

Thomas Sicola, R-I/DRS
610-337-5109
E-mail: tps1@nrc.gov

Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

August 3, 2005

NRC INFORMATION NOTICE 2005-24: NONCONSERVATISM IN LEAKAGE DETECTION SENSITIVITY

ADDRESSEES

All holders of operating license or construction permits for nuclear power reactors, except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees that the reactor coolant activity assumptions for containment radiation gas channel monitors may be nonconservative. As a result, the containment gas channel may not be able to detect a 1 gallon-per-minute (1-gpm) leak within 1 hour. It is expected that the recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Several nuclear power plant licensees have reported problems with the detection capabilities of containment radiation gas channel monitors. The following gives several examples of these reports.

On May 2, 2005, the McGuire nuclear power plant licensee reported that the containment atmosphere radioactivity monitors were not sensitive enough for their intended function of detecting a 1-gpm reactor coolant system (RCS) leak within 1 hour (Licensee Event Report (LER) 50-369/2005-01, ADAMS Accession No. ML051310167). This resulted in a Severity Level IV noncited violation.

The McGuire licensee declared the atmosphere monitors inoperable and performed compensatory actions in accordance with plant technical specifications. The compensating actions were to (1) establish temporary alarm setpoints to provide earlier notification should a significant RCS leak occur, (2) instruct operators on other methods of RCS leak detection, (3) establish sensitivities as low as practical based on actual RCS radioactivity levels, (4) periodically review the sensitivities for revision as needed, (5) provide additional training as needed, and (6) consider submitting a license amendment request to clarify the capabilities of the leak detection instrumentation.

ML051780073

In February 2005, NRC inspectors at the Catawba nuclear power plant identified a noncited violation of Technical Specification 5.4.1.a, "Written Procedures," because the licensee failed to establish and maintain an adequate procedure for the required containment atmosphere radioactivity monitor surveillance in that the associated alarm function was not set or tested to alarm at a value equivalent to 1 gpm in 1 hour for a realistic current reactor coolant activity level (NRC Integrated Inspection Report 50-413/2005-02 and 50-414/2005-02, ADAMS Accession No. ML051160367).

The Catawba licensee also declared these channels to be inoperable and is performing compensatory actions in accordance with plant technical specifications.

In June 2003, an NRC inspection made a similar finding at Callaway (NRC Inspection Report 50-483/2003-04, ADAMS Accession No. ML032020562) that resulted in a noncited violation. The gas channel monitor was not capable of performing its design basis function of detecting a 1 gpm RCS leak within 1 hour. The calculation for the gas channel monitor response used an RCS source term corresponding to an assumed 0.1 percent failed fuel but, because of improved fuel performance and RCS chemistry control, the plant operated with an RCS source term several orders of magnitude smaller.

The Callaway licensee responded to this situation similarly by (1) declaring the gas channel out of service to prevent its being credited for leakage detection and (2) considering a license amendment request to revise the final safety analysis report and technical specification bases to reflect actual leakage detection capabilities.

DISCUSSION

The NRC requires licensees to use a means of detecting and, to the extent practical, identifying the location of any sources of RCS leakage (Title 10 of the Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," Criterion 30, "Quality of Reactor Coolant Pressure Boundary"). The NRC provided guidance on meeting GDC 30 in Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." Some licensees committed to using RG 1.45 as the basis for meeting GDC 30.

RG 1.45 states that an acceptable means would provide for adequate sensitivity and response time of all leakage detection systems to detect a leakage rate of 1 gpm in less than 1 hour. Further, the acceptable means would employ at least three separate detection methods. Two of these methods are monitoring sump level and sump flow and monitoring airborne particulate radioactivity. The third method is either monitoring the condensate flow rate from air coolers or monitoring airborne gaseous radioactivity. The guide also states that a "realistic" primary radioactivity concentration should be assumed when analyzing the sensitivity of leak detection systems.

During original plant licensing, the typical calculation for the technical specification for gas channel monitor response used an RCS source term corresponding to an assumed 0.1 percent failed fuel. Nowadays, because of improvements in fuel performance and RCS chemistry control, the actual RCS source term can be orders of magnitude smaller. Though desirable, a small source term can result in reduced leakage monitoring capabilities. Using a realistic RCS source term, a 1 gpm RCS leak would likely not be detected by a gas channel monitor for a much greater time than within 1 hour. The 0.1-percent failed fuel assumption introduces a nonconservatism into the technical specifications. Guidance on resolving such a nonconservatism is given in NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety."

The consistency of leakage detection systems with RG 1.45 has been questioned at several nuclear power plants. See NUREG/CR-6861, "Barrier Integrity Research Program," December 2004 (ADAMS Accession No. ML043580207) for a good discussion of detector sensitivities.

This information notice requires no specific action or written response. Please direct any questions about this matter to the technical contact(s) listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/ By David C. Trimble Acting For/
Patrick L. Hiland, Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contacts: Michael Peck, RIV
573-676-3181
E-mail: msp@nrc.gov

Vernon Hodge, NRR
301-415-1861
E-mail: cvh@nrc.gov

Note: NRC generic communications may be found on the NRC public Website, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

August 1, 2005

INFORMATION NOTICE 2005-23: VIBRATION-INDUCED DEGRADATION OF BUTTERFLY VALVES

ADDRESSEES

All holders for operating licenses for nuclear power reactors except those who have permanently ceased operation and have certified that fuel has been permanently removed from the vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to the degradation of butterfly valves supplied by Fisher Controls and other manufacturers. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

On February 10, 2005, Southern California Edison declared component cooling water (CCW) outlet isolation valve 2HV6500 for the Train B shutdown cooling (SDC) heat exchanger in Unit 2 at the San Onofre Nuclear Generating Station (SONGS) inoperable in response to an abnormal reduction in flow through the valve. Valve 2HV6500 is an 18-inch butterfly valve manufactured by Fisher Controls. The operability of the containment spray (CS) system at SONGS Unit 2 depends on the availability of the SDC heat exchanger. Therefore, the licensee started a manual shutdown of SONGS Unit 2 on February 14, 2005, to repair the valve.

The licensee disassembled the valve and found that it could not fully open as a result of losing two taper pins that connect the valve disc to the valve stem. During the original installation, the taper pins are impact-driven into holes in the valve disc and stem and are intended to be held in place by the interference fit. The licensee could not determine the exact cause of the loss of the taper pins during plant operation. As corrective action, the licensee installed new taper pins and staked the pins to the valve disc to make them more secure.

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Since 1993, five Fisher Controls 28-inch butterfly valves in the CCW systems of SONGS Units 2 and 3 have lost one of the taper pins used to connect the valve disc to the valve stem. The licensee has also found additional Fisher Controls butterfly valves with improperly seated taper pins during internal inspections.

The design of the Fisher Controls butterfly valves can allow leakage through the valve if a taper pin is lost. For example, SONGS experienced leakage rates of approximately 50 gallons per minute (gpm) through 28-inch Fisher Controls butterfly valves in the CCW system in 1998 and 2004. After disassembling the butterfly valves, the licensee identified the cause of the leakage as the loss of a single taper pin in each of the valves.

Taper pins that come loose from butterfly valves can be carried with the system fluid and interfere with the operation of other plant equipment. For example, one of the missing taper pins from 2HV6500 at SONGS Unit 2 became lodged in train "B" CCW pump manual discharge isolation valve 2HCV6509, which is normally locked open and closed only for maintenance purposes. After maintenance on the train "B" CCW pump, the licensee had difficulty opening 2HCV6509 because of the taper pin lodged in the valve.

The licensee plans to review all butterfly valves in safety-related applications where loss of valve function or leakage because of a missing taper pin cannot be tolerated. On the basis of the review, the licensee will determine which butterfly valves to inspect during the upcoming refueling outages at SONGS Units 2 and 3. As part of the valve inspections, the licensee will stake the taper pins in the butterfly valves to ensure the pins remain in place during plant operation.

DISCUSSION

Over the years, nuclear power plants have experienced vibration-induced degradation of plant equipment during operation at the original licensed power and under power uprate conditions. The NRC has issued several information notices on vibration-induced degradation of plant equipment. For example, the NRC issued Information Notice (IN) No. 83-70, "Vibration-Induced Valve Failures," on October 25, 1983, to alert nuclear power plant licensees to valve failures and system inoperability as a result of normal operational vibration.

The degradation of Fisher Controls butterfly valves as a result of the loss of their taper pins at SONGS Units 2 and 3 is another example of vibration-induced degradation during plant operations. There have also been problems with the taper pins that connect the valve disc to the stem in butterfly valves supplied by other manufacturers. In 1989 Turkey Point Nuclear Plant, Unit 4, lost taper pins in a 36-inch intake cooling water head isolation butterfly valve manufactured by the Henry Pratt Company. In 2003 Davis-Besse Nuclear Power Station, Unit 1, lost taper pins in a 10-inch decay heat cooler butterfly valve with the brand name Valtek marketed by the Flowserve Corporation.

Depending on the valve design, the loss of a taper pin from a butterfly valve may result in significant leakage through the valve before interfering with valve operation. The size of the taper pin and fluid conditions can cause the leakage limits for the applicable plant system to be exceeded. In addition, leakage through a valve can be masked by another closed valve in the system until the second valve is opened.

Taper pins that come loose from butterfly valves can be carried with the system fluid and interfere with the operation of other plant equipment. The example of 2HCV6509 at SONGS Unit 2 had low safety significance because this valve is only used for maintenance at the plant.

Some nuclear power plants have experienced more severe vibration-induced degradation of equipment under power uprate conditions. For example, the NRC staff described vibration-induced degradation of plant equipment during power uprate operation in IN 2002-26, Supplement 2, "Additional Flow-Induced Vibration Failures After a Recent Power Uprate" (January 9, 2004). Increased steam and feedwater flow during power uprate operation can increase vibration of plant equipment, including valves and valve actuators. The higher vibration levels can impact the appropriate inspection intervals for some plant components.

In summary, degradation of butterfly valves supplied by Fisher Controls and other manufacturers has occurred during plant operation as a result of the loss of taper pins used to connect the valve disc to stem. The degradation can involve leakage and affect valve operation. Taper pins lost from butterfly valves can also interfere with the operation of other plant components in fluid systems. The cause of the loss of valve taper pins is not known for certain, but operating experience suggests that the most likely cause is vibration-induced degradation. Staking the taper pins after their installation in the butterfly valves is one method of providing a more secure interference fit of the pins. The increased steam and feedwater flow during power uprate operation can accelerate vibration-induced degradation of plant equipment, including valves and valve actuators.

RELATED GENERIC COMMUNICATIONS

NRC Information Notice 83-70, "Vibration-Induced Valve Failures," October 25, 1983.

NRC Information Notice 2002-26, Supplement 2, "Additional Flow-Induced Vibration Failures After a Recent Power Uprate," January 9, 2004.

This information notice requires no specific action or written response. However, recipients are reminded that they are required by 10 CFR 50.65 to consider industry-wide operating experience (including information presented in NRC information notices) where practical, when setting goals and performing periodic evaluations.

CONTACT

Please direct any questions about this matter to the technical contact(s) listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/D.C. Trimble for/
Patrick L. Hiland, Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical contacts:
Thomas G. Scarbrough, NRR
(301) 415-2794
E-mail: tgs@nrc.gov

Mark A. Sitek, Region IV
(949) 492-2641
E-mail: mas3@nrc.gov

Ronald V. Schmitt, NRR
(301) 415-4082
E-mail: rvs@nrc.gov

Note: NRC generic communications may be found on the NRC public Website, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
WASHINGTON, D.C. 20555

July 29, 2005

NRC INFORMATION NOTICE 2005-22: INADEQUATE CRITICALITY SAFETY ANALYSIS OF VENTILATION SYSTEMS AT FUEL CYCLE FACILITIES

ADDRESSEES

All licensees authorized to possess a critical mass of special nuclear material.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to alert addressees to a safety concern arising from inadequate criticality safety analysis of ventilation systems at fuel cycle facilities. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this IN are not new NRC requirements; therefore, no specific action nor written response is required.

DESCRIPTION OF CIRCUMSTANCES

Recently, two events occurred at NRC-licensed fuel cycle facilities involving the failure to implement criticality safety controls on process off-gas or ventilation systems when minor differences between otherwise similar systems, analyzed under a single broad criticality analysis, were not recognized by criticality safety analysts. The first instance was noted subsequent to a backflow event in an off-gas line from a uranium dissolver. The licensee used a single criticality safety analysis for ventilation systems in the facility. The ventilation analysis took credit for off-gas piping typically having either a siphon break and a drain, or two drains.

However, a concern about off-gas accumulation in an enclosed area led to a design modification for the off-gas line on the uranium dissolver such that only one drain was in the system. During preparation of the facility criticality safety analysis, criticality safety analysts failed to recognize that the design difference defeated the siphon break so that double contingency was not established.

The second instance was noted when a fuel cycle licensee observed an accumulation of uranium dioxide powder in a high-efficiency particulate air (HEPA) filter housing where no uranium was expected. The licensee determined that what criticality safety analysts thought was a breathing air-ventilation system was also connected to a process off-gas line from a hood on a uranium oxidation furnace. The licensee identified a design difference in the system in that ventilation and off-gas lines were connected differently, as they approached the HEPA filter, than was customary in the remainder of the plant. The licensee had several broad criticality safety analysis packages related to ventilation and process off-gas, grouping them as

ML051890406

breathing air, dry off-gas, and wet off-gas. The criticality safety analyst's failure to recognize the design difference in duct connections in one part of the plant from other areas of the plant led to the incorrect determination that the system was breathing air and criticality was not credible. This incorrect determination resulted in the failure to implement criticality safety controls typical for off-gas ventilation in the plant.

DISCUSSION

Under 10 CFR Parts 70 and 76, certain licensees processing, storing, or handling critical masses of fissile material are required to analyze accident scenarios leading to criticality and provide reliable controls to assure that inadvertent criticality events are highly unlikely. When processes, systems, equipment, or procedures are repeated in a facility, licensees frequently elect to combine similar processes, systems, equipment, or procedures into a single criticality analysis. The safety concern arises when modifications resulting in minor design differences between otherwise similar systems defeat the credited double-contingency arrangement or non-credibility determination.

In the two events described, the two licensees used a single criticality safety analysis to develop controls for groups of ventilation and process off-gas systems that were similar in form and function. While crafting the analyses, developing the criticality safety controls, and implementing the credited controls, licensee criticality safety analysts failed to recognize design differences between the systems that defeated some of the assumptions or credited controls used in some portion of the facility.

In the first instance, a design change occurred, during construction of the system, that involved placing an additional column into the system that effectively defeated the siphon break for the uranium dissolver. The criticality safety review for this design change looked at the analysis for the process, but did not consider the impact that the change would have on off-gas ventilation. In the second instance, contractors were constructing a new facility, and criticality safety analysts did not recognize design differences in the ventilation system.

Minor design changes during construction of new processes or facilities are common at fuel cycle licensees and may have a subtle effect on criticality controls. Licensees should consider actions, as appropriate, to mitigate this vulnerability. These actions could include reviewing all criticality safety analyses that group similar systems, to assure that all assumptions regarding the forms and functions of the systems are valid for all applications. Actions could also include verifying that the design change review process is adequate to trigger an in-depth criticality safety review for changes arising during construction.

The Part 70 integrated safety analysis (ISA) and the Part 76 safety analysis report (SAR) provide an integrated approach to assure that inter-relationships between accident scenarios and their controls are appropriately evaluated during related design and change activities. Licensees should consider whether their ISA/SAR provides an adequate integrated review of ventilation and related systems.

This IN requires no specific action nor written response. If you have any questions about the information in this notice, please contact the technical contact listed below.

/RA/

Robert C. Pierson, Director
Division of Fuel Cycle Safety
and Safeguards
Office of Nuclear Material Safety
and Safeguards

Technical Contact: Dennis Morey, NMSS
301-415-6107
E-mail: dcm@nrc.gov

Attachment: "List of Recently Issued NMSS Generic Communications"

Recently Issued NMSS Generic Communications

Date	GC No.	Subject	Addressees
07/13/05	RIS-05-13	NRC Incident Response and the National Response Plan	All licensees and certificate holders.
07/11/05	RIS-05-11	Requirements for Power Reactor Licensees in Possession of Devices Subject to the General License Requirements of 10 CFR 31.5	All holders of operating licenses for nuclear power reactors and generally licensed device vendors.
06/10/05	RIS-05-10	Performance-Based Approach for Associated Equipment in 10 CFR 34.20	All industrial radiography licensees and manufacturers and distributors of industrial radiography equipment.
04/18/05	RIS-05-06	Reporting Requirements for Gauges Damaged at Temporary Job Sites	All material licensees possessing portable gauges, regulated under 10 CFR Part 30.
6/23/05	IN-05-17	Manual Brachytherapy Source Jamming	All medical licensees authorized to possess a Mick applicator.
05/17/05	IN-05-013	Potential Non-conservative Error in Modeling Geometric Regions in the Keno-v.a Criticality Code	All licensees using the Keno-V.a criticality code module in Standardized Computer Analyses for Licensing Evaluation (SCALE) software developed by Oak Ridge National Laboratory (ORNL)
05/17/05	IN-05-012	Excessively Large Criticality Safety Limits Fail to Provide Double Contingency at Fuel Cycle Facility	All licensees authorized to possess a critical mass of special nuclear material.

Note: NRC generic communications may be found on the NRC public website, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

July 21, 2005

NRC INFORMATION NOTICE 2005-21: PLANT TRIP AND LOSS OF PREFERRED AC POWER FROM INADEQUATE SWITCHYARD MAINTENANCE

ADDRESSEES

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

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees about loss of power events as a result of inadequate preventive and corrective maintenance practices on switchyard breakers and current transformers. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

On May 5, 2004, Dresden Unit 3 was at full power and Dresden Unit 2 was shutdown when an automatic reactor scram and a subsequent loss of offsite power event occurred during activities to reconfigure breakers in the 345 kV switchyard. Operations personnel manually opened switchyard breaker 8-15 in accordance with the switching order. However, when the A and B phases opened, the C phase of switchyard breaker 8-15 failed to fully open within the required time. This failure produced current imbalances in Unit 2 and Unit 3 switchyard ring busses (tied together through a breaker), which led to the opening of several other switchyard breakers. Unit 3 scrammed due to turbine load reject, and offsite power was lost to the Unit 3 safety-related emergency core cooling system (ECCS) busses. The failed breaker was an I-T-E Imperial Corporation (current vendor ABB) sulfur hexafluoride (SF6) gas circuit breaker (type 362GA). This breaker used independent pole operators for each of the three phases. The breaker was built and installed in the Dresden 345 kV switchyard in the late 1970's.

On May 6, 2004, the licensee and personnel of the transmission and distribution company, Exelon Energy Delivery (EED), discovered that ABB, the current breaker vendor, had issued a product advisory in July 2003 for I-T-E Imperial Corporation GA and GB breakers to warn that the operating mechanisms may experience delayed trip or in some cases failures to trip due to age and application related problems. In addition, the advisory noted that the breakers at

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highest risk were those operated less than twice per year. The product advisory recommended that the operating mechanism in high-risk applications be rebuilt using new trip latch mechanism kits at the earliest convenience.

While disassembling the trip latch mechanism of Breaker 8-15, EED and licensee personnel discovered that the sealed bearing for the trip latch mechanism did not roll freely. The failure of the sealed bearing to roll freely, directly contributed to the failure of the C phase of Breaker 8-15 to open within the required time. The NRC special inspection team reviewed the maintenance history of Breaker 8-15. The last preventive maintenance on Breaker 8-15 was done on March 27, 2002, and included routine inspection, lubrication and maintenance, a contact resistance test, and a travel timing test. The inspection team noted that the breaker failed the timing test on the C Phase. The breaker was last cycled in October 2002 and then remained in the closed position until May 5, 2004.

The NRC inspection team noted that the EED procedure stated that the breaker should be lubricated after a failed timing test. However, the vendor manual stated that, the operating mechanism should be disassembled and cleaned and lubricated when the operating mechanism showed signs of difficult or sluggish operation. In addition, the manual stated that under ordinary circumstances, the life of the grease in sealed bearings should be at least 10 years and that if oxidation of the lubricant made the bearing sluggish, the bearing must be replaced. The EED preventive maintenance program and procedures for breakers did not include routine replacement of worn out breaker parts. In addition, the EED maintenance procedures did not instruct maintenance personnel to disassemble sluggish operating mechanisms to check for degraded bearings, nor did the procedures specify the appropriate lubricants for the various parts of the breaker.

On June 12, 2002, with DC Cook Unit 1 at approximately 68% power and Unit 2 at 100% power, an emergency alert condition was entered after a catastrophic failure and resultant fire of a current transformer for the 345 kV switchyard L breaker. The catastrophic failure of the current transformer and the subsequent switchyard switching actions resulted in the loss of the preferred offsite power source to Units 1 and 2. On June 19, 2002, the NRC special inspection team reviewed the licensee's preventive maintenance program for 345 kV switchyard current transformers. The vendor's preventive maintenance recommendations included annual inspections and transformer oil analysis every 2 years. The inspection team reviewed historical maintenance activities on the L breaker current transformers and determined that preventive maintenance activities were last done in October 1998. The periodicity of preventive maintenance activities was consistent with American Electric Power (AEP) system guidelines, but not with the vendor's recommendations. Additionally, the licensee did not periodically perform several vendor-recommended tests, including tests of oil dielectric strength and oil acid factor, and a measurement of the resistance of the current transformer primary (to compare with the results in the test report). During followup discussions, licensee personnel stated that the types of testing performed and the testing frequencies were based on AEP system operating experience rather than vendor recommendations. Licensee personnel were unable to readily provide specific operating experience data that justified the 4-year preventive maintenance testing frequency. Licensee personnel subsequently determined that there were approximately one hundred twenty six 345 kV current transformers in the AEP system similar in design to the transformers located in the DC Cook 345 kV switchyard. Since 1990, there have been two catastrophic failures (both associated with the D. C. Cook 345 kV switchyard L breaker). No current transformers of this type had been removed from service based on preventive maintenance testing.

Following the June 12, 2002, current transformer failure, AEP collected oil samples from the D.C. Cook 345 kV switchyard breaker current transformers for analysis. The oil analyses were completed 3 months before the normal schedule as part of the licensee's extent-of-condition evaluation. During the oil sampling, AEP personnel discovered that two current transformers for N1 switchyard breaker were last sampled in September 1998, with gas analyses results significantly above the acceptable level. Based on this result, licensee replaced the N1 breaker current transformers and returned the breaker to service on June 29, 2002. The AEP system operating experience data did not justify a less frequent analysis than recommended by the vendor.

DISCUSSION

The discrepancies, between the licensee's maintenance practices for switchyard breaker and current transformers and the vendor recommendations, contributed to the inadvertent switchyard breaker trips that resulted in a plant trip and loss of offsite power (LOOP) to safety busses. Unnecessary plant trips and LOOP events could be reduced by following vendor recommendations with feedback from operating experience to determine the appropriate schedule and extent of maintenance.

CONTACT

This information notice requires no specific action or written response. Please direct any questions about this matter to the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/
Patrick L. Hiland, Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contact: Thomas Koshy, NRR
301-415-1176
E-mail: txk@nrc.gov

Allan Barker, RIII
630-829-9679
E-mail: arb3@nrc.gov

NRR Project Manager: Richard Laura, NRR
301-415-1837
E-mail: ral1@nrc.gov

Note: NRC generic communications may be found on the NRC public Website, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

Inside NRC

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NRC's Office of Nuclear Reactor Regulation reorganizes for new work

Largely in response to an expected surge in new reactor licensing work, the NRC is undertaking a major reorganization of the Office of Nuclear Reactor Regulation (NRR).

Within days of receiving the staff's proposal for the changes, the commission approved a plan to flatten the office by broadening the number of divisions and eliminating a layer of management. The staff said the last major reorganization was initiated in March 1999 to provide better accountability of program activities.

Under the plan approved Aug. 25 and publicly announced Sept. 1, NRR's programs and processes are expected to improve. It also is expected to better align NRR for risk-informed regulatory activities.

The reorganization will remake the existing five large technical divisions—regulatory improvement programs; inspection program management; licensing project management; systems safety and analysis; and engineering—into nine smaller divisions. Those divisions will be: safety systems; component integrity; engineering; license renewal; operator reactor licensing; inspection and regional support; new reactor licensing; risk assessment; and policy and rulemaking.

The existing five technical divisions now have more than 100 people each.

The goal is to split each division into 70-80 people.

The number of first-line supervisors would increase, said NRR Deputy Director William Borchardt. Their titles as section chiefs will change to branch chiefs, he said. But the number of senior executive service (SES) positions will be reduced

Reorganization ...
to 24 from 30. No layoffs are expected, nor will anyone's grade level be downgraded, Borchardt said. The reductions are expected to occur through attrition and until then, some branches may have two deputies, he said.

Two additional associate director (AD) positions would be created, for a total of three in NRR. Currently there is only one, Brian Sheron, who is associate director for project licensing and technical analysis.

Borchardt said last week that the agency was still in the selection process for filling many of the posts. He said Jim Dyer would remain NRR director and that he would stay in his deputy director post. The ADs are expected to be announced soon, followed by those who will fill the division director positions, Borchardt said.

Last week, two names circulating around the agency as the possible new ADs were Gary Holahan, executive assistant for reactors and research in Chairman Nils Diaz's office, and Bruce Boger, director of the division of inspection program management.

The staff-to-supervisor ratio is expected to be greater than or equal to 8.5-to-1 after the reorganization, although the ratio will be less than that (with fewer staff to supervisors) during the transition period, the staff said in its proposal (Secy 05-146).

Currently there are about 600 NRR employees. NRC plans to hire 350 "new entry-level and experienced employees" by the end of 2006, the agency said in a Sept. 2 press release. Borchardt said last week that about 50 people are expected to be hired next year in NRR. He said the branches will grow over time to accommodate the additional staffers. "There will be a proportionate increase in first-line branch chiefs" as the agency's size increases, he said.

Meanwhile, NRC's Office of Administration is working with the U.S. General Services Administration to find additional office space for the staff, either within the two buildings at its Rockville, Md. campus or in leased space elsewhere. One NRC staffer said one way to make more room for employees is to move the contractors out of headquarters.

In its Aug. 25 staff requirements memorandum, the commission instructed the staff to ensure that the new hires have sufficient oversight and mentoring when they join the agency. The commissioner also told NRR managers "not to lose sight of the day-to-day business of overseeing the currently operating reactors" as the workload for new reactors increases.

In his Aug. 16 vote sheet, Diaz called the reorganization "a sound plan to address the expected growth of NRR in the coming years." He said he wanted the changes to be implemented by the start of the new fiscal year, Oct. 1, although the final decision allows the transformation to take place by Oct. 30.—*Jenny Weil, Washington*

Waterford-3 escapes storm damage but remains down because of grid

Entergy Operation's Waterford-3 remained shut down last week in the aftermath of Hurricane Katrina as federal officials worked through a checklist of restart requirements while fixes were being made to the electrical grid.
The plant was not expected to restart until sometime this week, NRC officials said.

Entergy declared an unusual event Aug. 27 after a hurricane warning was issued in the plant's locale, St. Charles Parish, La., and it began shutting the plant down the following day. The storm blew in Aug. 29, and though the plant escaped harm, the surrounding area and key infrastructure suffered from wind and water damage, particularly after two levees protecting New Orleans broke. The plant is about 20 miles west of the city.

In the aftermath, external communication systems were wiped out. Entergy reported to NRC Aug. 31 that it had lost the communications system to its Emergency Operations Facility, reactor auxiliary building, and emergency notification system. The utility said, however, that it had limited capability to use an "industrial hotline to a circuit" in St. Charles Parish. It was communicating with the NRC over satellite telephones, NRC and Entergy officials said.
Once the National Weather Service lifted the hurricane warning for the parish late Aug. 29, the plant still could not return to service because the off-site power was lost. Over the next few days, voltage fluctuation contributed to the instability of the electrical grid, forcing Entergy to keep the plant out of service. An NRC spokesman said Sept. 1 that there was power at the grid if Entergy needed to reconnect Waterford-3 in an emergency. But as a precautionary move, the plant remained unconnected, the spokesman said.
An extra shipment of fuel for the plant's emergency diesel generators, which have been running key safety systems, arrived at the site Aug. 30.

Waterford-3 cannot restart without approval of the NRC (Nucleonics Week, 1 Sept., 4). While safety equipment was not affected by the storm, the condition of emergency evacuation routes and operability of sirens were a concern, NRC and Entergy officials said.

NRC must ensure the adequacy of off-site emergency preparedness before giving Entergy the okay to restart Waterford-3. Input from the Department of Homeland

Security's Federal Emergency Management Agency and Louisiana state regulators will factor heavily into NRC's decision. NRC staff is following protocol under the agency's inspection guidelines Manual Chapter 1601, which spells out the procedures after a natural disaster, malevolent act, or extended plant shutdown. NRC staff said the guidelines were put in place following Hurricane Andrew in August 1992.

Ten years earlier, in December 1982, about 17,000 residents of St. Charles Parish had to be evacuated. But the exodus was not because of any incident connected to Waterford-3, which did not begin operating until September 1985. The evacuation was prompted by a leak at a chemical plant. State and local officials used a draft emergency preparedness plan for Waterford-3 to accomplish the evacuation, according to the Nuclear Energy Institute.

The day after Hurricane Katrina, both Entergy and NRC were making preparations for sending relief workers to the plant. NRC Region IV spokesman Victor Dricks said additional NRC employees were sent to the site Aug. 31 to provide a break to the resident inspectors and two additional regional staffers there. Entergy officials also made arrangements to provide relief workers and expand the staff on site for the recovery stage. During the storm, there had been two shifts of "core" team workers stationed at the plant, said Entergy spokesman Mike Bowling.

Two other Entergy plants affected by the storm, River Bend, which is about 24 miles northwest of Baton Rouge, La., and Grand Gulf, about 25 miles south of Vicksburg, Miss., had to reduce power because of electrical grid fluctuations and lower load demand, said Entergy spokesman Tim Chrisler.

By Sept. 2, both units had powered up to 97%. Standard & Poor's Ratings Services (S&P) put Entergy Corp. and its subsidiaries on its CreditWatch with negative implications Aug. 31 because of concerns about the company's credit quality. S&P is one of The McGraw-Hill Companies, as is Platts.

S&P said in a new release that the negative watch "reflects the potential that Entergy's underlying business may have been irreparably harmed by the devastation wrought by Hurricane Katrina." Changes in the CreditWatch listing will depend on such factors as restoration cost estimates, economic viability of the region, the level of responsiveness from state authorities, and the timeline for Entergy

to recover its storm costs, it said.
Entergy acknowledged Sept. 1 that there would be "a long and difficult restoration process." It said the storm was the worst in the company's history, knocking out the electrical system in New Orleans and causing extensive damage to the electrical system in other parts of Louisiana and Mississippi.—*Jenny Weil, Washington*

Debate ensues on long-term future of risk-informed regulation

NRC staff and industry representatives exchanged views at an Aug. 25 meeting on the future direction of riskinformed regulation, particularly as it relates to a "technology-neutral" licensing framework for new reactors now being developed by NRC and scheduled for completion next year (INRC, 21 March, 17).

Adrian Heymer, director of plant performance improvement at the Nuclear Energy Institute (NEI), proposed "an optional alternative" to current 10 CFR Part 50 licensing regulations, which would "build on" current regulations and NRC's reactor oversight process. As a "parallel activity," NRC staff should develop an advanced notice of proposed rulemaking with "a complete set of example regulations" by June 2006, Heymer said.

After a public comment period and revisions, draft regulations could be published "for information purposes" by June 2008, Heymer said. Industry can then "assess how the proposed regulations match up with certified designs and applications under review" for new reactor licensing, including combined construction permit-operating licenses, or COLs, he said. Revised regulations could be published for comment by 2012 and finalized by 2015.

NRC "may be nearing the end of risk-informing one regulation at a time," Anthony Pietrangelo, senior director of risk regulation at NEI, said at the meeting. Rulemaking on 10 CFR 50.46 cooling requirements and 10 CFR 50.69 on risk-informed special treatment requirements "enable quite a bit" of the benefits to licensees of risk-informed approaches without further initiatives, Pietrangelo said.

"Near-term" applicants for new plant licenses "don't have enough time to wait for a new regulatory framework," which would cover non-LWR designs, he said.
On the issue of whether a future risk-informed Part 50 should be technology-neutral, Heymer said "we're not smart enough at this stage" to know which specific reactor designs to focus on. NRC should "keep regulations technology-neutral, and have technology-specific guidance documents" to support them, he said.

Mark Rubin of NRC's probabilistic risk assessment branch offered a different view, saying that "until the technology-neutral framework is tested, [industry] will need a set of regulations for design and NRC staff will need them for regulation." Rubin cautioned against "a halfway framework," saying that "applicants in the interim are going to need some guidance. Otherwise, you are going to put us on a very difficult regulatory and design slope that is going to very painful."

Rubin said that advanced LWRs could be licensed "in a risk-informed way" using current Part 50 and 52 regulations, as could gas-cooled reactors "with some exemptions, ingenuity and flexibility." Given NRC staff resource constraints, the agency "has to decide where to put resources when an LWR focus would delay work on specific guidance needed for other designs," Rubin said. But a representative of Framatome ANP said his company did not believe that a gascooled reactor licensed under existing regulations could be "commercially viable. That's why we're focusing on the riskinformed approach."

Gareth Parry of NRC's division of system and safety analysis noted that the concept of a design basis accident in current regulations is closely tied to LWR technology and "could cause some problems" if NRC does not "have a clear picture" of how the concept will translate to non-LWR designs before regulations are developed.

NRC is now pursuing "two parallel paths," developing a regulatory structure for new plant licensing based on a technology-neutral framework and working on "activities to riskinform specific regulations within the current 10 CFR Part 50," Mary Drouin of NRC's Office of Nuclear Regulatory Research said at the meeting (INRC, 25 July, 6). Staff plans to present a draft framework to the Advisory Committee on Reactor Safeguards sometime this fall, with a staff plan due to the commission in December, she said.
—Steven Dolley, Washington

NEI seeks input for update on license renewal guidance

Though the public comment period has ended, the industry has further suggestions for revising NRC's license renewal guidance documents for power reactors, the Nuclear Energy Institute (NEI) told NRC staff during an Aug. 31 teleconference. NEI will provide some "editorial comments" on the update of NRC's generic aging lessons learned document (GALL), the latest draft of which was released last month, and will likely make a presentation at a Sept. 9 Advisory Committee on Reactor Safeguards (ACRS) meeting, James Ross of NEI said at the meeting. NRC's Jake Zimmerman asked NEI for advance notification of issues they planned to raise at the ACRS meeting so that appropriate agency staff could be there to address them.

There are probably "no big-ticket items on the mechanical side" remaining to be resolved before the GALL revision is published Sept. 30, Ross said, but industry wants to discuss some electrical systems issues.

NRC plans to reopen scoping for the update of its generic environmental impact statement for license renewal, work on which "had slowed to essentially a stop" since 2003 due to resource constraints, NRC's Andy Cooper said at the meeting. The process will resume in fiscal 2006 with plans to issue a notice for an additional scoping period, he said. A "fairly lengthy process" is needed to update the document, and some changes to 10 CFR 51 (environmental protection regulations for licensing) may be required, which would necessitate rulemaking that could extend into 2008, Cooper said.

NRC staff will use conference calls and quarterly meetings to keep industry apprised of the agency's plans to develop interim staff guidances (ISGs) on specific license renewal issues, Zimmerman said. Noting industry's objection that it gets little or no opportunity for input before ISGs are released in final form, Zimmerman said NRC staff "will engage [industry] earlier in the process as these issues come up" so that "the first time you hear about an ISG will not be when we issue it."

Industry and NRC staff agreed that the agency's use of site audits of aging management programs, initiated on a pilot basis in 2003 (INRC, 25 July, 9), has improved the license renewal review process. NEI will provide NRC with industry's assessment of the audit process later this month, Ross said. The reduction in staff requests for additional information (RAIs) due to audits is welcome but has not yet resulted in a commensurate reduction in agency fees for license renewal reviews, Ross noted.

While NRC is "keeping an eye on costs," the use of additional contractors and agency staff required by an increased number of renewal applications limits savings that can be realized, said P.T. Kuo, director of NRC's license renewal and environmental impacts program.

NRC is developing an internal audit guidance document for team leaders to improve the license review RAI process, Zimmerman said. An industry request to review draft audit reports prior to publication would add about a month to the review process, but NRC is considering the proposal, he said. Kathy Weaver, senior project manager in NRC's license renewal program, reminded industry that staff reviews license renewal applications for "sensitive, unclassified information" that might be "useful to a potential adversary or terrorist" prior to making the applications public. While no safeguards information has been submitted, some licensees have included detailed plant drawings that "showed all the vehicle barriers, BREs [bullet-resistant enclosures], fences and gates," which is "way beyond just a general layout drawing," Weaver said. NRC's preference is for licensees not to provide information "if it's not needed," Kuo said.

The next NRC-industry license renewal meeting is tentatively scheduled for Sept. 21. The latest GALL update and other license renewal documents are available on NRC's Web site at http://www.nrc.gov/reactors/operating/licensing/renewal/guidance.html.—*Steven Dolley, Washington*

Final RIS on SCWE drops language seen as prescriptive, regulatory

NRC last week issued the final version of its regulatory issue summary (RIS) on safety conscious work environment (SCWE) after adjusting the language in response to industry comments that the previous version was too prescriptive and could be seen as de facto regulation.

As the Aug. 25 RIS (2005-18) notes, NRC does not have regulations on SCWE. NRC staff had previously referred to the guidance as a "best practices" document, but that terminology had drawn objections from the industry (INRC, 13 Dec. '04, 7). Lisamarie Jarriel, the agency allegations adviser, said in an interview last week that the term "best practices" had been dropped in part because it "implies it's the best you can do."

The new RIS states that "advances beyond the practices described herein may be developed as industry practices in the area of SCWE mature and as licensees and their contractors strive for excellence and creativity." The RIS continues, "The NRC encourages such advances and provides the attached guidance not as a prescriptive definition of a SCWE but as a sample of practices which have been effective in some situations."

Expanding on language from the earlier version, which was published in the Oct. 14, 2004 Federal Register, the RIS says, "The NRC recognizes that some of these practices may not be practical for every licensee or contractor, depending on the existing work environment, the size, complexity, and hazards of the licensed activities, and/or other organizational factors, and that licensees and contractors have discretion regarding the manner in which a SCWE is maintained at a particular facility."

Jarriel said some of the RIS suggestions might be "overkill" for many licensees, especially those that do not operate reactors. One change the NRC staff "went to great pains" to make was to revise language saying licensees "should" do something, she said.

Nuclear Energy Institute (NEI) Vice President of Operations Michael Coyle said in an Aug. 31 interview the language in the new RIS was acceptable. However, he said, as with any generic communication, the question is "what [NRC staffers] try to do with it."

One improvement in the new document, he said, is language clarifying that the RIS is not a checklist for NRC inspectors or the basis for a performance indicator under NRC's Reactor Oversight Process (ROP). "No single indicator is sufficient in itself to identify weaknesses in the SCWE, nor are there absolute measurements that indicate an unhealthy environment," the RIS says.

In a separate document, the NRC staff said it was "in the process of adding clarifying guidance to relevant inspection guidance to specify that inspectors are not to use the RIS in assessing licensee performance." That document, released at the same time as the RIS, provided specific responses to the comments received on the draft RIS.

In general, Coyle said he saw little in the RIS that is "a whole lot different from what we do today."
Jarriel said, "There's a handful of people that are doing most of the stuff" described in the RIS, and many "are doing some of it." She added, "I would guess there are some that are not doing any of it." Many of the actions described in the RIS may be new to some licensees, such as medical licensees, she said.

But even for larger licensees, there may be new elements, she said. For example, licensees may believe their processes are effective but they may not have anything in place to monitor the effectiveness of those measures, she said.
In laying out the elements of a SCWE, the RIS says, "One factor that can significantly impact a SCWE is management behavior" and suggests including discussion of that point in SCWE training. The RIS lists behaviors that "may be effective at establishing and maintaining a SCWE," including an open door policy, awareness of employees' potential reluctance to raise concerns, and an understanding of the importance of identity protection. Such behaviors also include awareness of signs of a chilled environment and of "situations that may make [employees] less receptive to safety concerns, such as operational or maintenance goal pressures," the RIS says.
Jarriel said it is important for managers not just to talk about SCWE but to demonstrate that their commitment to it. She added, "People will act on what they believe." If employees believe that managers only want to hear about saving money, that is the issue employees will raise, she said.

Striking a balance
While the RIS cites the importance of expressing appreciation for employees who raise SCWE concerns, it also points to a balance that sometimes must be struck. In some cases, such appreciation could "inadvertently discourage" reporting

if the employees do not want recognition, it says.
The RIS also describes another balance, between holding employees accountable for their errors and maintaining an environment that is "conducive to the self-reporting of errors." Jarriel described a situation—without naming the plant—where an employee received a bonus for "pushing" a long-standing "legacy" issue to resolution. However, she said, the issue resulted in a finding of high safety significance by the NRC—an event that, under another provision of the plant's contracts, made ernployees ineligible for bonuses.

In such situations, she said, a company might consider allowing for bonuses if employees took prompt and comprehensive correction actions in response to the finding.
Similarly, if a self-reported error results in docking an employee's pay, the financial penalty imposed by the company might be reduced if the employee is part of the corrective action and agrees to brief colleagues on the error, she said.
The RIS also suggests tracking allegations as an indicator of SCWE. For large licensees, the RIS says, the number and trend of allegations made to NRC, compared to the number and trend of concerns raised within the licensee's organization "may be an indication of employee willingness to raise concerns internally." Licensees may track that information through an employee concerns program (ECP).

But Jarriel acknowledged there were limits to the use of allegations as an indicator because licensees do not have full access to that information. However, she said, if the NRC sees a trend in allegations from a site and the licensee does not see it through internal signals such as the ECP or the corrective action program, the agency will alert the licensee to the trend. Also, she said, NRC forwards allegations to licensees if the allegers approve of the agency's doing so.
She said she and her colleagues were still struggling with the issue of how to provide licensees with as much information as possible on allegations while preserving the confidentiality of allegers.

The RIS is available on NRC's Web site at http://www.nrc.gov/reading-rm/doc-collections/gencomm/reg-issues/2005/ri200518.pdf. The NRC response to comments on the draft RIS is available on NRC's electronic library Adams under accession number ML051100166.
Meanwhile, in a separate but related safety culture effort, NRC is planning a teleconference with industry officials as a follow-up to an Aug. 17 public meeting on incorporating safety culture into the ROP (INRC, 22 Aug., 1). NEI's Coyle said the teleconference, tentatively scheduled for this week,

is to include NEI, the Institute of Nuclear Power Operations, and possibly an industry executive.

Isabelle Schoenfeld of NRC's Office of Enforcement said the teleconference would not include non-industry stakeholders, but that there would be similar opportunities in the future for such participants.—*Daniel Horner, Washington*

Inside NRC

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Early site permit review schedules slip, but utilities expect little impact

NRC said last week that the review schedules for the first round of early site permit (ESP) applications would slip between four and nine months, but that the delays would have little impact on its first-of-a-kind new plant licensing work.

The unexpected delay, attributed primarily to the need to sort through thousands of comments the agency received on its three preliminary environmental reviews, pushes the targeted timeframe for a commission decision on the ESP applications to 32-39 months.

Initially, NRC staff estimated it would take about 30 months from the application submittal date to the granting of the permit. But later it revised the estimate to about 33 months—roughly breaking down to about 21 months for the safety evaluation and environmental review portions and 12 additional months for completing the mandatory hearing. More recently, some staffers had upped the estimate to 37 months for more complex applications.

The three ESP applications under review are for Dominion's North Anna site, Exelon Generation's Clinton site, and System Energy Resources Inc.'s (SERI) Grand Gulf site. SERI is a subsidiary of Entergy Corp.

Laura Dudes, chief of the new reactors section in NRC's Office of Nuclear

Reactor Regulation, said the number of comments sent to the agency took the staff by surprise. "Our previous experience had been with license renewal environmental impact statements, and we had not received anywhere near the number of comments," she told reporters in an Aug. 16 teleconference. "It's not the nature of the issues," Dudes said, "but we are required—and it's our job—to appropriately review each and every comment and provide a response to those comments."

New schedules

In an Aug. 16 letter to Dominion, William Beckner, program director of the New, Research and Test Reactors Program, said 1,300 people provided about 7,000 comments on the staff's draft environmental impact statement (EIS). "The number of comments significantly exceeded what had been planned for in the previous schedule," he said, in explaining the reason for a four-month delay in the final EIS. In separate letters to Exelon and SERI, Beckner said the effort devoted to responding to the draft EIS comments on the Dominion application would impact the review schedules for their applications.

Dudes said there appeared to be a similar number of "substantive" comments on the Exelon and SERI draft EIS reports. She said the final EIS for the Grand Gulf application would be pushed back four months to April—and now moved ahead of Exelon's Clinton application. The Clinton EIS was rescheduled for issuance in July 2006, a nine-month delay, she said.

Dominion and Exelon filed their applications on the same day, Sept. 25, 2003, and SERI followed about a month later, on Oct. 21, 2003. Dominion's application was put in the lead, with the review of the Exelon application next, and SERI's application as the final of the three. Now, SERI's application will be second and Exelon's application will be completed last.

Dudes said the staff had encountered another challenge unique to Exelon's application. Exelon's ESP referenced a seismic methodology that has not been previously reviewed by the staff.

While both Exelon and Dominion had originally submitted applications containing a new, performance-based methodology for seismic analyses, Dominion later revised its application to incorporate NRC-approved methodology after learning that the staff review would be slowed by a few months if it did not make the change. Exelon, however, stuck to the new methodology for determining the safe shutdown earthquake ground motion for the its site. Exelon spokesman Craig Nesbit said Aug. 17 that his company had not had time yet to determine the "practical effects" of the delay. But, he added, "We're disappointed NRC doesn't have the resources" for the review. Karl Neddenien, a Dominion spokesman, said his company understood the reason for the review schedule change. "Public participation is truly an important part of the process," he said. The review extension was not expected to have any significant impact on the company's plans, Neddenien said.

Dudes said the staff was using its experience on the first batch of ESP applications to prepare for other new licensing activities, particularly for combined construction permitoperating licensing (COL) applications, which might be filed starting around mid- to late 2007 and in 2008. "We've learned some lessons on how to resource and develop sufficient electronic tools," she said.

Budget considerations

Commissioner Jeffrey Merrifield cautioned the industry earlier this month that the agency's resources would play a significant factor in managing the work load of new plant licensing requests.

Merrifield said he worried there could be a "stacking up"—or backlog—of applications, particularly if licensees do not provide the NRC with enough notice of their intentions. And even with advance notice, NRC still might have to prioritize the work, he said.

"We will obviously be prepared to handle the few applications that we have been made aware of to date," Merrifield said in an Aug. 8 speech to the American Nuclear Society (ANS) conference in Amelia Island, Fla. "But beyond that, I think there is some uncertainty as to how the agency would handle any unexpected bow wave of 'surprise' applications for combined licenses, design certifications, or early site permits."

NRC had originally budgeted $37-million in fiscal 2006 for work on new reactor licensing, which included the three ESP applications, two design certification applications, and work on a technology-neutral regulatory framework for advanced reactor designs. But NRC officials told lawmakers in the spring that since it developed its budget, the demand for new reactor licensing projects has increased.

Congress responded to NRC's request, allocating another $20-million to support pre-application and other licensing work for COL applications expected to be filed starting in FY-08. Congress directed the agency to use the funds, which would be recouped through fees charged to licensees, to begin training new technical staffers to handle the expected load of three to five COL applications that could be submitted in the next couple years.
NRC Executive Director for Operations Luis Reyes, in a separate session at the ANS conference, said the agency might receive an application from Southern Co. in 2006 and another from Constellation in 2007. Several COL applications are expected in 2007—from Dominion, the NuStart consortium, and Duke. A chart included in his presentation showed that there might be a COL application from the Tennessee Valley Authority in 2007 and a second from NuStart in 2008.

In addition, design certification applications are expected for General Electric's Economic Simplified BWR, Areva's EPR, and PMBR Pty Ltd.'s Pebble Bed Modular Reactor. The agency also anticipates continuing licensing work on Atomic Energy of Canada Ltd.'s advanced Candu reactor design, and possibly Toshiba's 4S reactor and Westinghouse's International Reactor Innovative & Secure design.

Reyes told the conference that the agency would need to hire about 300 more technical staffers to prepare for the anticipated work. An NRC spokesman said last week that the staff increases would occur over fiscal years 2006-2009.
Adrian Heymer, director of plant performance improvement at the Nuclear Energy Institute (NEI), said the industry planned to discuss the cost issue for new reactor licensing, among other topics, at a chief nuclear officers' meeting at NEI on Aug. 18.

"NRC is trying to do the right thing—train up [staffers] to understand the [10 CFR] Part 52 process" for new plant licensing, Heymer said. "That way, when they get the applications, they are halfway down the runway, so to speak."
Heymer said some utilities have balked at having to cover costs associated with activities being pursued by a small group of companies. But to support the future of the industry, these expenses might have to be shared by all companies, he said.

Southern says Vogtle possible site for new reactor
Last week, Southern Nuclear Operating Co. told the

NRC it had selected Vogtle to evaluate for possible future reactors.

Southern Nuclear said it will file this summer either an application for an early site permit or information that would ultimately become part of a construction permitoperating license application.

Southern Nuclear, a Southern Co. subsidiary, emphasized that the plant's owners have not decided to build a new unit. Letting the NRC know of the company's plans will help ensure the agency has sufficient resources for the application review, it said. Also, Southern Nuclear said, selection of the Vogtle site doesn't preclude other sites within Southern Co.'s service area from being considered for future nuclear units.

Southern began seismic borings this week at Vogtle as part of its evaluation of that site. The drilling will check the site's integrity, including identifying any faults, said Southern spokesman Steve Higginbottom. The borings will also determine whether the site is stable enough to support containment and other structures for a new reactor, he said.

Vogtle is owned by Georgia Power, Oglethorpe Power Corp., the Municipal Electric Authority of Georgia, and the City of Dalton. It is operated by Southern Nuclear.

'First in' policy?

In his speech, Merrifield likened the possible influx of new reactor licensing applications to airport congestion and said NRC would have to play a role similar to that of air traffic controllers. "They know they have a limited number of gates with which to accommodate arriving and departing flights and a limited number of people who can arrive at the gates," Merrifield said.

"But they also know that sometimes there are far more planes trying to land than there are available gates and personnel to handle them," he said.

One solution, he suggested, would be to establish a "first in, first-out" policy, like the one it set up for license renewals. Also, the agency might have to limit the number of applications it could work on at any given time, he said. Merrifield also said he fully supported prioritizing work on reactor designs based on whether there was "licensee interest" rather than on designs that "vendors wish to certify in hopes of leveraging reactor orders."

Heymer said he believed the agency should be able to handle the work load, based on past history. He said the agency had received about five operating license requests per year in the 1970s and four per year in the 1980s. That meant there could be eight or nine license applications at any time.

"It would be somewhat sad to say we can't manage four or five COLs," he said. But, he said, "I think NRC should be able to handle the initial surge" of applications.
—Jenny Weil and Tom Harrison, Washington

NRC safety-culture plan prompts objections

NRC staff last week rolled out its preliminary plan for incorporating safety culture into the reactor oversight process (ROP), drawing both specific and general objections from participants at a meeting at the agency's headquarters in Rockville, Md.

Receiving stakeholder comment was a principal purpose of the Aug. 17 meeting, over half of which was devoted to questions and answers. But, adhering to the NRC commissioners' guidance in a staff requirements memorandum issued almost a year ago (INRC, 20 Sept. '04, 1), Michael Johnson, the director of NRC's Office of Enforcement and the chair of a committee overseeing the safety-culture project, said at the beginning of the session the question was "not whether but how" the agency would implement the revision to the ROP. Industry has objected that NRC guidance on safety culture would be too prescriptive and should not be issued at all (INRC, 13 Dec. '04, 7).

Both the agency presenters and members of the audience frequently invoked the NRC's regulatory principle of being "objective, risk-informed, understandable, and predictable." Johnson said the agency would "try to be true" to those principles, although "in some areas" doing so would be "challenging."

An overarching issue is how safety culture and safety conscious work environment (SCWE) can be integrated into the ROP, which is based on quantifiable "performance indicators." At the meeting, Anthony Pietrangelo, the senior director for risk regulation at the Nuclear Energy Institute, said the safety-culture effort was "questionable" because it departed from the ROP's principles of being performancebased objective, clear, and transparent. With 18 performance indicators and about 2,500 annual inspection hours for each licensed unit, the agency already has more than enough information, based on human and equipment performance, to draw conclusions about the safety focus of an operator, he said.

Earlier this month, NRC released the first version of documents that are to be the basis of the safety-culture effort. One of the documents is an "attributes table" listing and categorizing aspects of safety culture the agency intends to monitor. While some safety-culture features in the table list a quantifiable measure that helps track them—for example, "total number of NRC allegations" in a year is listed as a measure of the effectiveness of management actions and communication to promote a questioning attitude—others do not. For example, no quantitative measure is listed for "Site training program incorporates new and emerging issues."

Jeffrey Jacobson, a staffer in NRC's Office of Nuclear Reactor Regulation, said that while it may be feasible to look at some of the safety-culture features "objectively," the NRC "won't be able to run a PRA analysis"—a reference to probabilistic risk assessment—and "come up with a risk number." He added, "There will always be some qualitative element." But some of the quantifiable measures drew objections as well from industry representatives. Bill Mookhoek of STP Nuclear Operating Co. pointed to the NRC proposal to track the percentage of operators who fail their requalification examinations. In most cases, he said, the licensees' requirements go beyond NRC's, and so "licensees fail [the operators] where NRC would not have." Therefore, he said, a higher failure rate could in some cases be seen as evidence of a strengthened safety culture.

Julius Persensky of NRC's Office of Nuclear Regulatory Research, acknowledged that in some cases, "we don't know what direction is good." More broadly, the staff said in one of the documents explaining its approach, "Due to the uniqueness of the site specific programs, the difficulty that would exist in developing universal or site specific thresholds, and the potential for data manipulation, the Safety Culture Measures are not intended to be performance indicators analogous to those currently used in the ROP, but rather are meant to be data that could be used by inspectors as part of their inspections into associated areas, in order to highlight areas of potential concern for additional review." But Al Haeger of Exelon Generation Co. said many of the measures "don't have any standards against which they can be screened" and could turn the whole process into a "guessing game."

In the table developed by NRC, there are four "attributes" —safety conscious work environment; organizational learning and assessment; work planning and human performance; and organizational accountability—each of which encompasses three or four "elements." (For example, "continuous learning environment" is an element of "organizational learning and assessment.")

In the table, each element is matched with either of both of two additional types of information—qualitative "potential safety culture inspection information" and quantitative "potential safety culture measures." Jacobson said he and his colleagues were "pretty confident" they had "covered all the elements." The last two columns, on the tracking of those elements, are "more of a work in progress," he said. The table and other documents related to the meeting are in the "Safety Culture" section of NRC's Web site, http://www.nrc.gov/what-we-do/regulatory/enforcement/safety-culture.html.

The ROP currently has a cross-cutting issue for SCWE, which is considered one element of safety culture.

UCS 'skeptical'

Criticisms of NRC's general approach also were offered at the meeting by the Union of Concerned Scientists' David Lochbaum, who described himself afterward as "skeptical rather than opposed." He elaborated on his comments in a letter to Johnson the next day.

At the meeting, Johnson—making a point that other NRC officials have highlighted previously—said a major impetus for the increased emphasis on safety culture was the discovery of severe vessel head degradation and other problems at Davis-Besse. The problems occurred while the plant's performance indicators were green, indicating an absence of significant performance problems.

But NRC's proposed approach is the wrong response to Davis-Besse, Lochbaum said. Rather than trying to ferret out indications of defective safety culture, the NRC should determine why it failed to find the significant safety problems at Davis-Besse. As Lochbaum noted, the NRC eventually issued a number of findings at Davis-Besse, including a "red" and a "yellow," the two highest levels in NRC's four-level, colorcoded system for classifying safety significance.

There was "ample evidence" of trouble at Davis-Besse, but NRC inspectors failed to see it, he said at the meeting. The appropriate response for the NRC to this failing would be to provide "its existing army of inspectors with the means" to uncover such evidence, he said in the letter. Rather than making safety culture a component of the training for all inspectors, the NRC should have a group of safety culture experts, as it does in areas such as security and fire protection, Lochbaum said. When a performance indicator or inspection finding is "greater than green"—that is, not in the lowest of the four levels of safety significance— the NRC could assign one or more of the experts to determine if there is a safety culture problem, he said.

In response to Lochbaum's point, Johnson said the NRC might consider "a mix of the two" approaches—generalized safety-culture training for all inspectors and development of a specialized team of experts. Persensky said it was important to have the broad-based training, in part to give inspectors "a place to put the things that they see." But he also supported the idea of in-depth training for a smaller cadre of inspectors.

Isabelle Schoenfeld of the NRC's enforcement office said comments on the proposed table should be submitted by the end of this month. After that, there is to be a public meeting in October and a briefing of the Advisory Committee on Reactor Safeguards in Novernber. The inspection guidance on safety culture is to be finalized at the end of 2006 or beginning of 2007, according to the NRC schedule. There are a number of planned "external stakeholder interactions" along the way.

Jerry Roberts of Nebraska Public Power District suggested that the NRC staff involve stakeholders more directly, in part to avoid "unintended consequences."

The meeting attracted more than 20 non-NRC in-person participants, primarily from the industry, and there were more than a dozen people on the meeting's teleconference line. As Johnson noted, the West Coast telephone participants had to be awake early in the morning for a meeting that began at 8 a.m. in Maryland, which is three hours ahead of Pacific time.—*Daniel Horner, Washington*

NUCLEAR NEWS FLASHES - Friday, September 2, 2005

--ENTERGY's GRAND GULF AND RIVER BEND WERE UP TO 97% POWER THIS MORNING. The units had been operating at levels as low as 70%-75% power for the past few days because of fluctuations on the electricity grid caused by Monday's hurricane in the region. Waterford-3, also an Entergy unit, is not expected to return to service for several more days, NRC officials have said. Strong winds and water damage have destroyed or damaged major infrastructure in and around New Orleans, La. Waterford-3 is 20 miles west of the city.

NUCLEAR NEWS FLASHES - Wednesday, August 31, 2005

--ENTERGY AND NRC ARE SENDING IN RELIEF TO THE OPERATORS AND INSPECTORS at Waterford-3, which has been shut down since Sunday because of Hurricane Katrina. NRC Region IV spokesman Victor Dricks said additional people were sent to the site to provide a break to the resident inspectors and two additional regional staffers at the plant. Entergy spokesman Tim Crisler said the utility also gave "high priority" to getting relief workers on site. While there was no damage from the storm to the plant, voltage fluctuation is contributing to electrical grid instability. Waterford-3 has been running key safety systems from its emergency diesel generators, which yesterday received a shipment of additional fuel, Dricks said. Another NRC spokesman said today that the unit may not be able to restart until next week at the earliest, due to repairs needed to the grid and other nearby infrastructure.

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

August 18, 2005

RECEIVED
CRS/ACNW
AUG 23 2005

MEMORANDUM TO: Richard Barrett, Director, DET:RES
Charles Ader, Director, DRAA:RES
James Lyons, Director, DSSA:NRR
John Larkins, Executive Director, ACRS
Graham B. Wallis, Chairman, ACRS

FROM: Farouk Eltawila, Director, DSARE:RES

SUBJECT: COOPERATIVE SEVERE ACCIDENT RESEARCH PROGRAM (CSARP), MELCOR CODE ASSESSMENT PROGRAM (MCAP) TECHNICAL REVIEW MEETINGS, AND MELCOR WORKSHOP

This is to inform you and your staff of the upcoming Cooperative Severe Accident Research Program (CSARP) technical review meeting, and the MELCOR Code Assessment Program (MCAP) meeting to be held from September 20 through 23, 2005, at the Doubletree Hotel, Albuquerque, New Mexico. A preliminary agenda for the meeting is attached. As you may know, the Office of Research organizes the CSARP meeting each year. The meeting serves as an international forum for exchanging technical information and research findings in the field of severe accidents. The meeting is attended by delegates from seventeen countries who are CSARP members and by delegates from the national laboratories, academia, and other organizations who are engaged in severe accident research. This year, following CSARP and MCAP meetings, a 5-day MELCOR user workshop will be held from September 26 through September 30, 2005, at the same hotel. The agenda for this workshop can be found at SNL website: http://melcor.sandia.gov/. Dr. Ali Behbahani of my staff is the coordinator of the CSARP meeting. Please call Ali at 301-415-6768 or e-mail him at axb@nrc.gov if you have any questions about the meeting or would like to notify us of your interest in attending the meeting. Your participation at the meeting is valued by us.

Attachment: As stated

cc w/att.:
C. Paperiello, RES
J. Wiggins, RES

CSARP Meeting, September 20-21, 2005
MCAP Meeting, September 22-23, 2005
Doubletree Hotel, Albuquerque, New Mexico
(Limited Attendance)
Preliminary Agenda

Tuesday, September 20, 2005

8:00 am	Registration	
8:45 am	Opening Remarks	SNL/NRC

Technical Session 1 - OECD-Sponsored Severe Accident Research
Co-Chairs:A. Behbahani, NRC, and J. Bradlay, IRSN

9:00 am	Overview of the Progress in the OECD MASCA-2 Project	V. Strizhov, RAS
9:30 am	Progress of the OECD-MCCI Program	S. Basu, NRC
10:00 am	BREAK	
11:00 am	Results of the OECD SERENA Phase 1 programme on FCI	D. Magallon, CEA

Technical Session 2 - Other Cooperative Research Activities
Richard Lee, NRC, and W. Frid, SKI

11:30 pm	IRSN R&D on severe accidents (goals , programs...)	J. BARDELAY, IRSN
12:00 pm	Late In-Vessel Phase and Ex-Vessel Melt Behavior Experiments at FZK"	A. Miassoedov, FZK
12:30 noon	LUNCH	
2:00 pm	Severe Accident Research in Sweden: The APRI project	W. Frid, SKI
2:30 pm	Presentation on the source term program	B. CLEMENT, IRSN
3:00 pm	Preliminary results and interpretation of PHEBUS FPT3 test by Bernard CLEMENT	B. CLEMENT, IRSN
3:30 pm	BREAK	
4:00 pm	Degraded core reflood: present knowledge based on experimental and Analytical Data	W. Hering, C. Homann,FZK

4:30 pm	Overview of NRI Activities in Severe Accident Research	J. Dienstbier, et,al., NRI
5:00 pm	Severe Accident Research at the RIT Sweden Status and Plan	N. Dinh, RIT
5:30 pm	ADJOURN	

Wedensday, September 21, 2005

Technical Session 2 - Other Cooperative Research Activities (Cont'd)
Co-Chairs: S. Basu, NRC, and W. Scholtyssek, FzK

8:30 am	Progress of the ARTIST projects	D. Powers, SNL
9:00 am	CFD Analysis of Aerosol Retention in Steam Separator and Dryer in the ARTIST Program	M. Ogino, JNES
9:30 am	Low pressure Melt Ejection Experiments in the DISCO Facility	L. Meyer, FZK
10:00 am	BREAK	
10:30 am	Partners' Meeting (open to Partners' only)	
12:00 noon	LUNCH	

Technical Session 3 - Severe Accident Codes: Development and Assessment
Co-Chairs: I. Madni, NRC, and M. Sonnenkalb, GRS

1:00 pm	Overview of MELCOR Development Activities	R. Gauntt, et al., SNL
1:30 pm	MELCOR 1.8.6, Lower head Modeling	L. Humpheries, SNL
2:00 pm	MELCOR Modernization Project: Current Status	V. Belikov, V. Strizhov, RAS
2:30 pm	Overview of MELCOR Thermal Hydraulic Modeling for Containment Analysis	J. Tills, Jack Tills & Assoc., Inc.
3:00pm	BREAK	
3:30 pm	Analysis of TMI-2 with MELCOR1.8.5 and supportive analysis using SCDAP	Haste, et. al, PSI, B. Jaeckel
4:00 pm	Status and evolution of ASTEC severe accident code	B. CLEMENT, IRSN
4:30 pm	Application of MELCOR 1.8.5 at Nuclear Regulatory Authority of the Slovak Republic.	S Stubnova, NRA SA
5:00 pm	MELCOR CODE ASSESSMENT BY SIMULATION OF TMI-2 PHASES 1 AND 2	C. Burns, K. Vierow, Purdue Univ.

5:30 pm	ADJOURN	

Thursday, September 22, 2005

Technical Session 3 - Severe Accident Codes: Development and Assessment (Cont'd)

Co-Chairs: I. Madni, NRC, and M. Sonnenkalb, GRS

8:30	MELCOR Large Break LOCA Analyses in Support of Risk Informing 10 CFR 50.46	I. Madni, NRC
9:00 am	MELCOR Activities at JNES	M. Ogino, JNES
9:30 am	MODELING OF SEVERE ACCIDENTS IN THE ADVANCED CANDU REACTOR USING MELCOR	M. Zavisca, ERI
10:00am	ADJOURN	

MELCOR Cooperative Assessment Program (MCAP) Meeting
September, 22-23, 2005

Thursday, September 22, 2005

10:30 am	Registration	
11:00 am	Introduction and Opening Remarks	NRC/ SNL

Technical Session 1

11:30 am	Current MELCOR Development Activities	R. Gauntt, et al., SNL
12:15 am	Post-test Analysis of ISP42 (PANDA Tests) using the MELCOR Code	J. Tills, Jack Tills & Assoc., Inc.
1:00 am	Lunch	
2:00pm	Verification of MELCOR 1.8.6 against RASPLAV tests (or Sandia LHF tests).	V.Nosatov, RAS
2:45pm	Summary of MELCOR applications to German NPP	M. Sonnenkalb, GRS
3:30 pm	Two Examples of MELCOR Code Application to Solution of Severe Accident Topics	J. Duspiva, et. al., NRI
4:15 pm	BREAK	
4:45 pm	H2 SEQUENCE IN A FRENCH PWR 900: COMPARISON BETWEEN MELCOR AND ASTEC codes	F. De Rosa, ENEA
5:30 pm	ADJOURN	

Friday, September 23, 2005

Technical Session 2

8:30 am	MELCOR Application to Beyond Design Basis Large Break LOCA Analyses for a Westinghouse Three-Loop Plant	I. Madni, NRC
9:15 am	Best Practices in Modeling Severe Accident Progression in Nuclear Power Plants	SNL

10:15 am	Users' Forum
11:00 am	ADJOURN

Dominion

Millstone Units 2 and 3

License Renewal Presentation to ACRS

September 8, 2005

Bill Watson
MPS LR Supervisor
Dominion Nuclear Connecticut

2

Dominion

Participants

- Paul Aitken - Innsbrook LR Supervisor
- Support Staff
  - Marc Hotchkiss
  - Charlie Sorrell
  - Gary Komosky
  - Tom Hendy

Dominion

Introduction

- Description of MPS-2 and MPS-3
- Plant Performance & Operating History
- License Renewal Application
- Corrective Action Process
- LR Commitments
- License Renewal Implementation

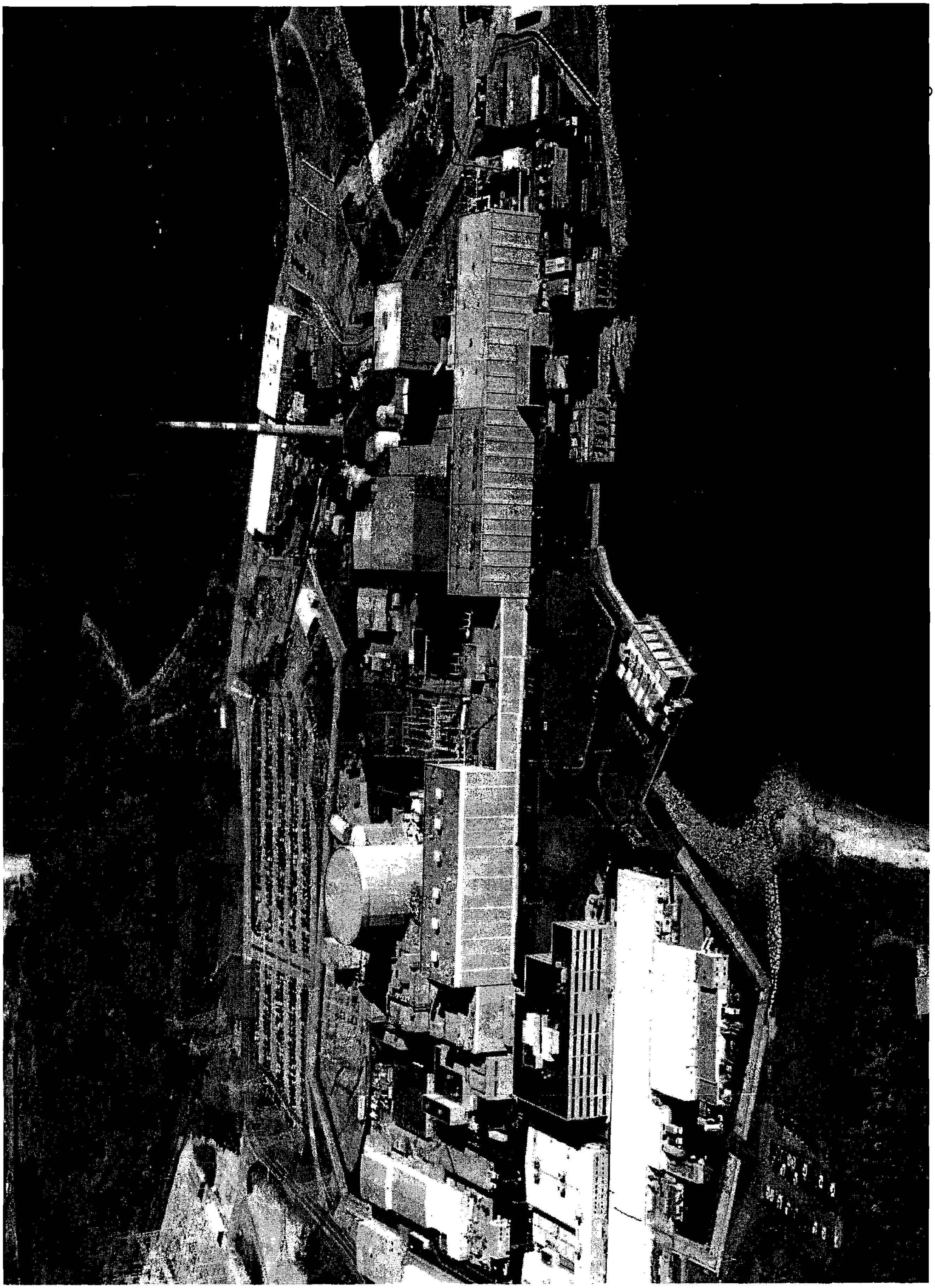
Description of Millstone Unit 2

- NSSS Supplier - Combustion Engineering, Inc.
  - 2-Loop design (2 hot legs and 4 cold legs)
  - 2 Recirculating Steam Generators (S/Gs)
  - 4 RCPs
- Architect/Engineer - Bechtel Corp.
- Initial Ops: 1975
- Electrical capacity: 895 MWe

Dominion

Description of Millstone Unit 3

- NSSS Supplier - Westinghouse Corp.
  - 4-Loop design
  - 4 Recirculating S/Gs
  - 4 RCPs
- Architect/Engineer - Stone and Webster Engineering
- Initial Ops: 1985
- Electrical capacity: 1195 MWe



Dominion

Plant Performance - Five Years

MILLSTONE UNIT 2				
	Cycle 14 6/00-2/02	Cycle 15 4/02-11/03	Cycle 16 11/03-4/05	Cycle 17 5/05-
Cycle Capacity Factor	95.6%	92.4%	98.0%	98.2% as of 8/22/2005
Corresponding Outage Duration	45.7 days	51.0 days	39.4 days	

MILLSTONE UNIT 3				
	Cycle 7 6/99-2/01	Cycle 8 3/01-9/02	Cycle 9 10/02-4/04	Cycle 10 5/04-
Cycle Capacity Factor	98.7%	97.3%	97.0%	96.1% as of 8/22/2005
Corresponding Outage Duration	56.2 days	30.6 days	36.8 days	

Dominion

Millstone Unit 2 Operating History

- Unit 2 - Operating for 115 days since last refueling outage.
- Lower portions of the two S/Gs were replaced with corrosion resistant material (including tubes and tubesheet).
- RV Head replaced in the Spring 2005 RFO.
- Pressurizer is scheduled to be replaced in the 2006 RFO (Fall).
- Unit 2 does not have any bottom mounted instrumentation (BMI).

Dominion

Millstone Unit 3 Operating History

- Unit 3 - Operating 132 days since last unit shutdown (automatic reactor trip).
- RV Head not currently scheduled for replacement.
  - RV Head susceptibility ranking is in the lowest susceptibility category.
  - During 2002 RFO, the RV head visual inspection identified that there was no evidence of material degradation or RCS leakage.
- During 2004 RFO, the BMI visual inspection identified that there was no evidence of material degradation or RCS leakage.

Dominion

Millstone Unit 1

- Unit 1 is permanently defueled.
- Unit 1 SSCs were evaluated for affect on Units 2 & 3.
- Certain Unit 1 Structures were included in LR scope:
  - Turbine Building
  - Control Room/Radwaste Treatment Building
- Appropriate Unit 1 FP equipment was reassigned as Unit 3 equipment when Unit 1 was defueled and has been included in scope.
- Unit 1 is in a safe store condition until the site is decommissioned.

License Renewal Application

- Original License Expiration
  - Unit 2 - July 31, 2015
  - Unit 3 - November 25, 2025
- Application Submitted - January 22, 2004
- LRA process
  - Standard LRA Format
  - Extensive use of past precedence
  - Participated (post-pilot plants) in the Consistent with GALL Audits

Dominion

Corrective Action Process

- Establishes the measures to be taken to assure that conditions adverse to quality are promptly corrected.
- Establishes measures to provide reasonable assurance that:
  - The cause of the condition is determined
  - Corrective action preclude repetition
  - Corrective action is taken in a timely and accurate manner

Corrective Action Process

- NRC Inspection (2004) of activities related to problem identification and resolution concluded generally, problems were properly identified, evaluated, and corrected.
- A recent Nuclear Oversite audit of the Corrective Action Program and Independent Review Activities concluded regulatory requirements are being met.

Dominion

LR Commitments

- Proposed commitments were submitted in the LRA and modified as needed during NRC review.
  - 37 Commitments for Unit 2
  - 37 Commitments for Unit 3
- New Chapter in each Unit's FSAR will identify Commitments.
- These will be treated as obligations under the operating license, requiring NRC approval to change (except to status as "complete").

Dominion

License Renewal Implementation

- License Renewal implementation has begun.
  - Training is being provided to all affected departments
  - A License Renewal Program Owner has been assigned
  - Procedures are being marked up and changes are being processed
  - License Renewal implementation impact assessment is being conducted to support schedule and budget

Dominion

License Renewal Implementation

- Individual tasks for each commitment will be loaded into the Action Item Tracking and Trending System.
- Commitments will be implemented prior to the period of extended operation or sooner.
- The FSAR will be updated upon satisfactory completion of a license renewal commitment.

Dominion

MPS License Renewal

Questions?

Millstone Power Station Units 2 and 3 License Renewal Safety Evaluation Report

Staff Presentation to the ACRS
Johnny Eads, Sr. Project Manager
Office of Nuclear Reactor Regulation
September 8, 2005

3

Overview

- Two License Renewal Applications submitted by letter dated January 20, 2004
- Unit 2 OL expires July 31, 2015 and Unit 3 OL expires on November 25, 2025
- Unit 2 - Combustion Engineering design with two steam generators and four coolant loops
- Unit 3 – Westinghouse design with four steam generators and four coolant loops

NRC Review Process

- Scoping and Screening Methodology Audit
- Consistency with GALL Audits
  - AMPs
  - AMRs
- Regional inspections
  - Scoping and Screening Inspection
  - AMP Inspection

NRC Review Process (continued)

- AMP GALL Audit
  - March 29 – April 1, 2004
- Scoping and Screening Methodology Audit
  - May 3 – 7, 2004
- AMR GALL Audit
  - May 3 - 13, 2004
  - June 7- 10, 2004
- AMP/AMR Audit Exit Meeting
  - July 13, 2004
- Regional Scoping and Screening Inspection
  - July 26 – 30, 2004
- Regional AMP Inspection
  - September 13 – 17, 2004 and September 27 – October 1, 2004

SER Overview

- SER with Open Items issued on February 24, 2005
    - 6 Open Items
    - 6 Confirmatory Items
    - 3 License Conditions
- SER issued August 1, 2005 with all Open and Confirmatory Items closed

SER Open Items

- Open Item 2.1.3-1 related to NSR criteria pursuant to 10 CFR 54.4 (a)(2)
- Open Item 3.1.2-6, Scoping of the Rx vessel flange leak detection line
- Open Item 3.0.3.2.18-1, Bolting loss of preload for non-class 1 bolting
- Open Item 3.0.3.2.18-2, Bolting Integrity AMP references to EPRI Good Bolting Practices
- Open Item 4.7.3-1(a), Unit 2 Reactor Coolant Pump Code Case N-481
- Open Item 4.7.4-1, Leak-Before-Break Analysis

Fire Protection Issue

- Original GALL Exception:
  - No aging effects requiring management for halon and carbon dioxide systems
- Resolution:
  - Based on ACRS Subcommittee comments and follow-up staff review, exception withdrawn
  - Applicant committed to aging management of Halon and CO2 systems per GALL

Millstone Unit 2
2Q/2005 Performance Summary

Reactor Safety

Radiation Safety

Safeguards

Initiating Events → *Mitigating Systems* → *Barrier Integrity*

Emergency Preparedness

Occupational Radiation Safety

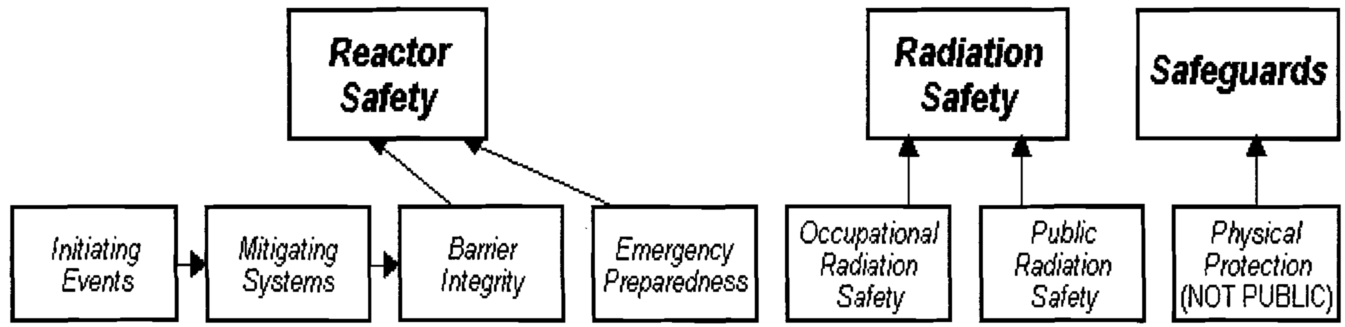
Public Radiation Safety

Physical Protection (NOT PUBLIC)

Performance Indicators

Initiating Events	Mitigating Systems	Barrier Integrity	Emergency Preparedness	Occupational Radiation Safety	Public Radiation Safety
Unplanned Scrams (G)	Emergency AC Power System Unavailability (G)	Reactor Coolant System Activity (G)	Drill/Exercise Performance (G)	Occupational Exposure Control Effectiveness (G)	RETS/ODCM Radiological Effluent (G)
Scrams With Loss of Normal Heat Removal (G)	High Pressure Injection System Unavailability (G)	Reactor Coolant System Leakage (G)	ERO Drill Participation (G)		
Unplanned Power Changes (G)	Heat Removal System Unavailability (G)		Alert and Notification System (G)		
	Residual Heat Removal System Unavailability (G)				
	Safety System Functional Failures (G)				

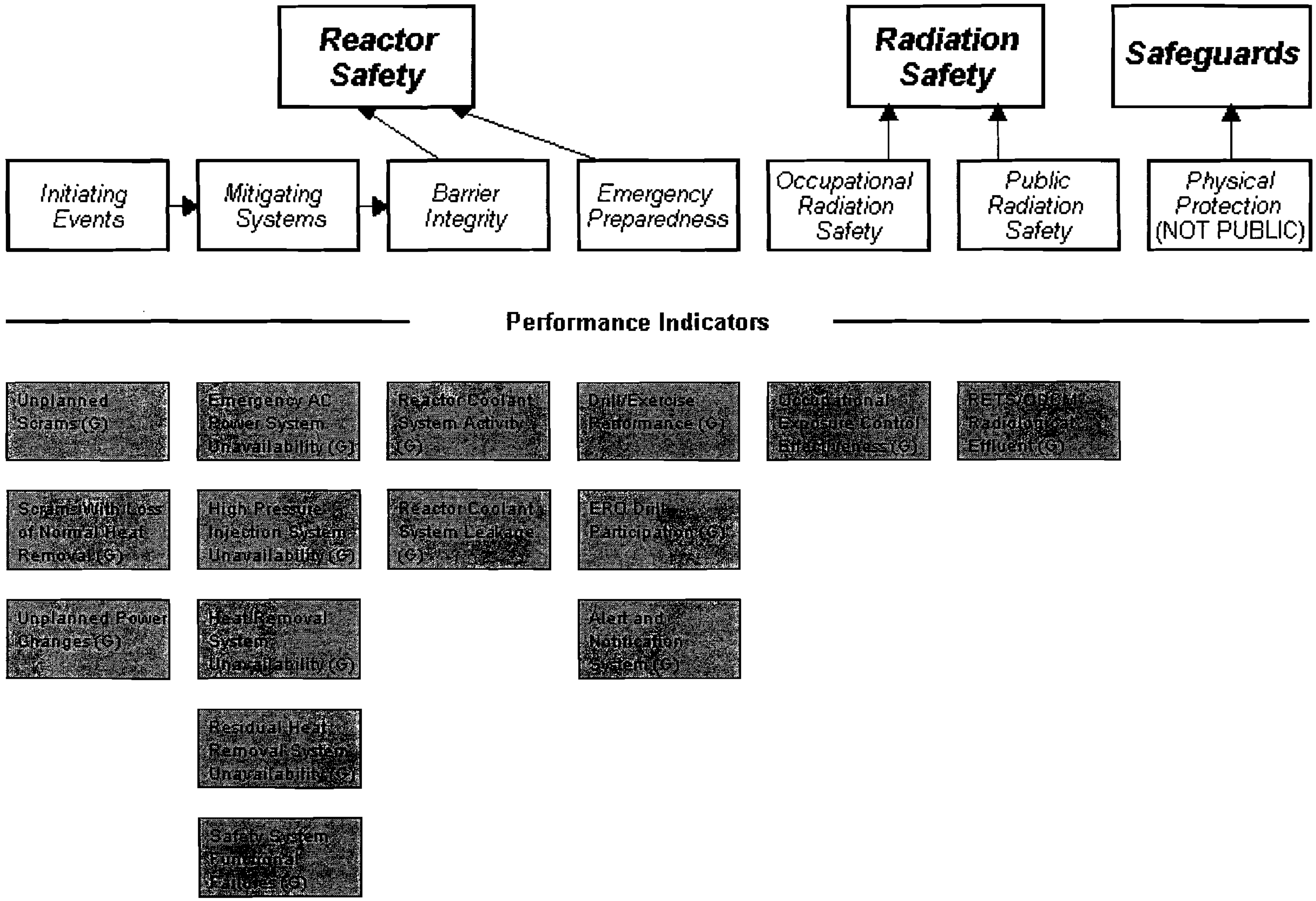
Millstone Unit 2
2Q/2005 Performance Summary



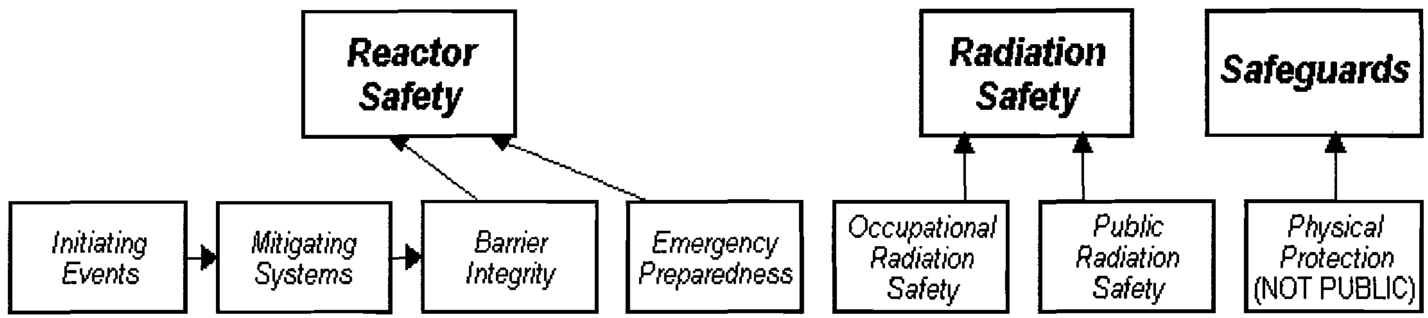
Most Significant Inspection Findings

2Q/2005	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter
1Q/2005	G	G	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter
4Q/2004	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	G	No findings this quarter
3Q/2004	G	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter

Millstone Unit 3
2Q/2005 Performance Summary



Millstone Unit 3
2Q/2005 Performance Summary



Most Significant Inspection Findings

	Initiating Events	Mitigating Systems	Barrier Integrity	Emergency Preparedness	Occupational Radiation Safety	Public Radiation Safety
2Q/2005	G	G	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter
1Q/2005	No findings this quarter	G	G	No findings this quarter	No findings this quarter	No findings this quarter
4Q/2004	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter
3Q/2004	No findings this quarter	G	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter

Conclusions

- The staff has concluded that there is reasonable assurance that the activities authorized by the renewed licenses will continue to be conducted in accordance with CLB, and that any changes made to the MPS CLB in order to comply with 10 CFR 54.29(a) are in accord with the Act and Commission's regulations.

(4)

Millstone 2
2Q/2005 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2005
Identified By: NRC
Item Type: FIN Finding
FAILURE TO ADEQUATELY ADDRESS CONCERNS RELATED TO FREEZE PROTECTION OF THE OUTDOOR TEMPORARY AIR COMPRESSOR
The inspectors identified a self-revealing finding for the failure to adequately address issues related to the operation of an outdoor temporary air compressor and associated air dryer skid during cold weather conditions. On November 11, 2004, Dominion had identified that additional freeze protection actions were required to ensure the availability of the compressor during cold weather. Subsequently, the inspectors identified two occasions where actions taken to ensure availability of the compressor were not adequate. On December 17, 2004, the inspectors identified that a heat trace for the system dryer was deenergized. On February 1, 2005, the temporary air compressor failed causing the "B" instrument air compressor to start. Following the air transient, Dominion conducted an investigation and concluded that the cause of the temporary air compressor failure was freezing of the pre-filter on the air dryer skid. Dominion replaced the compressor, installed a tent around the air-dryer towers, and placed a heating unit inside the tent. The finding was more than minor because it affected the equipment performance attribute of the Initiating Events cornerstone objective of limiting the likelihood of events that upset plant stability at power. The performance issue associated with this finding was the failure to take adequate actions to ensure that adverse weather conditions did not affect the availability of the temporary instrument air system. The risk of this finding was determined to be of very low safety significance (Green), because, although the temporary air compressor system became unavailable, the standby instrument air compressor restored instrument air system pressure. The instrument air system pressure stabilized and recovered such that the instrument air header pressure did not cause a reactor trip. This finding was related to the cross-cutting area of Problem Identification and Resolution in that Dominion failed to take adequate corrective actions to prevent the air dryer skid from freezing.
Inspection Report# : 2005002*(pdf)*

Significance: G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO PROPERLY ESTABLISH AND IMPLEMENT 10 CFR 50, APPENDIX B, CRITERION XVI, TO ADDRESS REPEATED LIFTING OF MAIN STEAM CODE SAFETY VALVES
The inspectors identified a non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, for the failure to take effective corrective actions to preclude main steam code safety valves from lifting following design basis turbine trips/reactor trips from 100% power. Following two uncomplicated reactor trips at Unit 2 in March 2004, the inspectors noted that main steam code safety valves lifted and reseated. The inspectors determined that Unit 2 had a history of main steam code safety valves lifting and reseating following uncomplicated trips. The inspectors concluded that cycling main steam code safety valves following trips from full power increases the likelihood that they may not reseat. Dominion had not taken effective corrective actions to correct this longstanding issue. Dominion has undertaken a study (to complete by the end of 2004) to evaluate this system condition and to specify long term design changes which will be scheduled for completion in refueling outage 2R17 (fall of 2006). Dominion has entered this issue into their corrective action program. This issue is more than minor because it affects the equipment performance attribute of the Initiating Events Cornerstone and the objective to limit the likelihood of those events that upset plant stability. Cycling of main steam code safety valves results in a greater likelihood that the valves will not reseat properly during an event. The finding was determined to have a very low safety significance since it did not

contribute to the likelihood of a primary loss of coolant accident, did not contribute to both the likelihood of a reactor trip and the unavailability of mitigating equipment, and did not increase the likelihood of a fire or internal/external flood. This finding is related to the cross-cutting area of Problem Identification and Resolution.
Inspection Report# : 2004007*(pdf)*

Significance: Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
I&C TECHNICIANS AND OPERATIONS PERSONNEL DID NOT VERIFY ALL APPROPRIATE PREREQUISITES OR PERFORM ALL APPLICABLE PROCEDURAL STEPS WHICH THEN RESULTED IN THE ADVERTENT ACTUATION OF A SAFETY-RELATED SYST
The inspectors identified a non-cited violation of Technical Specification (TS) 6.8.1, for the failure to adequately implement post-maintenance testing following replacement of a pressurizer level instrument. On July 28, 2004, Operations and Maintenance personnel failed to meet a "Unit 2 Shutdown" procedural prerequisite and did not perform a procedure step to place charging pump controls in pull-to-lock during post-maintenance testing of pressurizer level control circuitry. As a result, both standby charging pumps started with one charging pump already operating. Dominion has specified training for both Operations and Maintenance organizations describing the circumstances of this event and management expectations for work evolution briefs, peer checking, and actions to be taken for unexpected conditions. Additionally, Maintenance management reinforced work practice expectations for the use of "N/A" in procedures and work planning process improvements. Dominion has entered this issue into their corrective action program. This issue is more than minor because it is associated with the human performance attribute of the Initiating Events Cornerstone and the objective to limit the likelihood of those events that upset plant stability. The start of both standby charging pumps with one charging pump already operating was the precursor to the failure of the charging system on March 7, 2003. The finding was determined to have a very low safety significance since it did not contribute to the likelihood of a primary loss of coolant accident, did not contribute to both the likelihood of a reactor trip and the unavailability of mitigating equipment, and did not increase the likelihood of a fire or internal/external flood. This finding is related to the cross-cutting area of Human Performance.
Inspection Report# : 2004007*(pdf)*

Mitigating Systems

Significance: Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO IMPLEMENT PROCEDURES TO CORRECTLY INSTALL TEMPORARY COOLING TO THE EAST 480 VOLT SWITCHGEAR
The inspectors identified a self-revealing non-cited violation of Technical Specification 6.8.1a, "Procedures and Programs," for the failure to adequately implement the procedure for installing temporary ventilation through the East 480 volt vital switchgear room when normal cooling was disabled for maintenance. The procedure establishes the required flow path in the switchgear room when compensatory cooling measures were required. On January 12, 2005, operators failed to perform the procedure step that opens doors to provide for an exhaust path to allow warm air to leave the switchgear room. The finding was greater than minor because the failure to install the compensatory cooling system, per the procedure, caused the air flow through the East 480 volt switchgear room to be below the minimum required to support cooling of the 480 volt system for initiating events (transients), mitigating systems, and barrier integrity systems. The finding was associated with the equipment performance attribute of the initiating events and mitigating systems cornerstones, and the containment structures, systems, and components and barrier performance attribute of the barrier integrity cornerstone. Since more than one cornerstone was affected, a Reactor Safety Significance Determination Process Phase 2 analysis was performed. The analysis resulted in a finding of very low safety significance (Green) because the improper installation of the compensatory measures did not result in an actual loss of the supported 480 volt AC system or electro hydraulic control functions. This finding was related to the cross-cutting area of Human Performance in that both Engineering and Operations personnel failed to correctly implement the procedure for compensatory cooling.
Inspection Report# : 2005002*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Significance: Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
HIGH CONCENTRATION OF AIRBORNE RADIOACTIVE MATERIAL DURING FILTER TRANSFERS
Dominion did not use process or other engineering controls, to the extent practical, to control the concentration of radioactive material in air during handling of radioactive spent Unit 2 filters on September 29, 2004. As a result, elevated concentrations of radioactive material in air was generated and two workers sustained unplanned intakes of airborne radioactive material. This was a self-revealing, non-cited violation of 10 CFR 20.1701, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas, Use of Process or Other Engineering Controls." The finding was greater than minor, in that it was associated with the program and processes for exposure control and monitoring attribute of the Radiation Safety Cornerstone attributes and did affect the objective of the Cornerstone. The finding was determined to be of very low risk significance (Green) using NRC Manual Chapter 0609, Appendix C, in that it involved an ALARA exposure control finding, but the three year rolling average collective occupational dose for Millstone did not exceed 135 person-rem. Dominion suspended the work activity and initiated a root cause investigation. This finding was related to the cross-cutting area of Human Performance in that Dominion did not use process or engineering controls, to the extent practical, resulting in exposure of two workers to elevated concentrations of airborne radioactive material..
Inspection Report# : 2004008*(pdf)*

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Last modified : August 24, 2005

Millstone 3
2Q/2005 Plant Inspection Findings

Initiating Events

Significance: Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO EVALUATE EXCEEDING SPECIFIED FIRE LOADING LIMIT FOR MAIN STEAM VALVE ENCLOSURE
The inspectors identified a non-cited violation of License Condition 2.H to Facility Operating License NPF-49 for the failure to properly evaluate transient combustible fire loading for the Main Steam Valve Enclosure Building (Fire Area, MSV-1) from April 1999 to July 2005. Specifically, Dominion did not accurately account for the amount of transient combustibles present in the area which caused the licensee to unknowingly, and without evaluation, exceed the fire severity classification threshold for this area. The inspectors determined that the failure to properly evaluate the transient combustibles for the fire area MSV-1 was more than minor based on a similar example described in Manual Chapter 0612, "Power Reactor Inspection Reports", Appendix E, "Examples of Minor Issues", Section 4k. Specifically, the fire loading exceeded the fire hazard analysis and was not properly evaluated. This finding is associated with the initiating event cornerstone and involves the fire initiator attribute of cornerstone. The safety significance of the finding was determined to be low based on the plywood being fire retardant and the increase in the fire loading remained significantly less than the maximum allowed by the higher severity classification of "low". This finding is related to the cross-cutting area of Problem Identification and Resolution in that neither the monthly inspection of the fire areas and permits nor the annual review of temporary fire permits identified the issue despite the condition having existed for approximately six years.
Inspection Report# : 2005003*(pdf)*

Significance: May 18, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
LESS THAN ADEQUATE CORRECTIVE ACTIONS FOR POTENTIAL RCS PRESSURE BOUNDARD DEGRADATION DUE TO BORIC ACID CORROSION
The inspectors identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action" in that DNC's did not promptly identify and correct a condition adverse to quality involving boric acid leaks in containment. The finding was more than minor because it affected the Initiating Events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations; if left uncorrected it could become a more significant concern, such as excessive leakage or the loss of RCS integrity. In addition, this performance deficiency is related to the cross-cutting area of problem identification and resolution in two respects. First, after approximately six days and several containment entries, DNC had not identified the presence of 12 additional boric acid leaks. Second, although aware of the leak on a loop drain isolation valve, DNC did not re-evaluate or resolve the leakage impact on adjacent safety-related SSCs until questioned by the inspectors. This finding was determined to be Green (very low safety significance) based on IMC 0609, Appendix A, Phase 1 SDP worksheet for at-power situations. The leakage is characterized as a LOCA initiator, but assuming worst case degradation, the leakage would not have resulted in exceeding a TS limit for identified RCS leakage or have adversely impacted other mitigating systems.
Inspection Report# : 2005012*(pdf)*

Mitigating Systems

S[redacted]ance: May 18, 2005
I[redacted]d By: NRC
Item Type: FIN Finding
IMPROPER EVENT DIAGNOSIS LED TO E-PLAN DECLARATION
The inspectors identified a Green finding because procedure MP-14-MMM, Revision 006-01, "Operations" was not adequately implemented. The team identified problems with crew diagnosis and communications during the event which led to an emergency plan declaration when actual conditions for that declaration did not exist. This NRC-identified finding is considered to be of more than minor safety significance because if left uncorrected, ineffective monitoring and diagnosis of plant conditions during significant plant events could lead to a more significant safety concern. In addition, this performance deficiency is related to the cross cutting area of human performance in that, during the actual event, the operating crew did not diagnose that the MSSVs were functioning as designed and crew briefings did not provide a complete perspective of known plant conditions. This finding was not suitable for the an NRC SDP evaluation, but was reviewed by NRC management in accordance with IMC 0612, Section 05.04c and determined to be of very low safety significance (Green).
Inspection Report# : 2005012*(pdf)*

Significance: May 18, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO IMPLEMENT APPROPRIATE PMS ON THE TDAFW PUMP CONTROL VALVE
The inspectors identified a Green non-cited violation of TS 6.8.1 regarding the deletion an 18-month control valve PM for TDAFW pump in August 2000 without performing a thorough change evaluation per CBM 105, Revision 004-03, Preventive Maintenance Program. This performance deficiency was a primary contributor to the TDAFW pump overspeed trip. This NRC-identified finding was of more than minor safety significance because it affected the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, because the PM was not completed, the reliability of the TDAFW pump was adversely affected. In ev[redacted]ting this finding, the Significance Determination Process (SDP) (Phase 1) screening identified that a SDP workbook ([redacted]2) evaluation was needed because the TDAFW pump was potentially inoperable in excess of its TS Allowed Outage T[redacted] three days. Since the Phase 2 evaluation exceeded a risk threshold, an NRC Region I Senior Reactor Analyst (SRA) conducted a Phase 3 evaluation to more accurately account for the exposure time and to appropriately credit operator actions to recover the TDAFW pump after it automatically tripped on April 17. The Phase 3 evaluation determined that this finding represented a change in core damage probability of low to mid E-7, which is of very low risk significance (Green).
Inspection Report# : 2005012*(pdf)*

Significance: May 18, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
EOP E-0 STEP NOT PERFORMED AS REQUIRED
The inspectors identified a Green non-cited violation of Technical Specification (TS) 6.8.1 because the operating crew did not take control of reactor coolant system (RCS) temperature in accordance with Step 21 of Emergency Operating Procedure (EOP), E-0, "Reactor Trip or Safety Injection". Consequently, the main steam safety valves (MSSVs) automatically operated to control RCS temperature for approximately 30 minutes longer than was necessary. This NRC-identified finding is considered to be of more than minor significance because it adversely impacts the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the unnecessary cycling of the MSSVs increased the chance that a previously cycled MSSV would not open or would fail to reseat following an additional opening. The finding was determined to be Green (very low safety significance) in accordance with IMC 0609, Appendix A, Phase 1 SDP worksheet for at-power situations.
Inspection Report# : 2005012*(pdf)*

Significance: May 18, 2005
Id[redacted]d By: NRC
It[redacted]e: NCV NonCited Violation
SI[redacted]ATOR RESPONSE DID NOT ADEQUATELY MODEL MSSV RESPONSE
The inspectors identified a Green non-cited violation for failure of the Millstone Unit 3 simulator to correctly model main steam

safety valve operation as required by 10 CFR 55.46(c)(1), "Plant-Referenced Simulators." This NRC- identified finding is more than minor because it affected the human performance attribute of the mitigating systems cornerstone. This finding was evaluated using the Operator Requalification Human Performance SDP (IMC 0609 Appendix I) because it is a requalification training issue related to simulator fidelity. The SDP, Appendix I, Block 12, requires the inspector to determine if deviations between the plant and simulator could result in negative training or could have a negative impact on operator actions. "Negative Training" is defined, in a later version of the standard (ANSI 3.5-1993), as "training on a simulator whose configuration or performance leads the operator to incorrect response or understanding of the reference unit." During the event of April 17, 2005, operators were influenced by negative training on the simulator to erroneously believe that a safety valve in the plant was stuck open when it was actually still functioning as designed.
Inspection Report# : 2005012*(pdf)*

Significance: May 18, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FALSE OR MISLEADING CONTROL ROOM INDICATIONS
The inspectors identified a Green non-cited violation in that DNC did not comply with 10 CFR 50, Appendix B, Criterion III, "Design Control," regarding the suitability of a control room indicator in providing information needed by operators to ensure appropriate decision making while implementing emergency operating procedures. This violation is related to the misleading control room indication for Charging/Safety Injection (CHG/SI) flow indication which led operators to take improper actions in EOP E-0, "Reactor Trip or Safety Injection" because the flow indicator (3SIH-FI917), despite the existence of adequate injection flow to the core, indicated zero gallons per minute (GPM) flow. This self-revealing finding was of more than minor safety significance because it was associated with the design control attribute of the mitigating systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be Green (very low safety significance) based upon IMC 0609, Appendix A, Phase 1 SDP worksheet for at-power situations. The inspectors determined that the finding represented a design deficiency that did not result in a loss function per Generic Letter (GL) 91-18, Revision 1.
Inspection Report# : 2005012*(pdf)*

Significance: Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO PROMPTLY EVALUATE AND CORRECT A DEGRADED CONDITION ASSOCIATED WITH THE DIVIDER PLATE FOR ALL THREE RPCCW HXS
The inspector identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for Dominion's failure to take prompt and appropriate corrective actions to address a condition adverse to quality. Specifically, Dominion did not promptly evaluate and correct a degraded condition associated with the divider plate for all three reactor plant component cooling water (RPCCW) heat exchangers (HXs). The inspector determined that this issue was more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone, and it potentially affected the objective to ensure the availability and reliability of the RPCCW HXs. The finding was of very low safety significance (Green), because the finding was a qualification deficiency confirmed not to result in loss of a function. The issue was similarly of very low risk in the Initiating Events cornerstone because the finding did not increase the likelihood of a reactor trip or a loss of service water (SW) event. The finding was associated with the cross-cutting area of problem identification and resolution (PI&R) in that Dominion's inadequate evaluation and untimely corrective actions for a degraded condition potentially affected the RPCCW HXs.
Inspection Report# : 2005002*(pdf)*

Significance: Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO ADEQUATELY IMPLEMENT TESTING PROCEDURES FOR RESTORING THE "a" EDG TO SERVICE
The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," for the failure to adequately implement post-maintenance test (PMT) procedures for restoring the "A" emergency diesel generator (EDG) to service following maintenance of the neutral breaker. On March 1, 2005, Dominion conducted maintenance and doble testing of the "A" EDG neutral breaker. The Maintenance Department turned the breaker over to Operations for final post-maintenance testing and restoration. After racking in the breaker, Operations noted that the red light on the front of the EDG neutral breaker panel did not light as expected. Contrary to the PMT acceptance criteria, Operations assessed that the PMT was satisfactorily completed and exited the EDG technical specification. The oncoming shift investigated and determined the

red light was not lit because there was a problem with the neutral breaker trip circuit. Operations declared the EDG inoperable and re-entered the EDG technical specification. This issue was more than minor because it was associated with the reliability of the "A" EDG. The inspectors determined that the finding was of very low safety significance (Green) because it did not involve a design or qualification deficiency, represent an actual loss of safety function of the "A" EDG, or involve seismic, flooding, or severe weather initiating events. This finding was related to the cross-cutting area of Human Performance in that Dominion personnel signed the PMT as satisfactory and restored the EDG neutral breaker to an operable status although the acceptance criteria was not met.
Inspection Report# : 2005002*(pdf)*

Significance: Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO TAKE PROMPT CORRECTIVE ACTIONS TO DETERMINE THE EXTENT OF CONDITION OF AIR TRAPPED IN THE RHR SUCTION AND DISCHARGE PIPING
The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," which requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. From May to October 2004, Dominion failed to properly assess and correct a degraded "A" Residual Heat Removal (RHR) system during an extent of condition examination for air found in the RHR discharge piping. Specifically, after discovering a significant amount of air in the "A" RHR piping system in May 2004, Dominion vented the system but did not adequately evaluate whether the corrective actions were effective in removing air from the RHR heat-exchanger tubing. As a result, Dominion did not evaluate the effect of the remaining air on the RHR and high pressure injection systems. Dominion subsequently instituted compensatory measures to vent the suction piping after every RHR pump run and performed a special procedure to flush the air out of the heat exchanger. This finding was more than minor because it affected the equipment performance attribute and the availability, reliability, and capability objective of the Mitigating system cornerstone. Specifically, Dominion's extent of condition evaluation did not determine that a significant volume of air remained in the "A" RHR heat exchanger tubing even though air was found in several other sections of piping subsequent to their initial corrective actions. This air could have caused the "A" RHR pump to become inoperable if enough air had migrated to the suction of the RHR pump and could have adversely affected high pressure injection pumps if air had migrated to crossover piping. This finding was determined to be of very low safety significance (Green) since an actual loss of RHR would not have occurred with the amount of air identified and no air pockets were subsequently identified in crossover piping to the charging and high pressure injection systems; the finding did not involve a design or qualification deficiency; or involve seismic, flooding, or severe weather initiating events. This finding was related to the cross-cutting area of Problem Identification and Resolution in that Dominion failed to perform an adequate extent-of-condition review to fully evaluate the effect of air that had been introduced into the "A" RHR system.
Inspection Report# : 2005002*(pdf)*

Significance: Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO PROPERLY IMPLEMENT TS 3.8.3.2, ONSITE POWER DISTRIBUTION - SHUTDOWN
The inspectors identified a non-cited violation of Technical Specification (TS) 3.8.3.2, Onsite Power Distribution - Shutdown, for the failure to enter Technical Specifications following the loss of a vital inverter. The required actions were to immediately stop all reactivity additions. However, operators failed to stop both a plant heatup and reactor coolant system (RCS) dilutions (hydrazine addition), which resulted in positive reactivity additions to the reactor. Dominion specified operator training to reinforce the management expectation for completing procedures, however, additional corrective actions will be specified in an upcoming revision to the Licensee Event Report based on the issues identified by the inspectors in the finding description. Dominion has entered this issue into their corrective action program. This issue is more than minor because it is associated with the human performance attribute of the Mitigating System Cornerstone and the objective of ensuring the availability of systems to respond to initiating events to prevent undesirable circumstances. The failure of the vital inverter resulted in an electrical lineup that did not meet the TS requirements for one complete train of electrical buses. Additionally, the failure to recognize the need to enter TS precluded taking corrective actions to prevent adding positive reactivity with this electrical lineup. Several positive reactivity additions from heatup and RCS dilutions occurred as a result. The finding is of very low safety significance because the reactivity addition from the heatup and the dilutions was small compared to the reactivity needed for criticality. Additionally, the finding did not increase the likelihood of a loss of RCS inventory, degrade Dominion's ability to add inventory if needed, or degrade the ability to recover the residual heat removal system if it was lost. This finding is related to the cross-cutting issue of Human Performance.
Inspection Report# : 2004007*(pdf)*

Significance: Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
DOMINION FAILED TO ESTABLISH PRECAUTIONS AND PREREQUISITES TO PREVENT PLANT CONFIGURATION CHANGES THAT COULD LEAD TO AIR ENTRAINMENT IN THE RHR SYSTEM
The inspectors identified a non-cited violation of Technical Specification (TS) 6.8.1a for the failure to adequately implement procedures for venting the reactor coolant system (RCS) and the residual heat removal (RHR) system. On May 28, 2004, Dominion conducted a quarterly vent and valve lineup of the "A" train of the RHR system in which air was vented from several vent valves. The inspectors investigated whether the voids in the "A" train of the RHR system and portions of suction piping leading to both trains of the safety injection (SI) and charging systems would have adversely affected these systems' ability to respond to a small break loss of coolant accident (SBLOCA). The inspectors reviewed the engineering technical evaluation and determined that the amount of air in the RHR system did not adversely impact the RHR pumps, SI pumps, or the charging pumps. The inspectors reviewed Dominion's root cause investigation and determined that the cause of the entrapped air was due to securing one of the two RHR pumps on April 28, 2004, during the RCS sweep and vent procedure following completion of the refueling outage. Dominion revised the RCS sweep and vent procedure to add a precaution to avoid securing an RHR pump during this procedure. Dominion has entered this issue into their corrective action program. This issue is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and the objective to ensure availability of systems that respond to initiating events to prevent undesirable consequences. The entrapped air had the potential to make the "A" RHR pump, SI pumps, and charging pumps inoperable. The finding is of very low safety significance because it did not represent an actual loss of safety function of the RHR, SI, or charging system since the amount of air identified in these systems would not have prevented them from functioning. This finding is related to the cross-cutting issue of Human Performance.
Inspection Report# : 2004007*(pdf)*

Barrier Integrity

Significance: Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO ADEQUATELY PERFORM POST-MAINTENANCE TESTING ON HYDROGEN RECOMBINER
The inspectors identified a non-cited violation of Technical Specification (TS) 3.6.4.2, "Electric Hydrogen Recombiners," which requires that two independent hydrogen recombiner systems remain Operable. On February 22, 2005, Dominion performed maintenance on the "A" train hydrogen monitor. On February 23, 2005, Dominion identified that pipe fittings for the "A" train hydrogen monitor had been disassembled, however, a post-maintenance test had not been conducted to prove operability of the system. Dominion performed a leak test on February 24, 2005, however, the test failed. Dominions' investigation determined that the leakage was from a mechanical joint that had been worked on December 2, 2004, but that this joint had not been disturbed during the February 22, 2005, maintenance. Additionally, Dominion determined that following the work in December 2004 no post-maintenance leak test had been performed to verify system operability. The inspectors identified that the leakage would have resulted in the shutdown of the "A" hydrogen recombiner, under post-accident conditions. Therefore, the train would not have been considered operable from December 2, 2004 to March 1, 2005. Following the identification of the failed joint, Dominion repaired the joint, leak tested the system, and restored the "A" train hydrogen monitor to service. This issue was more than minor because it was associated with the Barrier Integrity cornerstone attribute of configuration control in that it affected containment boundary preservation and maintaining containment design parameters. The failure to specify adequate PMT resulted in loose mechanical joints in the system not being detected which would have allowed an open pathway to the atmosphere from containment during post accident conditions. Additionally, Dominion postulated that the post accident leakage from these joints would have caused a radiation monitor alarm which would have isolated the "A" hydrogen recombiner. This violation was evaluated using an IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," Phase 2 analysis, and was determined to be of very low safety significance (Green). Specifically, the leak was not of the magnitude to recycle the containment atmosphere in a 24 hour period, post event. This finding was related to the cross-cutting issue of Human Performance in that Dominion failed to adequately perform post-maintenance testing to ensure incorrect maintenance activities were identified prior to returning the hydrogen monitor to service.
Inspection Report# : 2005002*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Last modified : August 24, 2005

Exelon.
Nuclear

ACRS Presentation
September 8, 2005

Early Site Permit Application
Clinton Power Station Site
Draft Safety Evaluation Report

SUNSI

5

Exelon. Nuclear

Agenda

- ESP Project Team
- General ESP Information
- ESP Site Information
- SSAR/EP Development Approach
- Geotechnical Results
- Seismic Analysis Demonstration
- Ground Motion Determination Methodology

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Nuclear

Exelon ESP Project Team

- CH2M Hill (Prime Contractor)
  - Environmental / Redress
  - Geotechnical
  - EP
- CH2M Hill Subcontractors
  - WorleyParsons
    - Safety
  - Geomatrix
    - Seismic
  - Seismic Board of Review
    - Expert, independent review
  - Others
- RPK Structural Mechanics Consulting
  - Seismic
- Sargent and Lundy
  - Draft Application Review
- Morgan Lewis
  - Legal counsel

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General ESP Information

- 10 CFR, Part 52, Subpart A, "Early Site Permits"
- EGC Application content
  - Administrative Information
  - Site Safety Analysis Report (SSAR)
  - Emergency Planning Information
  - Environmental Report (ER)
  - Site Redress Plan

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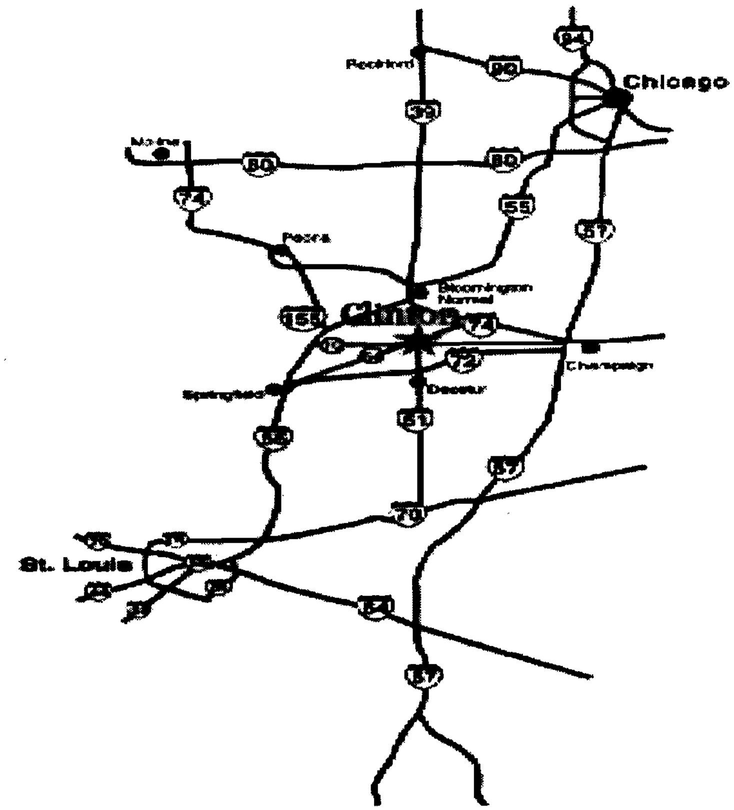
Nuclear

General ESP Information (cont'd)

- Applicant
  - Exelon Generation Company, LLC (EGC)
    - Wholly owned subsidiary of Exelon Corporation
- ESP Site Location
  - Central Illinois
  - Clinton Power Station Property
  - AmerGen Owned (EGC Subsidiary)



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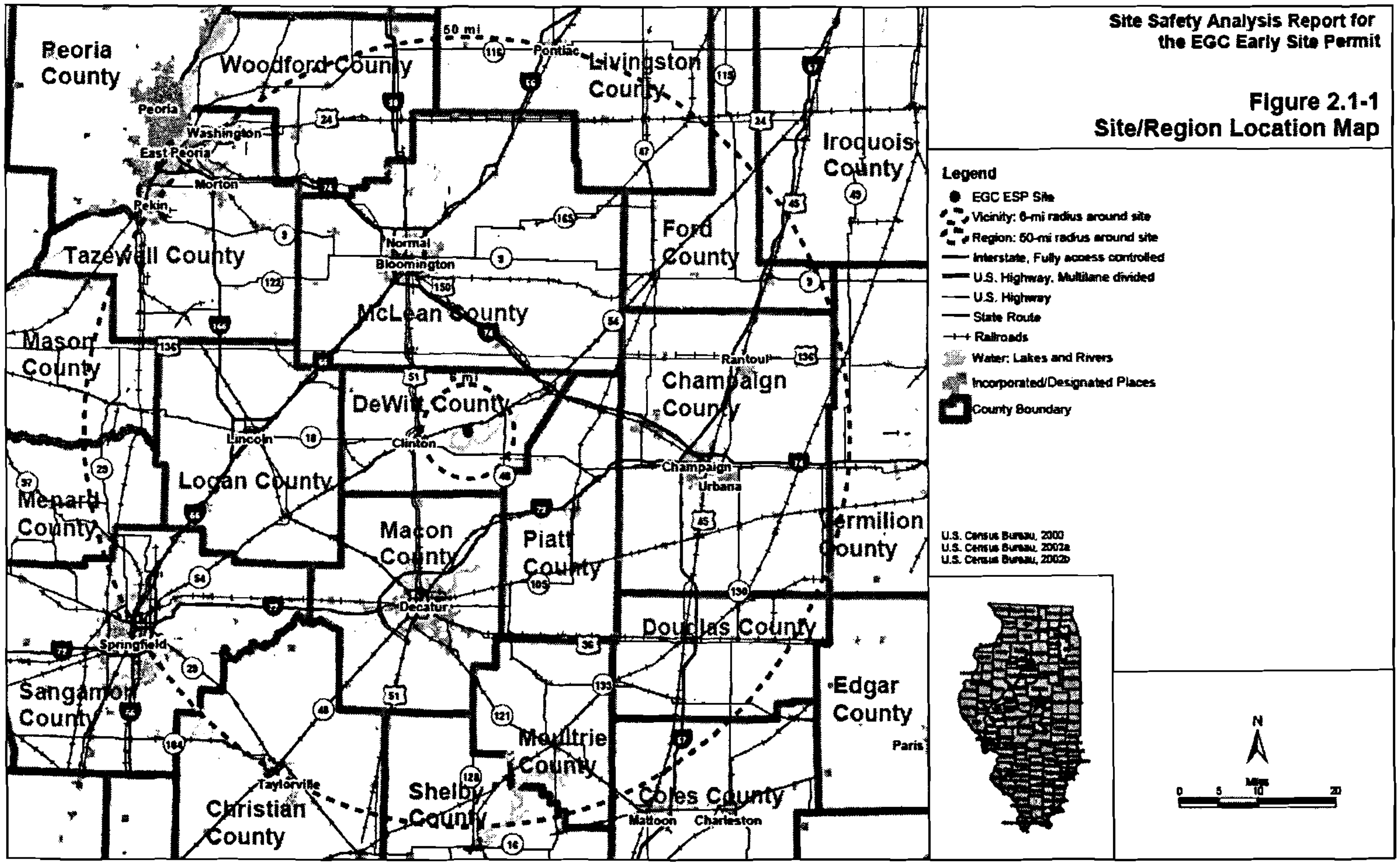
ESP Site Information



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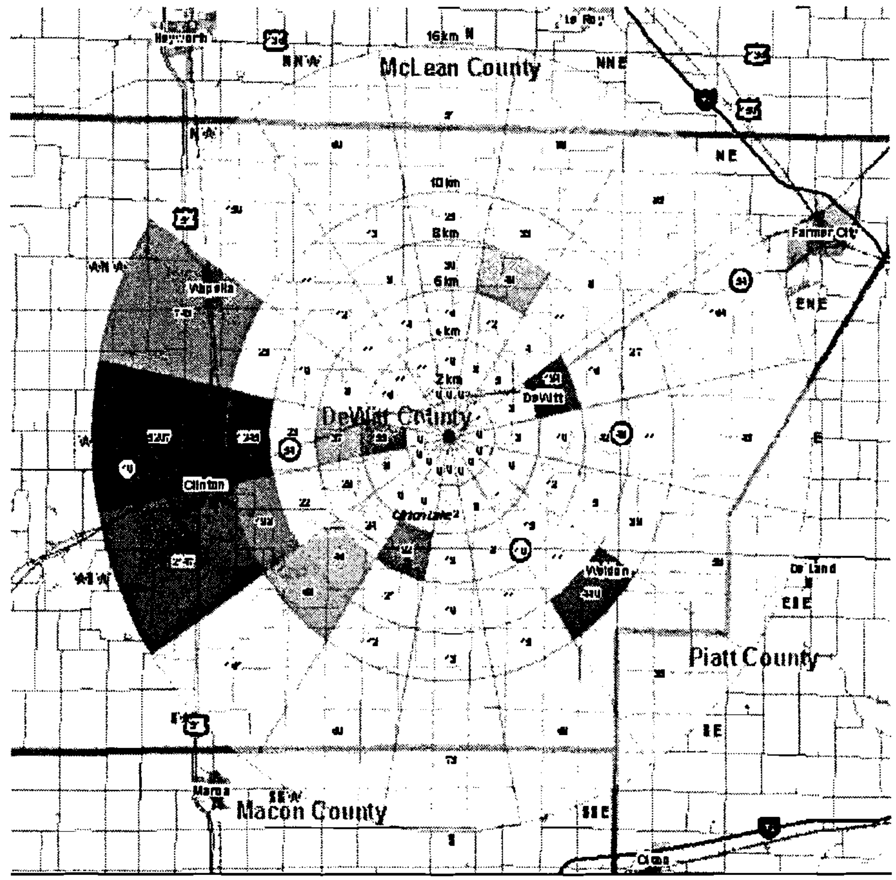
ESP Site Information (cont'd)

- 10 mi. EPZ
- Mostly rural
  - Clinton (W)
  - DeWitt (E)
  - Weldon (SE)
  - Wapella NW)



Persons per Sq Mi
- 0 - 20
- 21 - 50
- 51 - 100
- 101 - 200
- > 200

Census 2000 Population shown for each radial grid sector

ESP Site Information (cont'd)

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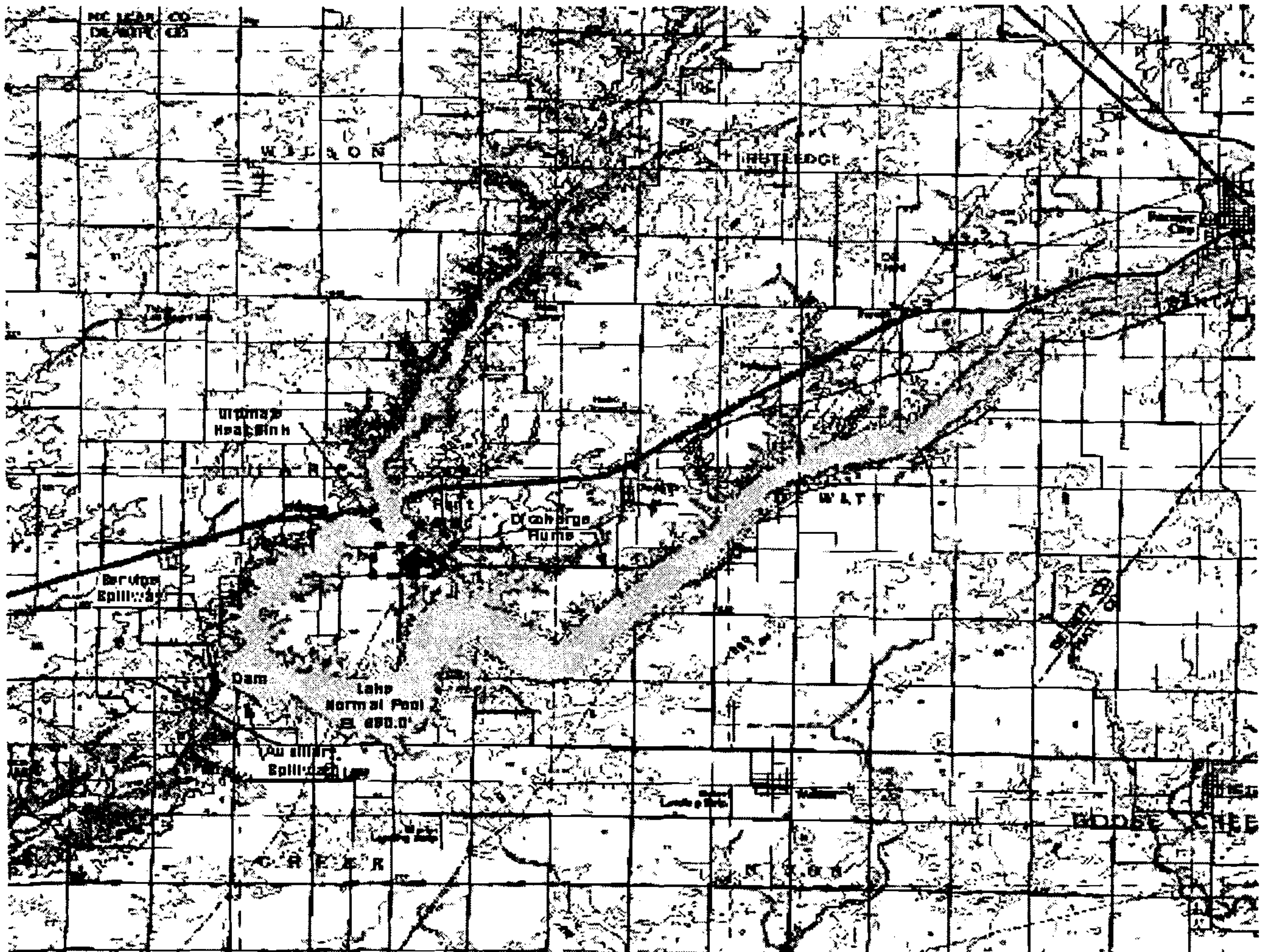

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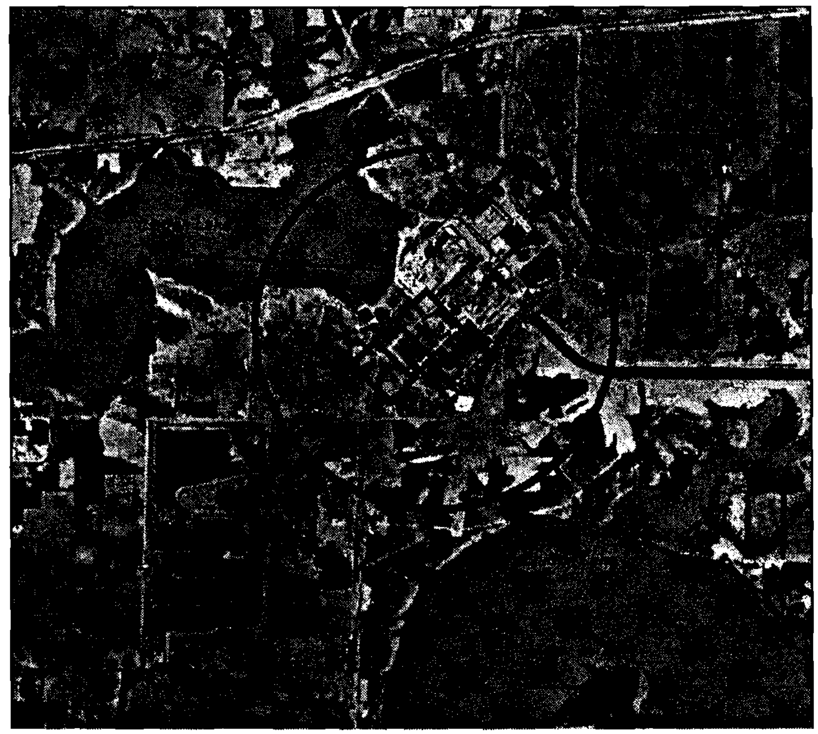
ESP Site Information (cont'd)

- ESP Location
  - Exclusion Area Boundary (EAB)
  - Power Block Footprint
  - Heat Sinks
  - Intake Structure

- Other Items
  - Clinton Lake
  - CPS UHS (baffle)
  - Discharge Canal



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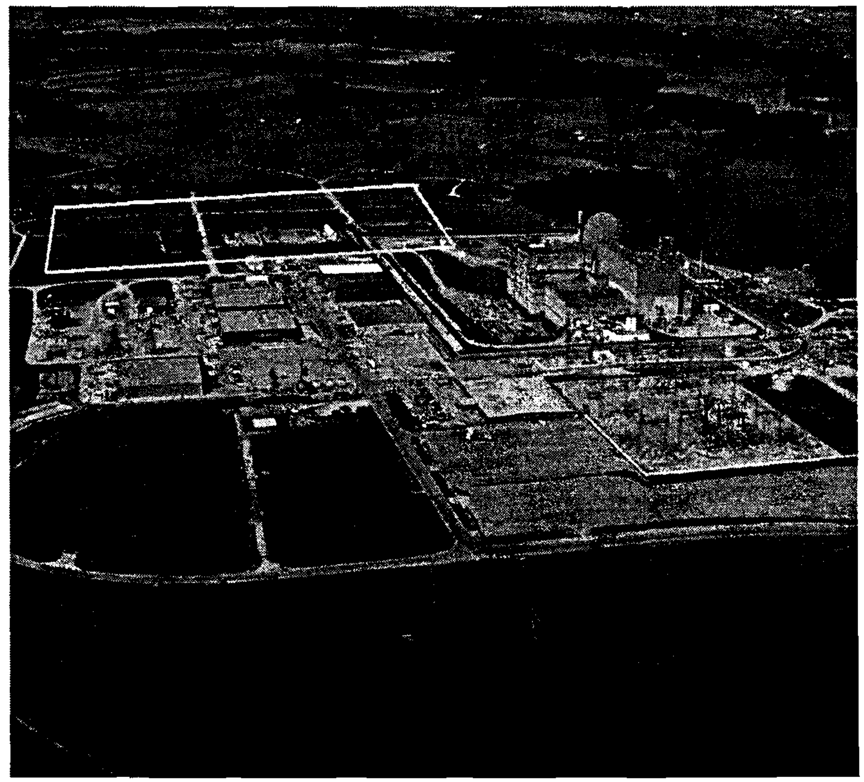
ESP Site Information (cont'd)

- ESP Location
  - (Yellow outline)
- CPS
  - (Red outline)
    - Cancelled Unit 2 area
- Clinton Lake & UHS



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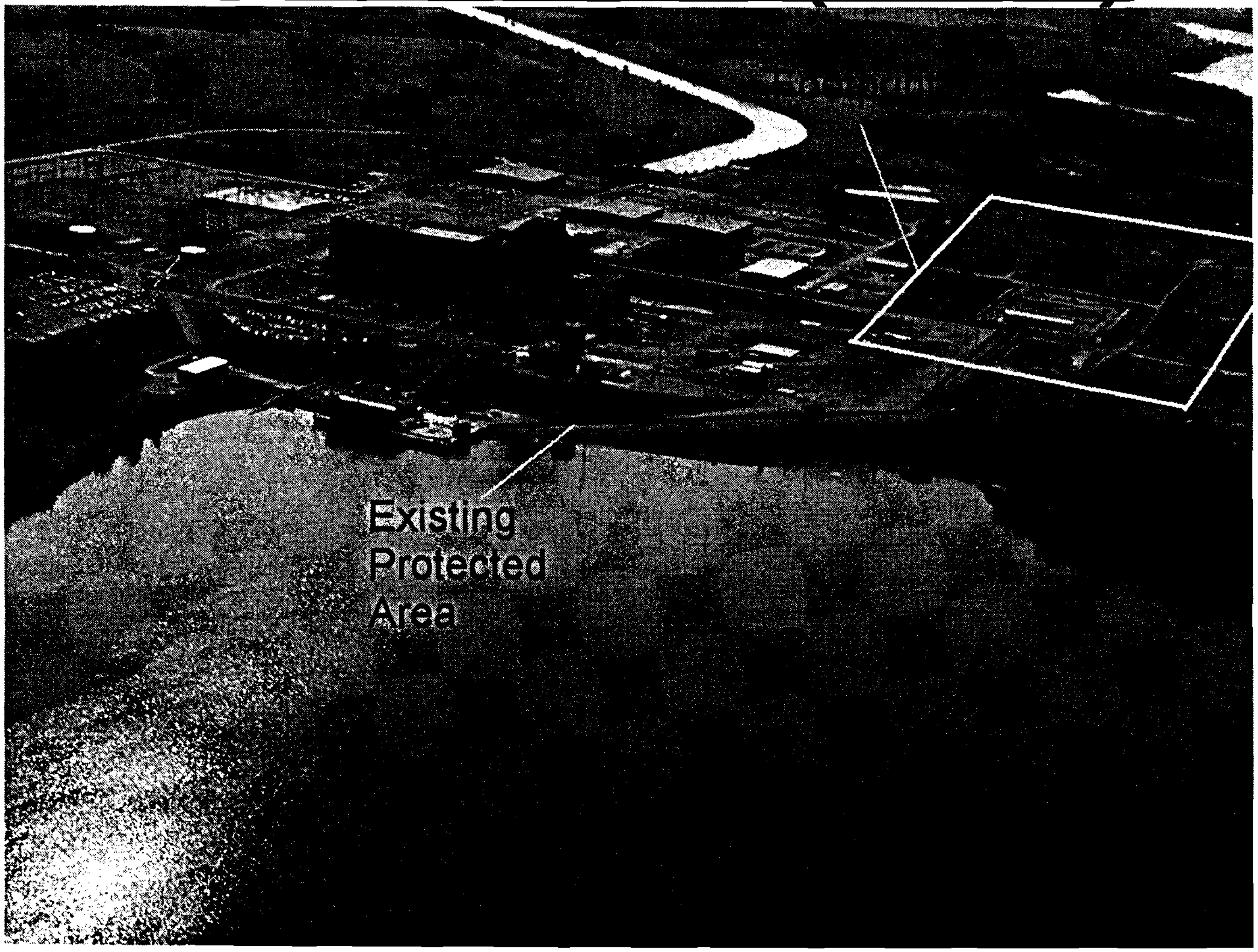
ESP Site Information (cont'd)



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SSAR/EP Development Approach

- Site Safety Analysis Report (SSAR)
  - Maximum use of existing information
    - Evaluate and update as necessary
  - Gather new data
  - Based on "plant parameter envelope"

- Emergency Planning Information
  - Maximum use of existing plans
  - Establish "major features"

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Geotechnical Results

- Confirmed ESP local soil properties similar to the established CPS soil properties
  - Sufficient information to establish site geotechnical characteristics for ESP
  - Updated dynamic soil properties
- Site suitable for future development

Seismic Analysis Demonstration

- Establish analysis precedent for new nuclear plants through use of:
  - Latest industry methods
  - Analysis consistent with risk-informed philosophy
  - Methods that achieve regulatory stability
- Performance based methodology
  - Advocated by nuclear industry
    - NEI Seismic Issues Task Force / EPRI

SSE Ground Motion Determination

- Complies with 10 CFR 100.23
- Applies RG 1.165 guidance
  - One variation
    - Uses ASCE Standard 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities"
      - Performance based criterion
      - Industry consensus standard

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SSE Ground Motion Determination Methodology Comparison

RG 1.165 Methodology	*EGC Application*
➢ Investigations	➢ Same
➢ Seismic sources update	➢ Same
➢ SSHAC assessment	➢ Same
➢ PSHA	➢ Same
➢ Determine SSE ground motion spectra • **Relative based -- Reference Hazard Probability Criterion**	➢ Determine SSE ground motion spectra • **Performance based – Core Damage Frequency Criterion**

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Methodology Comparison (cont'd)

RG 1.165 Methodology

- De-aggregate to identify controlling earthquakes
- Account for site effects

EGC Application

- Same
- Same [NUREG/CR-6728]

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Methodology Comparison (cont'd)

Reference Hazard
RG 1.165, App. B

- Reference probability
  - The annual probability level such that 50% of the set of most modern design currently operating plants has an annual median probability of exceeding the SSE that is below this level (1E-5) determined at an average of the 5 and 10 Hz SSE spectra with 5% damping.

Performance-based
ASCE 43-05

- Performance Based
  - SSCs will have a target mean annual frequency of 1E-5 for seismic induced onset of significant inelastic deformation.
  - Significant margin against SSC failures that might lead to core damage.
  - Leads to seismically induced CDF significantly less than for existing plants

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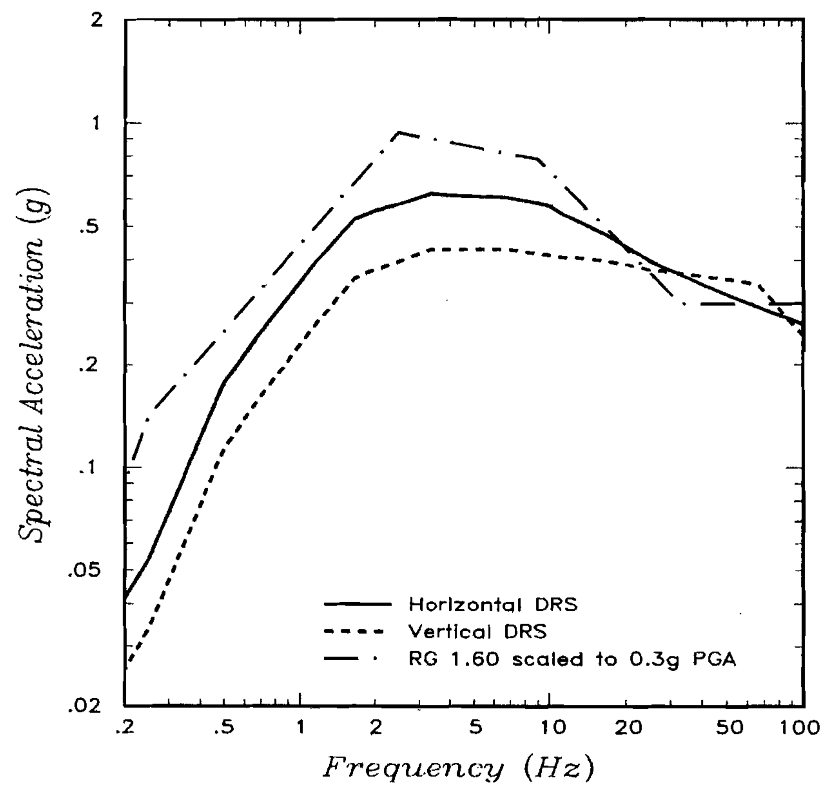
EGC ESP SSE Ground Motion Spectra

- Performance Based
  - Horizontal DRS
  - Vertical DRS
  - RG 1.60 0.3g PGA



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SUMMARY

- ESP site next to existing operating nuclear plant
- Maximized use of existing information
- Plant parameter envelope established
- Site characteristics identified
  - Geotechnical - Simple and suitable site geology
  - Determined SSE ground motion
    - Evaluated using latest regulatory guidance and industry practice
- Requesting 20-year ESP

Exelon Early Site Permit
Safety Review Status

September 8, 2005

Advisory Committee on Reactor Safeguards
Early Site Permit Full Committee Meeting

John Segala, Senior Project Manager
Office of Nuclear Reactor Regulation

Purpose

- Brief the Full Committee on the Exelon early site permit (ESP) application and the status of the NRC staff's safety review
- Provide overview of the remaining open items
- Support the Full Committee's review of the application and subsequent interim ACRS letter
- Answer the Full Committee's questions

Meeting Agenda

- Key Review Areas
- Permit Conditions/COL Action Items
- DSER Conclusions
- Open Items
- Schedule Milestones
- Presentation Conclusions
- Discussion / Subcommittee questions

Key Review Areas

- Exclusion Area Authority and Control
- Nearby Industrial, Transportation, and Military Facilities
- Meteorology
- Hydrology
- Seismology and Geology
- Radiological Effluents
- Thermal Discharges
- Radiological Consequences of Accidents
- Physical Security
- Aircraft Hazards
- Emergency Planning
- Quality Assurance

Principal Contributors

Brad Harvey - Meteorology

Goutam Bagchi – Hydrology

- Contract support from PNNL

Kazimieras Campe - Site Hazards

- Contract support from PNNL

Clifford Munson and Tom Cheng – Geology, Seismology, and Geotechnical

- Support from U.S. Geologic Survey and BNL

Jay Lee – Demography, Geography, and Radiological Consequence Analysis

Robert Moody - Emergency Planning

- Consultation with FEMA

Paul Prescott - Quality Assurance

Al Tardiff - Physical Security

Proposed Permit Conditions and COL Action Items

- There are 15 proposed Permit Conditions

- There are 17 proposed COL Action Items

- Applying new criteria developed during the review of the North Anna ESP application

DSER Conclusions

- DSER defers conclusion regarding site safety and suitability to FSER after open items addressed
- Some conclusions from individual sections without open items:
  - Potential hazards associated with nearby transportation routes, industrial and military facilities pose no undue risk to facility that might be constructed on the site.

DSER Conclusions

- Additional conclusions from individual sections without open items
  - The proposed site is acceptable for constructing a plant falling within the PPE with respect to radiological effluent release dose consequences from normal operation
  - Site characteristics are such that adequate security plans and measures can be developed

Open Items

Review Area	Open Items
Exclusion Area Authority and Control	1
Meteorology	3
Hydrology	21
Seismology and Geology	7
Radiological Consequences of Accidents	1
Emergency Planning	6
Quality Assurance	1
Total:	40

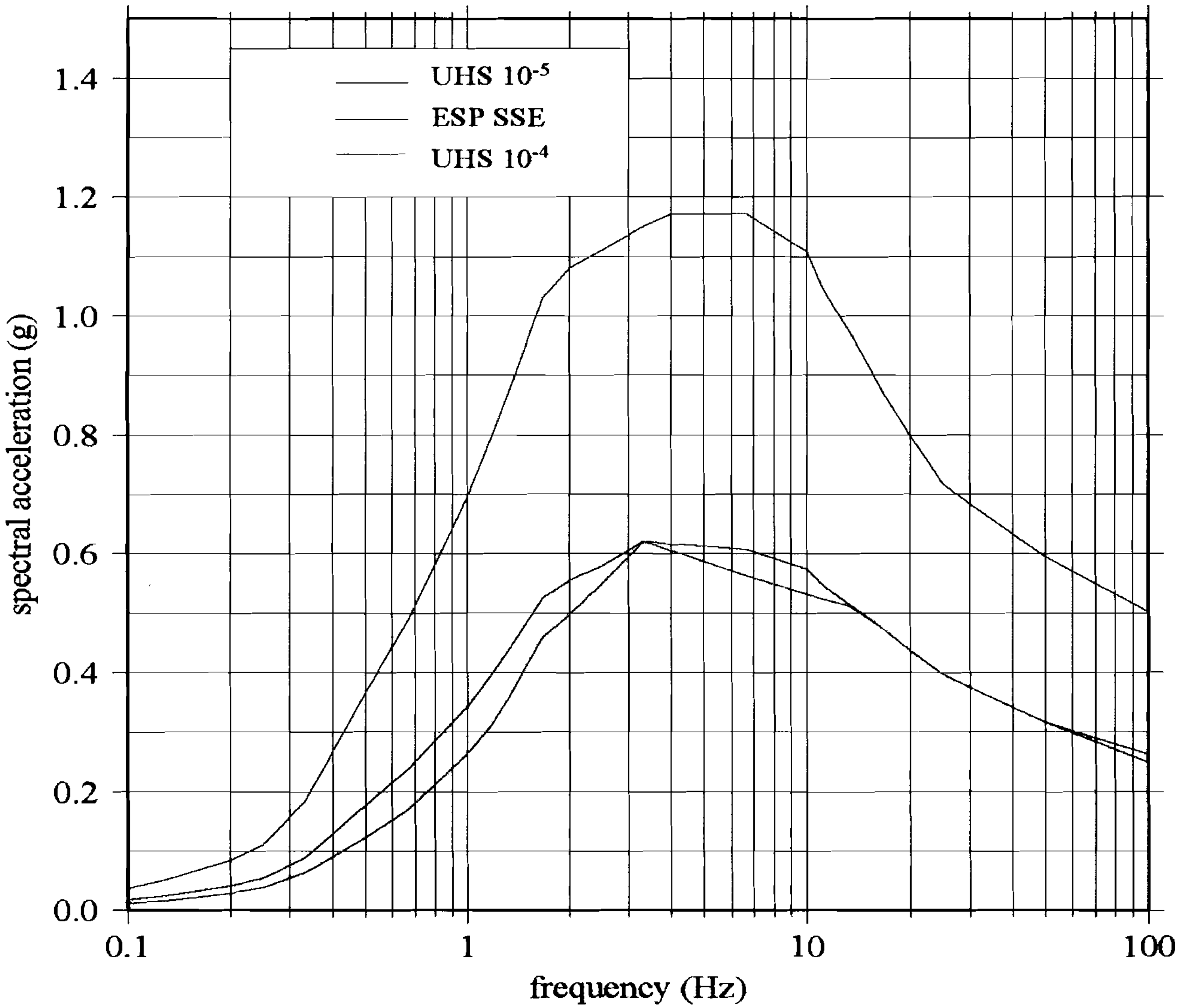
Seismic Open Items 2.5.2-4 and -5

- Exelon proposed new "performanced-based" approach for determining safe shutdown earthquake (SSE)
  - Not entirely consistent with NRC-approved method in RG 1.165
  - ASCE Standard 43-05 describes this approach
  - Risk-based approach that targets performance goal
    - $1x10^{-5}$ annual probability of unacceptable performance under seismic loading of Category 1 SSCs
    - Target performance probability based on seismic PRAs for existing nuclear power plants
  - Staff reviewed applicant's final SSE to determine the appropriateness of the performance-based approach

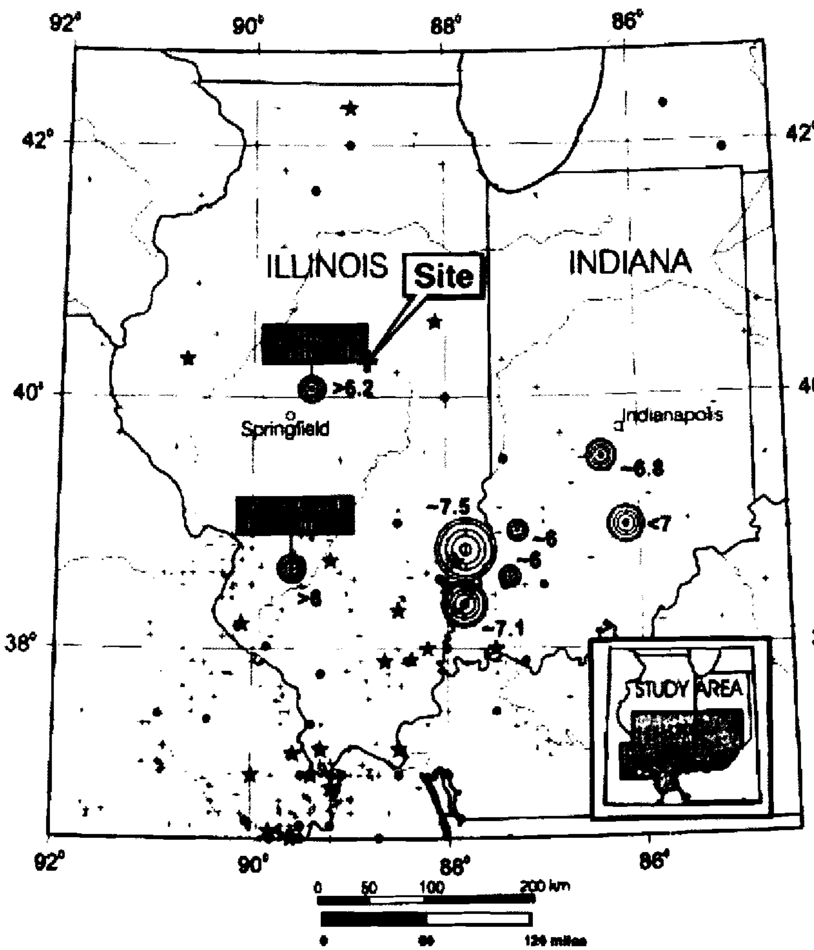
Seismic Open Items 2.5.2-4 and -5

- Open Item 2.5.2-4:
  - The performance-based SSE spectrum for the ESP site is approximately equal to the mean 10^{-4} uniform hazard spectrum
  - The performance-based SSE at 10^{-4} may not adequately represent the seismic hazard from local earthquakes
- Open Item 2.5.2-5:
  - Assumptions underlying the performance-based approach

Comparison of performance-based SSE spectrum for the ESP site and the mean 10-4 and 10-5 spectra



Historical Seismicity and Estimated Centers of Large Prehistoric Earthquakes in Site Region



A star represents a magnitude of 5 or higher. A solid circle represents a magnitude between 4.5 and 5. A plus sign represents a magnitude between about 2.3 and 4.5. Historical earthquake data are from USGS/NEIC Global Hypocenter Data Base CD-ROM (Version 3.0). Concentric circles show estimated energy centers of large prehistoric earthquakes. The estimated moment magnitude M for a prehistoric earthquake is located near the circle

Seismic Open Items 2.5.2-4 and -5

The performance-based approach with a target 10^{-5} annual performance goal may not be suitable for determining the SSE for the Clinton ESP site

Other Seismic Open Items

- 2.5.1-1, Incorporate most recent New Madrid seismic source model into the PSHA and SSE

- 2.5.2-1, Clarify and justify the EPRI ground motion attenuation study distance-conversion method

Geotechnical Open Items

- 2.5.2-2, Site response model does not adequately represent variability of soil properties
- 2.5.2-3, Site response analysis should use appropriate shear modulus and damping curves
- 2.5.4-1, Further soil exploration needed for COL

Completed Milestones

- Received Exelon ESP application - September 25, 2003
- FRN published announcing acceptance – October 31, 2003
- FRN published for mandatory hearing – December 12, 2003
- RAIs issued to the Applicant – July, 27, 2004
- Draft SER issued – February 10, 2005
- Applicant responds to Draft SER open items – April 26, 2005
- Supplemental Draft SER issued – August 26, 2005
- ACRS Subcommittee Meeting - September 7, 2005

Remaining Milestones

- ACRS interim letter assumed – September 28, 2005
- Staff provides Final SER to ACRS – February 8, 2006
- Staff issues Final SER – February 17, 2006
- ACRS Full Committee Meeting – March 9, 2006
- ACRS letter assumed – March 30, 2006
- Staff incorporates ACRS letter and issues Final SER as NUREG – May 1, 2006
- Mandatory hearings begin Fall 2006
- Commission decision assumed mid 2007

Summary

- All open items resolved except for:
  - 7 Seismic open items
  - 1 Hydrology open item

- Working to resolve the remaining open items

- Looking forward to receiving the interim ACRS letter

- Questions or comments?

Proposed Revisions to Generic License Renewal Guidance Documents

Jerry Dozier
Amy Hull

Office of Nuclear Reactor Regulation (NRR)
Division of Regulatory Improvement Programs (DRIP)
License Renewal & Environmental Impacts Program
License Renewal Section B

Presented at the 525th ACRS Meeting
September 9, 2005

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Agenda and Introduction

- Schedule
- Focus on License Renewal Guidance (LRG) documents for safety review
  - per 10 CFR Part 54 *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*
- Overview of selected significant changes since the last ACRS meeting (3/4/05)

Revised LRG Documents

- NUREG-1800, Rev. 1, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR)*
- NUREG-1801, Rev. 1, *Generic Aging Lessons Learned* (GALL) *Report*
- RG 1.188, Rev. 1, *Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses*

New LRG Documents

- NUREG-1832, *Analysis of Public Comments on the Revised License Renewal Guidance Documents*
- NUREG-1833, *Technical Bases for Revision to the License Renewal Guidance Documents*

Schedule: Looking Ahead

Date	Activity
9/13/2005	CRGR meeting
9/30/2005	GALL, SRP-LR, RG 1.188, NUREG-1832 in ADAMS and on Website
10/31/2005	NUREG-1833 in ADAMS and on Website
10/31/2005	Official bound copies of GALL, SRP-LR, RG 1.188, NUREG-1832 available
11/30/2005	Official bound copies of NUREG-1833

NUREG-1832, *Analysis of Public Comments*

- Appendix A - NEI Comments
- Appendix B - ACRS Comments
- Appendix C - Comments from the 3/02/05 workshop
- Appendix D - Public stakeholder comments
- Appendix E - Comparison of the AMR line-items from 1/05 GALL to 9/05 GALL

Federal Register Notice Request

- Requested comments for changing aging management review (AMR) line-items from “plant-specific” to generic aging management programs (AMP)
- Our subsequent resolution included pointing to existing AMPs and in some cases developing new AMPs

Rationale for New AMPs

- Provide generic program that can be credited in an AMR line-item
- Incorporate Interim Staff Guidance
- Provide an acceptable way to address an emerging issue

New AMPs for Mechanical Systems

XI.M11A	*Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of PWRs*
XI.M35	*One-time Inspection of ASME Code Class 1 Small-Bore Piping*
XI.M36	*External Surfaces Monitoring*
XI.M37	*Flux Thimble Tube Inspection*
XI.M38	*Inspection of Internal Surfaces in Miscellaneous Piping & Ducting Components*
XI.M39	*Lubricating Oil Analysis*

New AMPs for Electrical Systems

XI.E4 *Metal-Enclosed Bus*

XI.E5 *Fuse Holders*

XI.E6 *Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements*

- Transparent Process

NRC: Schedule and Background For Guidance Updates - Microsoft Internet Explorer provided by USNRC

File Edit View Favorites Tools Help

Address http://www.nrc.gov/reactors/operating/licensing/renewal/guidance/updated-guidance.htm

Date	Description
08/23/05	Excel spreadsheet providing AMR line items in Draft (August Version) GALL Volume II

08/15/05	NRC staff has transmitted the following draft documents for ACRS Review:
	• NUREG-1800, Rev. 1, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants
	• NUREG-1801, Rev. 1, Generic Aging Lessons Learned (GALL) Report
	  o Volume 1
	  o Volume 2
	• NUREG-1832, Analysis of Public Comments on the Revised License Renewal Guidance Documents
	• RG 1.188, Rev. 1, Regulatory Guide for Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses which endorses, NEI 95-10, Rev. 6, Industry Guidelines for Implementing the Requirements of 10 CRF Part 54 - The License Renewal Rule

06/10/05	Written comments on the License Renewal Guidance Documents • Public • NEI
05/16/05	Public Meeting to discuss Interim Staff Guidance and selected NEI comments on the License Renewal Guidance Update (5/16/05, 9:00-4:00, 01F16) • Notice • Meeting Summary

ACRS Issues

- NUREG-1833
  - Provides link for Interim Staff Guidance and revised documents (located in affected sections)
  - Traceability of GALL'01 AMR line-items (Appendix C)
- Clarify under what circumstances aging effects would be expected from halon/carbon dioxide in the fire suppression system
- Risk-Informed ISI

RG 1.188, Rev.1, Endorses NEI 95-10, Rev.6

- NEI incorporated NRC comments on two previous exceptions to NEI 95-10, Rev 5:
  - Exposure duration criteria
  - Criteria for scoping of non-safety-related piping and supports

Endnote

- The success of this update process is due to the efforts of numerous NRC staff, contractors, and stakeholders.
- The collection of interrelated documents reflect the staff's current position (based on technically rigorous and generically applicable precedents) and considers stakeholder comments and interactions.

Conclusion

- The update was completed in about 14 months (initial contract June 2004)
- A process for continuing stakeholder dialogue and resolution is in place
- The new documents increase the efficiency, effectiveness, and consistency of the license renewal review

(8)

ACRS Meeting with EDO and Office Directors

Jim Dyer, Director

Office of Nuclear Reactor Regulation

September 9, 2005

Overview

License Renewal Program

New Reactors

Power Uprate Issues

Fire Protection

PWR Sump Performance

2

License Renewal Program

Renewal status
- Approximately 50% of plants either received renewed licenses or are currently under review

Guidance document updates
- ACRS Full Committee Meeting: 9/9/05
- Final version to be issued: 9/30/05

Future reviews
- Projected to receive approximately 6 applications per year for the next 4 - 5 years

3

New Reactors

ACRS review/support will be needed for the following:
- Design certification: ESBWR and AP1000 Rulemaking
- Early site permit reviews
  - Completed: North Anna (ACRS meeting held: 7/6/05)
  - Scheduled: Grand Gulf (ACRS meeting scheduled: 12/8/05) and Clinton (ACRS meeting scheduled: 3/9/06)
  - Submittal Planned: Southern Nuclear Operating Company (Summer 2006)
- Infrastructure: 10 Part 52 Proposed Rule, update of infrastructure

Combined licenses reviews are planned for:
- FY 2007: Dominion
- FY 2008: NuStart (2 applications), Duke, and Progress Energy
- TBD: South Carolina Electric & Gas

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Power Uprates

BWR steam dryer issues
- Achieving resolution to steam dryer failures
- Better understanding of steam dryer loadings with extended power uprate (EPU)

New technical challenge
- Accident/Transient analysis codes and methods issues

Use of EPU Review Standard RS-001

Power uprate review status
- 12 PU applications under NRC review (7 are EPUs)
- 20 PU applications in next 5 years (3 EPUs in FY 06)

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Fire Protection

Performance-Based Fire Protection Rule (NFPA 805)
- ACRS Meeting on Draft Regulatory Guide: 10/05

Circuit Issue Resolution
- ACRS Meeting on Draft Generic Letter: 2/06

Hemyc/MT Fire Barrier
- ACRS Meeting on Draft Generic Letter: 12/05

Manual Actions Rulemaking
- Public Meeting on Issue Closeout: 9/05

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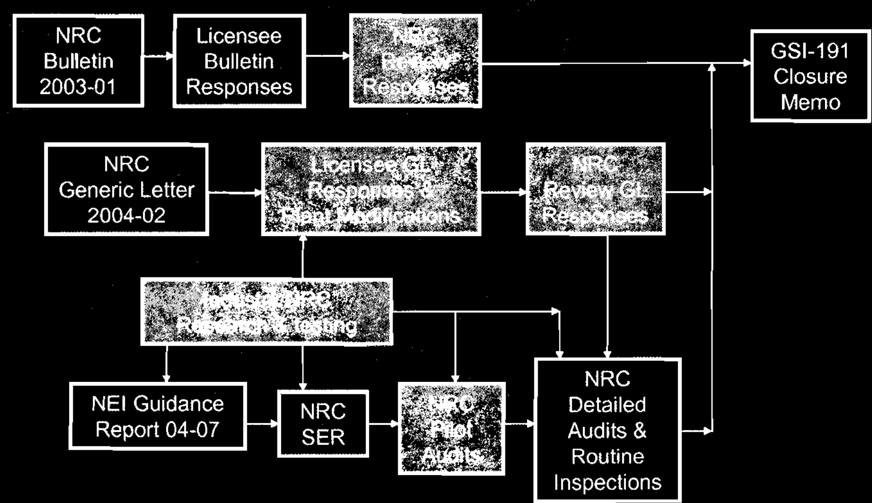
PWR Sump Performance (GSI-191)



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PWR Sump Performance (GSI-191) Expected Schedule for Key Activities

NRC Pilot Audits: 6/05 to 10/05
Licensee GL Submittals: 9/1/05
Research Test Activities: through 4/06
- ACRS Subcommittee Meetings: 12/05 and 4/06

NRC Audits/Inspections to be completed by 12/31/07
- ACRS Subcommittee Meeting by Spring 2007

Plant Modifications to be Completed by 12/31/07
GSI-191 Closure: 6/30/08
- ACRS Subcommittee Meeting: Winter 2008
- ACRS Full Committee Meeting: Spring 2008

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Research Supporting GSI-191 Resolution

NRC staff briefed ACRS subcommittee in July 2005

Research is focused in four technical areas:

- Chemical effects
- Head loss
- Downstream effects
- Coating transport

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ACRS Meeting with EDO and Office Directors

Dr. Carl Paperiello, Director
Office of Nuclear Regulatory Research
September 9, 2005

Overview

Technology Neutral Framework

Other Expected Major Topics

Review of NRC Research Program

Assessment of Quality of Selected NRC Research Projects

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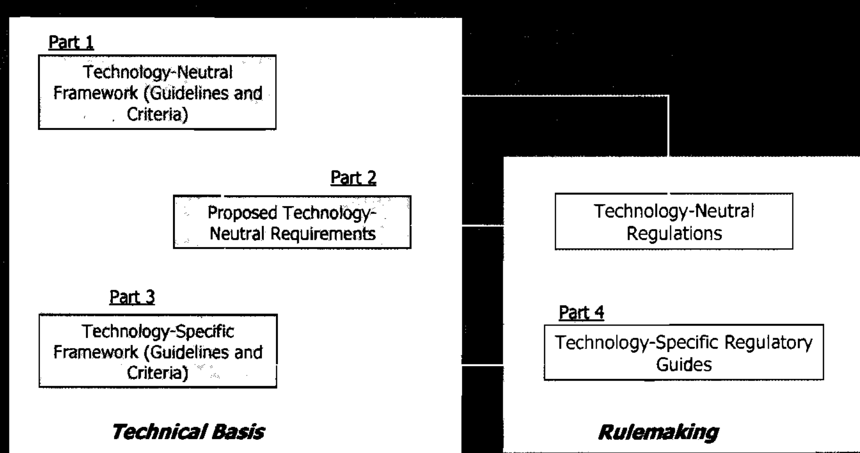
Regulatory Structure for Technology Neutral Framework (TNF)

(Focus of TNF is on reactor designs beyond those currently under review)



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Regulatory Structure for Technology Neutral Framework

- Staff has focused on the technology neutral framework
  - Initiated efforts to start testing the criteria
  - Two policy issue currently under review (level of safety and integrated risk)
- Draft framework for public review and comment scheduled for June 2006
  - Discuss staff position with ACRS on policy and technical issues
- Staff initiating work on the other parts
  - Requirements, Framework, and Reg Guides

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Framework Issues to be Discussed with ACRS, Examples

- Probabilistic approach to establish plant licensing basis
- Defense in Depth (DID)
- Containment performance standards
- Emergency planning considerations
- PRA requirements
- Integration of security into the design

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Other Major Topics

I & C Research Plan and Results
Regulatory Guides
Generic Issues
- **GI 80, Pipe Break Effects on CRDM**
- **GI 186, Heavy Loads**
- **GI 193, BWR ECCS Suction**

Codes
- **Thermal Hydraulics**
- **Severe Accidents**

Human Reliability Analysis
Risk Informing Part 50
PRA Standards and RG 1.200

ACRS Meeting with EDO and Office Directors

Roy Zimmerman, Director
Office of Nuclear Security and Incident Response
September 9, 2005

EP Initiatives

Post 9/11 EP Rulemaking

Post 9/11 EP Guidance Revisions

New Reactors

- Early Site Permit Reviews
- Updated Guidance and Inspection Program
- Standard Design Certification
- Rulemaking for Part 50/52

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Thank you for your support!

18

goes w/#10

Assessment of the Quality of NRC Research Projects by the Advisory Committee on Reactor Safeguards

Steam Generator Tube Integrity and Integrity Predictions

Job Code No. Y–6588, Rev 3, ANL - Task 3

September, 2005

U.S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards
Washington, DC 20555-0001

ABSTRACT

In this report, the Advisory Committee on Reactor Safeguards (ACRS) presents the results of its assessment of the quality of one of the selected research projects sponsored by the Office of Nuclear Regulatory Research (RES) of the NRC. The project selected and evaluated was Job Code No. Y-6588, Task 3 of the Steam Generator Tube Integrity Program, performed by NRC RES and Argonne National Laboratory (ANL).

An analytic/deliberative methodology was adopted by the Committee to guide its review of research projects. The methods of multi-attribute utility theory were utilized to structure the objectives of the review and develop numerical scales for rating the project with respect to each objective.

The results of our evaluation of the quality of this research project are that the performance of this research project is that this research project is professional work that satisfies research objectives.

1.0 METHODOLOGY FOR EVALUATING THE QUALITY OF RESEARCH PROJECTS

To guide its review of research projects, the ACRS has adopted an analytic/deliberative methodology [Ref. 1]. The analytical part utilizes methods of multi-attribute utility theory (MAUT) to structure the objectives of the review and develop numerical scales for rating the project with respect to each objective. The objectives were developed in a hierarchical manner (in the form of a "value tree") and weights reflecting their relative importance were developed. The value tree and the relative weights developed by the full Committee are shown in Figure 1.

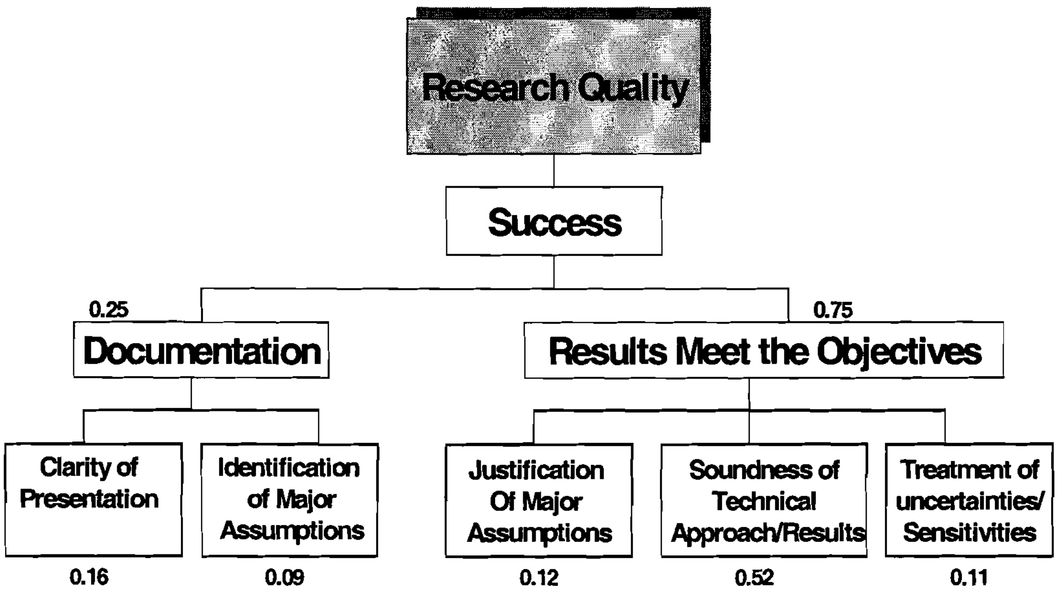

Figure 1 The value tree used for evaluating the quality of research projects

The quality of projects is evaluated in terms of the degree to which the results meet the objectives of the research and of the adequacy of the documentation of the research. It is the consensus of the ACRS that meeting the objectives of the research should have a weight of 0.75 in the overall evaluation of the research project. Adequacy of the documentation was assigned a weight of 0.25. Within these two broad categories, research projects were evaluated in terms of subsidiary "performance measures":

- Justification of major assumptions (weight: 0.12)
- Soundness of the technical approach and reliability of results (weight: 0.52)
- Treatment of uncertainties and characterization of sensitivities (weight: 0.1)

Documentation of the research was evaluated in terms of the following performance measures:

- Clarity of presentation (weight: 0.16)
- Identification of major assumptions (weight: 0.09)

To evaluate how well the research project performed with respect to each performance measure, constructed scales were developed as shown in Table 1. The starting point is a rating of 5, Satisfactory (professional work that satisfies the research objectives). Often in evaluations of this nature, a grade that is less than excellent is interpreted as pejorative. In this ACRS evaluation, a grade of 5 should be interpreted literally as satisfactory. Although innovation and excellent work are to be encouraged, The ACRS realizes that time and cost place constraints on innovation. Furthermore, research projects are constrained by the work scope that has been agreed upon. The score was, then, increased or decreased according to the attributes shown in the table. Multiplying each score by the corresponding weight of the performance measure and adding all the weighted scores produce the overall score of the project.

The value tree, weights, and constructed scales were the result of extensive deliberations of the whole ACRS. A panel of three ACRS members was formed to review each selected research project. Each member of the review panel independently evaluated the project in terms of the performance measures shown in the value tree. The panel deliberated the assigned scores and developed a consensus score, which was not necessarily the arithmetic average of individual scores. The panel's consensus score was discussed by the full Committee and adjusted in response to ACRS members' comments. The final consensus scores were multiplied by the appropriate weights, the weighted scores of all the categories were summed and an overall score for the project was produced. A set of comments justifying the ratings was also produced.

Table 1. Constructed Scales for the Performance Measures

SCORE	LABEL	INTERPRETATION
10	Outstanding	Creative and uniformly excellent
8	Excellent	Important elements of innovation or insight
5	Satisfactory	Professional work that satisfies research objectives
3	Marginal	Some deficiencies identified; marginally satisfies research objectives
0	Unacceptable	Results do not satisfy the objectives or are not reliable

2.0 Background

The objectives of task 3 were to answer four basic questions:

- Determine if the flow stress of MA Nickel Alloy 600 tube material exhibits dependence on the stress rate or the strain rate (i.e.: the rate of internal pressurization).
- Determine the relationship between crack or ligament size (width, depth and length), orientation, geometry, morphology, and number of ligaments and the tube leak rate and burst pressure.
- Confirm the validation of the tube leak rate correlation model and its relevance to choked two-phase flow expected at operating temperatures and pressures, including the relative uncertainties involved under various conditions.
- Compare laboratory leak rate and burst pressure models with the results of tests of samples of defective steam generator tubes removed from a decommissioned steam generator from McGuire Nuclear Plant.

To satisfy these primary objectives, the following test plan was developed by ANL and approved by NRC RES.

3.0 Scope and Milestones of Task 3

Task 3 of Job Code No. Y-6588 (Ref 2) consists of 11 subtasks, as listed below.

1. Based on experimental results and analyses from the previous steam generator tube integrity program at ANL, and augmented with additional testing and analyses, define the ligament size for various multiple crack orientations that will lead to crack interaction (joining to produce a single long crack) under normal operating pressure differential, and under MSLB pressure as a function of length and depth of individual initial cracks.
2. Evaluate the simple orifice model developed in the earlier program for producing leak rates of cracked tubes and define the limits of applicability with respect to through-wall crack length and crack tightness.
3. Perform pressure and leak tests of cracked tubes removed from the McGuire plant identified in Task 1. Eddy current tests and pressure and leak tests of these tubes will be conducted consistent with industry practice so that the results can be used to augment the existing data base and to evaluate industry models.
4. Evaluate whether high pressurization rates increase the burst strength of cracked SG tubes. Previous work has shown the potential for this phenomenon to occur. It seems to depend on SCC geometry and presence of ligaments. Plan tests at various pressurization rates on specimens with notches of controlled sizes, geometry (rectangular, trapezoidal, and triangular) and numbers of cracks and ligaments. Evaluate these results against tests conducted on stress corrosion cracks of various morphologies (planar, complex and ligamented). When possible, the pressure tests should be conducted without using bladders or foils.
5. Prepare a topical report providing results of these four subtasks immediately above. (Ref. xx)
6. Use information from existing analyses and from new results from RES thermo-hydraulic calculations and sensitivity studies during a MSLB and other secondary-side depressurization events to estimate upper-bound loads, cycles, and displacements on support structures and tubes.
7. Estimate crack growth, if any, for a range of crack types and sizes using bounding loads and displacements in addition to the pressure stresses. Include also any effects from cyclic loads.
8. Estimate the margins for crack propagation for a range of crack sizes for MSLB and other secondary-side depressurization loads and displacements in addition to the pressure stress.
9. Based on the margins calculated over and above the bounding loads, decide if more refined thermo-hydraulic analyses are required to obtain the forces and displacements of structures under MSLB and other secondary-side depressurization conditions.
10. Prepare a topical report on the results from the four subtasks immediately above (Ref 3) evaluating the potential for growth of preexisting cracks in steam generator tubes under MSLB and other secondary-side depressurization accidents.
11. Perform tests of flawed steam generator tubes to validate the improved methodologies for predicting failure pressures and leak rates of tubes with complex morphology cracks. Tests will include complex machined notches, as well as lab-produced

4.0 Evaluation of the Quality of Task 3

To evaluate Task 3 performance and results, we reviewed two ANL draft reports (References 3 and 4).

4.1 **Objectives of the research**

4,1,1 Soundness of the technical approach and reliability of results (weight: 0.52)

The scope of work defined in Ref. 2 was thorough in identifying the major steps and the technical approach to be used by the investigators in Task 3. The investigators used sound scientific and engineering methods to conduct these investigations. In addition, it is clear that the investigators followed up on anomalies and results that differed from prior assumptions to gain insights into the phenomenon that they were investigating. These new insights were factored into the analytical predictive models under development to the extent that they could be and uncertainties were estimated for data that had a range of numerical results. The investigators stated that the models provided conservative predictions.

Pressurization rate effects

The first reported task is the confirmation of claims that rupture of flawed tubes is dependent on the rate of pressurization. The approach undertaken is to test a variety of flawed tubes similar to those used by investigators making the claim of a pressurization rate effect. The testing is, however, done in a consistent fashion unlike the testing done by those making the claims. Testing was done at pressurization rates that varied from quasi-static to greater than 69 Mpa/s. This range includes, apparently, the pressurization rate used by those making the claims of a pressurization rate effect. Whether it includes prototypic pressurization rates is not stated, but it appears likely that it does. Tests were done at enough pressurization rates that it should be possible to infer by interpolation results for any pressurization rate likely to be of practical interest. This appears to be a technically sound defensible approach. In addition, tests are planned on cracks that were formed by a stress corrosion cracking process. Results of these tests presumably will be used to relate results of tests with machined flaws to more realistic cracks. Again, this seems a prudent and reasonable approach.

Development of failure "maps"

To prepare failure maps the authors have correlated data on the ligament ruptures of two types of flaws in tubes. The correlation model is a simple polynomial and does not seem to have been selected based on deep considerations of theory. Details of the fitting procedure are not spelled out to any extent. It is apparent that the polynomial is a very approximate description of the data and the parametric values must be changed for different crack lengths. Fitting neglected apparently the uncertainties in the data. Had these uncertainties been recognized, it might have been possible to use simpler correlation expressions. A similar polynomial correlation was developed for rupture pressure for the case of two cracks separated by a circumferential ligament.

It appears that the data used for correlation may have come from room temperature tests, but documentation is not definitive on this point and salient references have not yet been retrieved.

The correlations were then used to develop maps of crack length versus ligament width showing behavior for various pressure differences and crack geometries assuming cracks were 80 and 90% through wall. This approach is common and technically sound for maps involving two cracks separated by an axial or a radial ligament provided, of course, that the correlations developed from test data are applicable at the assumed 300°C.

Maps were also prepared for cases with four cracks and six cracks. There seems to be no demonstration that the correlations of ligament rupture and tube rupture obtained for two cracks are applicable to cases with four or more cracks. To be sure there is an extrapolation taking place here that is not especially well highlighted in the documentation. Nevertheless, one must concede that if this extrapolation is palatable, the approach adopted in preparing the maps is widely accepted one. Use of the maps, on the other hand, would demand a great deal more than is attempted in this limited effort. A reader is left hungering for some comparison of the map predictions for the multiple crack cases to data.

Leak Rate Studies

The leak rate studies were undertaken to determine the limits of applicability with respect to the through wall crack length and crack tightness of the simple orifice model for predicting leak rates of cracked tubes. The effort undertaken focused on conditions that will lead to "flashing" of the coolant within the crack. Crack length divided by the hydraulic diameter of the crack was used as the metric for cracks in tubes used in the tests. This is acceptable because realistic cracks are used in the test program. Analysis of the results was supplemented by data from the literature concerning flow through better instrumented slits in plates. The technical approach appears then adequate to the task.

Results obtained in the effort only address conditions for sub cooling in the range of 50-60°C . Such a sub cooling range corresponds to cold leg conditions. A plausibility argument is advanced that "conservative" results will be predicted for hot leg conditions that are more appropriate for issues associated with steam generator tube leakage. Thus, results only marginally meet the objective if the objective is to find limits of applicability of the orifice model for conditions where it is likely to be of interest to apply.

Rupture and Leak Rate Predictions for McGuire Steam Generator Tubes

The technical approach for this effort involved acquisition of flawed tubes form the McGuire facility and characterization of the flaws first by nondestructive methods and later by fractography. The tubes were then tested for leaking in a facility that is presumably well established and well described in some other publication. Unfortunately, no reference was provided to validate this presumption. No description of the method for measuring leak rates was provided. Presumably a well established method exists and the authors could have informed the reader about this method by means of a reference. Though poorly documented, the technical approach appears sound.

Overall assessment of the soundness of the Technical Approach

Though quibbles abound in the review of the technical approach there were no flaws identified that would detract from the value of the results in any major way. On the other hand, the technical approaches adopted in the four efforts were not inspired so no bases for higher scores were identified either. With regard to the attribute of soundness of the technical approach and reliability of results, the ACRS panel gave this attribute a score of _5.0_.

4.1.2 Justification of major assumptions (weight: 0.12)

In the statement of Scope and Milestones, there are certain assumptions implicit in the statement of scope. However the design of the work plan and scope was such that the major assumptions would be tested by experiment to verify the validity of these assumptions. An example was the assumption that flow stress is virtually independent of the rate at which stress and strain are applied to the specimen. This assumption had its origins in earlier test work performed by others prior to the in-depth study undertaken by ANL in Task 3. ANL could not confirm the validity of this assumption and undertook to determine why a rate effect was observed in their tests and not in the earlier tests. Other examples of implicit assumptions involved issues such as ligament linkage and its relationship to both leakage and burst pressure, the quantification of choke flow leakage through cracks with two-phase flow, and the existence of a correlation between leakage and crack growth,

The documentation is as is discussed further below exceptionally informal. The investigators do not make an explicit effort to identify assumptions and justify these assumptions. In many respects there are no assumptions simply because it is not evident how results obtained in the work will be used beyond the particular measurement activity described. That is, there is no implication that the results reported in Ref. 3 will be applied to real tubes under real accident conditions. There are occasions where assumptions arise. For example, in connection with the development of failure 'maps' it is asserted that the complex ligament geometries of real cracks can be idealized as either purely axial, purely circumferential or radial. The report does not go on to argue that this manifestly possible idealization is at all a meaningful representation of reality. As noted above, there is a further assumption about application of correlation developed for two cracks being applicable to configurations with four and six cracks that is neither articulated nor justified.

In some cases, the assumed level of familiarity with previous work limits the discussion to the extent that the bases for assumptions are not clear. For example, in the predictions of ligament rupture against McGuire tests, the ligament rupture pressure of each test was predicted by the equivalent rectangular crack methods. It is not explained why this is the appropriate model, which would be worthwhile given that the benchmark is only partially successful. The abstract states that this is the "latest correlation". But some additional explanation would have contributed to a better understanding.

Much of the work on MSLB effects on damaged tubes (Ref.4) rely on analytical simulation with TRAC-M and RELAP-5. The ability of these codes to model appropriately pressure drops is complex geometries such as those of SG tube bundles and TSPs has been questioned. The report does not discuss this issue. There are good comparisons of results from the two codes and FEA results too, but applicability of these models is an important issue that deserved some discussion.

Overall, the report can be critiqued for not explicitly addressing assumptions in a defined part of the report. But, the context of the report suggests that most readers and the sponsors of the work have a general understanding of the assumptions and these assumptions do not detract from the use of the results.

With regard to the attribute of justification of major assumptions, the ACRS panel gave this attribute a score of __4.7__.

4.1.3 Treatment of uncertainties and characterization of sensitivities (weight: 0.11)

The comparison of predictive models of leak rate and rupture as applied to actual tubes removed from a retired McGuire steam generator with leakage and burst test data of these tubes showed reasonable agreement. In the discussion, explanations as to why the predictive models differed from the actual test results were suggested. A range of uncertainty and the degree of conservatism between the models and observed results was estimated, in order to establish the degree of usefulness of the correlations developed. Because of the complex nature of SCC cracks, predictive uncertainty exists and has been estimated and factored into the resulting conclusions.

The investigators do a rather good job in developing their experimental projects in considering sensitivities such as sensitivity to the number of cracks, ligament sizes, crack orientation and the like. The investigators have consistently refused to estimate a single uncertainty associated with any measured value that they report. Where they have fit data to a parametric correlation, they have failed to cite any uncertainties in the parametric values and certainly have not reported covariance matrices for models involving more than two parameters. They do not report on the uncertainties of predictions derived from correlations. Episodically the authors report linear correlation coefficients that are essentially useless in the interpretation of the quality of a fit of a parameterized equation to data without a great deal more information about the fitting results.

The adequacy of the authors' treatments of sensitivities in the development of their research efforts is acknowledged. Neglect of uncertainties in reports of measurements is the basis for reduction of the score in this category.

With regard to the attribute of treatment of uncertainties and characterization of sensitivities, the ACRS panel gave this attribute a score of __4.3__.

5.0 Documentation of the research

5.1 Clarity of presentation (weight: 0.16)

The manuscripts Ref. 3 and Ref. 4 are exceptionally informal. The documents read like laboratory reports prepared by technicians and sent to professional staff to be used in the preparation of a more formal report. Both manuscripts are rather more summary in nature. This terse informality of documentation makes the reports more readable. The reports are inadequate for the archival documentation of expensive tests. Experimental methods are mentioned in casual ways with no effort even by reference to show that these methods are adequate or produce reliable, reproducible results. Calibration and qualification of instruments is not discussed at all. Theoretical models and even data analysis methods are mentioned without reference. Figures showing data and correlations are exceptionally difficult to interpret since minimal legends and labeling are employed despite the figures being quite "busy". The leak rate studies (page 34 of Ref. 3), but for specimen SLG900, no results are provided. The discussion on page 44 is not clear when correlating L/D ratios and choked flow. A reader who does not routinely examine reports from this laboratory and is not intimately familiar with the equipment and methods of the laboratory struggles to understand the documentation. (Only after reading Ref 4 did I come to understand that the unlabeled scale in some photos in Ref. 3 was an inch scale and not a centimeter scale despite all the text on lengths referring to millimeters!) In the end, one can understand the points the authors are trying to make in manuscript Ref. 3, but it is with difficulty. Clarity of presentation is not high, but it is adequate for the work to be understood. It is dubious that the experimental results could ever be used directly in a regulatory process involving licensees. The qualification of methods and calibration of instruments simply will not be acceptable for such direct use.

As noted above, the manuscript Ref 4 is simply too crude to be readily reviewed and evaluated.

With regard to the attribute of clarity of presentation, the ACRS panel gave this attribute a score of 4.7 .

5.2 Identification of major assumptions (weight: 0.09)

The identification of the major assumptions employed is not separately and explicitly stated but some of these assumptions are embedded in the text. In a complex report such as this, it is an acceptable and appropriate practice to state assumptions in the context of the issues where they are used or evaluated and rejected.

As noted above, identification and justification of assumptions is a hard issue to evaluate. There is not a coherent effort to do this in the document largely because there is no sense that results have any applicability beyond the explicit measurement being made. That is, there is no sense that results for notched specimens discussed in the document will be used to infer the behavior of real cracks in tubes under accident conditions.

The investigators have done a better job identifying factors that will affect the experimental results and including these sensitivities in their test programs. The documentation does not go to any great lengths to justify the sensitivities that are included nor does it advance arguments concerning other factors that can be excluded. The document fails completely to address uncertainties in measurements or to provide adequate descriptions of parametric uncertainties in reporting results of fits of data to correlations. Presumably, if needed, these uncertainties as well as uncertainties in measurements could be extracted so only a modest reduction in the score has been imposed.

With regard to the attribute of Identification of major assumptions, the ACRS panel gave this attribute a score of __4.7__.

6.0 Overall rating of research project

Based on the evaluation of the listed attributes, the sub-committee, the consensus scores for this research project are as follows:

Table 2 - Summary Results of ACRS Assessment of the Quality of the Project on Task 3, Research on Tube Integrity and Integrity Predictions

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	4.7	0.16	0.752
Identification of major assumptions	4.7	0.09	0.423
Justification of major assumptions	4.7	0.12	0.564
Soundness of technical approach/results	5.0	0.52	2.600
Treatment of uncertainties/sensitivities	4.3	0.11	0.473
Overall Score:			4.812

Based on our evaluation, we consider this project to be professional work that satisfies research objectives and therefore is rated as satisfactory.

References:

1. Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards, dated November 2004.

1. Job Code No. Y-6588, Task 3 of the Steam Generator Tube Integrity Program, performed by NRC RES and Argonne National Laboratory (ANL).

2. "Pressurization Rate Effect on Flawed Tube Rupture, Failure Maps for Complex Multiple Cracks, Validation of Leak Rate Correlation Model and Rupture and Leak Rate Predictions for McGuire Steam Generator Tubes" by S. Majumdar, K. Kasza, Sasan Bakhtiari, J. Oras, J. Franklin, and C. Vulyak, Jr., dated April 2004.

4. "Sensitivity Studies of Failure of Steam Generator Tubes during Main Steam Line Break and Other Secondary Side Depressurization Events" by S. Majumdar, K. Kasza, J. Oras, J. Franklin and C. Vulyak, Jr., dated April 2000