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

October 2, 2007

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Klein:

SUBJECT: SUMMARY REPORT–545th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, SEPTEMBER 6-8, 2007, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 545th meeting, September 6-8, 2007, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letter, and memorandum:

REPORTS

Reports to Dale E. Klein, Chairman, NRC, from William J. Shack, Chairman, ACRS:

- Development of a Technology-Neutral Regulatory Framework, dated September 26, 2007.
- Report on the Safety Aspects of the License Renewal Application for the Pilgrim Nuclear Power Station, dated September 26, 2007.

LETTER

Letter to Luis A. Reyes, Executive Director for Operations, NRC, from William J. Shack, Chairman, ACRS:

- Proposed Recommendation for Resolving Generic Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment," September 26, 2007.

MEMORANDUM

Memorandum to Luis A. Reyes, Executive Director for Operations, NRC, from Frank P. Gillespie, Executive Director, ACRS:

- Draft Final Amendment to 10 CFR 50.55a, "Codes and Standards," and Revisions to Regulatory Guides Regarding ASME Code Cases, dated September 13, 2007.

HIGHLIGHTS OF KEY ISSUES

1. Final Review of the License Renewal Application for the Pilgrim Nuclear Power Station

The Committee met with the representatives of the Entergy Nuclear Operations, Inc. (Entergy, the applicant) and the NRC staff to discuss the license renewal application for the Pilgrim Nuclear Power Station (PNPS) and the associated final Safety Evaluation Report (SER). The operating license for PNPS expires on June 8, 2012. The applicant has requested approval for continued operation for a period of 20 years beyond the current license expiration date. The applicant stated that Pilgrim does not have the same scoping issues identified as the Vermont Yankee (VY) license renewal application because different scoping methodologies were used for PNPS. The applicant described the resolution of the open items related to the containment inservice inspection program, neutron fluence, and intrusion of groundwater into the torus room, as well as the actions taken and the commitments made to resolve these issues.

The applicant addressed the open item associated with neutron fluence by committing to complete the benchmarking of the code used in the fluence calculation. The applicant committed to submit a correctly benchmarked fluence calculation to the NRC on or before June 8, 2010, to confirm that the limiting fluence value will not be reached during the period of the extended operation. The staff is making the applicant's commitment a license condition to ensure adequate resolution of this issue. Regarding groundwater intrusion into the torus room, the applicant committed to enhance the structures monitoring program and perform periodic testing of the water for aggressiveness to concrete.

The staff described its review and inspection of the applicant's scoping, screening, and aging management programs; the program implementation at PNPS; and resolution of the open items. The staff also confirmed that the applicant has committed to follow the Generic Aging Lessons Learned Report, without exceptions, regarding monitoring of the cumulative usage factor for environmentally assisted fatigue. The staff stated that it plans to issue a supplemental SER to document this commitment.

Committee Action

The Committee issued a report to the NRC Chairman on this matter, dated September 26, 2007. The Committee concluded that the license conditions proposed by the staff are appropriate and recommended that the application of Entergy Nuclear Operations, Inc., for renewal of the operating license for PNPS be approved with the proposed license conditions.

2. Revisions to Standard Review Plan (SRP) Sections 19.0 and 19.2

The Committee met with representatives of the NRC staff to discuss revisions to SRP Sections 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," and 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance."

SRP Section 19.0 is associated with Regulatory Guide 1.206 (Combined License Applications for Nuclear Power Plants) and provides guidance to NRC staff reviewers for evaluating the content of combined license (COL) applications. The staff stated that design certification (DC) and COL applicants are required to submit a "description of their PRA and its results." The staff outlined the scope and level of detail that a COL applicant's probabilistic risk assessment (PRA) must meet. The staff also summarized the requirements for updating and upgrading COL holders' PRAs. The staff elaborated on what the description of an applicant's PRA should include and what results the staff expects to see in an applicant's submittal. The staff mentioned that additional guidance in several areas related to PRA is needed and it plans to issue interim staff guidance (ISG) to convey this additional guidance to industry.

SRP Section 19.2 is associated with Revision 1 to Regulatory Guide 1.174 (An Approach for Using Probabilistic Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis) and provides guidance to NRC staff reviewers for evaluating risk-informed changes to a plant's licensing basis. A member expressed concern that the modeling of digital instrumentation and control (I&C) systems in PRAs may not be adequate because the failure modes of digital I&C systems are not well understood. Another member expressed concern that the word "large" as used in the expressions "large early release frequency" (LERF) and "large release frequency" (LRF) is not well defined.

Committee Action

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

3. Proposed Recommendation for Resolving Generic Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment"

The Committee met with representatives of the NRC staff to discuss the proposed recommendation for resolving Generic Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment." The staff described the history of the issue, its prioritization through the Generic Issues Program, the specific investigations performed for Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs), and the outcome of these analyses. This issue was relevant to reactors that were designed and licensed prior to the issuance of the first SRP. Investigations narrowed the focus of the possible effects of pipe whip and jet impingement inside containment to the possible breach of the containment shell in BWR Mark 1 plants and the possible failure of instrumentation and control systems in PWRs. More detailed and quantitative analyses showed that for the 51 plants that originally fell within the scope of this issue, the designs were satisfactory and that no further actions were required on the part of licensees. Therefore, staff is recommending that this issue be closed.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated September 26, 2007, concurring with the staff's recommendation that Generic Issue 156.6.1 be closed out and that no further actions on the part of the staff or licensees are necessary.

4. Status of NRR Activities in the Fire Protection Area

The Committee met with representatives of the NRR staff to discuss the ongoing NRC activities in the fire protection area. The staff described several major activities, including those associated with implementation of the National Fire Protection Association (NFPA) Standard 805, "Fire-Induced Multiple Spurious Actuations and Manual Operator Actions." The staff discussed the plants that are transitioning to the NFPA 805 Standard and the lessons learned, as well as the status of industry guidance development for fire modeling. The staff also provided its views on the Nuclear Energy Institute multiple spurious actuation methodology and recent interaction with industry in addressing this issue. In addition, the staff discussed post-fire manual operator actions and recent staff guidance for addressing this issue. The staff also provided an update on the Hemyc and MT fire barrier issue, and the industry progress in addressing Generic Letter 2006-03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations."

Committee Action

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

5. Subcommittee Report on Plant License Renewal

The Chairman of the Plant License Renewal Subcommittee provided a report to the Committee summarizing the results of the September 5, 2007, meeting with the NRC staff and representatives of Entergy to review the draft SER with Open Items related to the license renewal application for the James A. Fitzpatrick Nuclear Power Plant. The current operating license expires on October 17, 2014. Entergy submitted the license renewal application on July 31, 2006. The staff's draft SER was issued on July 31, 2006, and contains two open items and no confirmatory items. The two open items are related to reactor vessel neutron fluence and environmentally assisted fatigue. For determining reactor vessel neutron fluence, the staff finds that the projected fluence values are unacceptable. Entergy stated that it will submit a new fluence calculation to the staff for review. Entergy also stated that it will demonstrate that cumulative usage factors (CUF) of the most fatigue sensitive locations are less than 1.0 throughout the license renewal period, and it will submit the results of the CUF calculations to the staff for review and approval. Other discussion topics included drywell and torus monitoring, and torus repair.

Committee Action

The Committee plans to discuss the final SER related to the license renewal application for the James A. Fitzpatrick Nuclear Power Plant in a future meeting.

6. Draft Report on Quality Assessment of Selected NRC Research Projects

The Committee was briefed by the members of the ACRS panels regarding the results of their assessment of the quality of the NRC research projects on Cable Response to Live Fire (CAROLFIRE) Testing, Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping, and Technical Review of the Online Monitoring Techniques for Performance Assessment.

Committee Action

The Committee plans to complete a final report on the results of its assessment of the quality of the above NRC research projects during its October 2007 meeting.

7. Draft ACRS Report on the NRC Safety Research Program

The ACRS provides the Commission a biennial report that presents the Committee's observations and recommendations concerning the overall NRC Safety Research Program. During the September 2007 meeting, the Committee was briefed by the lead members of ACRS regarding the status of their evaluation of research activities in specific technical disciplines.

Committee Action

The Committee plans to continue its discussion of the draft ACRS report on the NRC Safety Research Program during its October 2007 meeting.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of August 20, 2007, to comments and conclusions included in the July 24, 2007, ACRS report concerning the staff's approach to verifying the closure of inspections, tests, analyses, and acceptance criteria (ITAAC) through a sample-based inspection program. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of July 30, 2007, to comments and recommendations included in the June 22, 2007, ACRS letter concerning General Electric (GE) Licensing Topical Reports on Maximum Extended Load Line Limit Analysis Plus (MELLLA+) and Applicability of GE Methods to Expanded Operating Domains. The Committee decided that it was satisfied with the EDO's response. In its response, the EDO committed to the following:

  - **The modifications identified during ACRS discussions associated with anticipated transients without scram (ATWS) instability will be included in the final Safety Evaluation prepared by the staff.**

  - **The Committee will be provided the opportunity to review the first few plant-specific MELLLA+ applications, and any significant changes in the final Safety Evaluation prepared by the staff, including any changes to the limitations. The Committee will be provided the opportunity to review any future significant changes to the limitations currently applied to the safety limit already defined in the minimum critical power ratio (MCPR), the operating limit MCPR, and on bypass voiding.**

- The Committee considered the EDO's response of August 29, 2007, to conclusions and recommendations included in the July 27, 2007, ACRS letter on draft NUREG/CR, titled, "Review of NUREG-0654, Supplement 3, 'Protective Action Recommendations for Severe Accidents'." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of July 23, 2007, to comments and recommendations included in the June 18, 2007, ACRS letter concerning the final draft NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of July 11, 2007, to comments and recommendations included in the May 23, 2007, ACRS letter regarding proposed technical basis for the revision to 10 CFR 50.46 LOCA embrittlement criteria for fuel cladding materials. The Committee decided to discuss this matter during a future meeting.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

The following Subcommittee meetings were held during the period from July 13, 2007, through September 5, 2007:

- Plant Operations – August 14 – 16, 2007

The Subcommittee visited NRC Region IV offices and the San Onofre Nuclear Generating Station (SONGS) to discuss plant operations issues.

- Planning and Procedures – September 5, 2007

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

- Plant License Renewal - September 5, 2007

The Subcommittee reviewed the license renewal application and the associated NRC staff's Safety Evaluation Report with Open Items for the James A. Fitzpatrick Nuclear Power Plant.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee would like the opportunity to review the AP1000 technical reports related to PRA (and associated draft safety evaluation) that will form the basis, in part, for anticipated amendment to the AP1000 certified design.

- The Committee plans to discuss the final draft report on the results of its assessment of the quality of the selected NRC research projects during its October 2007 meeting.

- The Committee plans to continue discussion on its draft report on the NRC Safety Research Program during its October 2007 meeting.

- The Committee plans to discuss the final SER related to the license renewal application for the James A. Fitzpatrick Nuclear Power Plant during a future meeting.

- The Committee plans to discuss the report on the proposed technical basis for the revision to 10 CFR 50.46 LOCA embrittlement criteria for fuel cladding materials during a future meeting.

PROPOSED SCHEDULE FOR THE 546th ACRS MEETING

The Committee agreed to consider the following topics during the 546th ACRS meeting, to be held on October 4-6, 2007:

- Digital I&C Project Plan and Interim Staff Guidance
- Draft final Generic Letter 2007-XX, "Managing Gas Intrusion in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems"
- Dissimilar Metal Weld Issue
- Draft ACRS Report on the NRC Safety Research Program
- Draft final Report on Quality Assessment of Selected NRC Research Projects
- Meeting with NEI, EPRI, and INPO to discuss Industry Activities

Sincerely,

William J. Shack

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/RA/

William J. Shack

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

October 29, 2007

MEMORANDUM TO: Carol A. Brown, Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: William J. Shack
ACRS Chairman

SUBJECT: MINUTES OF THE 545th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), September 6-8, 2007

I certify that based on my review of the minutes from the 545th 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.

NA
Comments

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NAME	JFlack		
DATE	10/29/2007		

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NAME	JFlack		
DATE	10/29/2007		

CERTIFIED

Date Issued:
Date Certified:

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MINUTES OF THE **545th** ACRS MEETING

September 6-8, 2007

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1. Development of a Technology-Neutral Regulatory Framework, dated September 26, 2007.

2. Report on the Safety Aspects of the License Renewal Application for the Pilgrim Nuclear Power Station, dated September 26, 2007.

LETTER:

The following letter to Luis A. Reyes, Executive Director for Operations, NRC, from William J. Shack, Chairman, ACRS:

1. Proposed Recommendation for Resolving Generic Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment," September 26, 2007.

MEMORANDUM:

The following memorandum to Luis A. Reyes, Executive Director for Operations, NRC, from, Frank P. Gillespie, Executive Director, ACRS:

1. Draft Final Amendment to 10 CFR 50.55a, "Codes and Standards," and Revisions to Regulatory Guides Regarding ASME Code Cases, dated September 13, 2007.

APPENDICES

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

MINUTES OF THE 545th MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
September 6-8, 2007
ROCKVILLE, MARYLAND

The **545th** meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on **September 6-8, 2007**. Notice of this meeting was published in the *Federal Register* on **August 14, 2007** (72 FR 45452) (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.

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: Dr. William J. Shack (Chairman), Dr. Mario V. Bonaca (Vice-Chairman), Dr. Said Abdel-Khalik (Member-at-Large), Dr. George E. Apostolakis, Dr. Sam Armijo, Dr. Michael Corradini, Mr. Otto L. Maynard, Dr. Dana A. Powers and Mr. John Stetkar. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

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

Dr. William J. Shack, Committee Chairman, convened the meeting at 8:30 A.M. He announced in his opening remarks that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. In addition, he reviewed the agenda for the meeting and noted that no written comments or requests for time to make oral statements from members of the public had been received. Dr. Shack also noted that a transcript of the open portions of the meeting was being kept and speakers were requested to identify themselves and speak with clarity and volume. He discussed the items of current interest and administrative details for consideration by the full Committee.

II. Final Review of the License Renewal Application for the Pilgrim Nuclear Power Station

[Note: Ms. Maitri Banerjee was the Designated Federal Official for this portion of the meeting.]

The Committee met with the representatives of the Entergy Nuclear Operations, Inc. (Entergy, the applicant) and the NRC staff to discuss the license renewal application for the Pilgrim Nuclear Power Station (PNPS) and the associated final Safety Evaluation Report (SER). The operating license for PNPS expires on June 8, 2012. The applicant has requested approval for continued operation for a period of 20 years beyond the current license expiration date. The applicant stated that Pilgrim does not have the same scoping issues identified as the Vermont Yankee (VY) license renewal application because different scoping methodologies were used for PNPS. The applicant described the resolution of the open items related to the containment inservice inspection program, neutron fluence, and intrusion of groundwater into the torus room, as well as the actions taken and the commitments made to resolve these issues.

The applicant addressed the open item associated with neutron fluence by committing to complete the benchmarking of the code used in the fluence calculation. The applicant committed to submit a correctly benchmarked fluence calculation to the NRC on or before June 8, 2010, to confirm that the limiting fluence value will not be reached during the period of the extended operation. The staff is making the applicant's commitment a license condition to ensure adequate resolution of this issue. Regarding groundwater intrusion into the torus room, the applicant committed to enhance the structures monitoring program and perform periodic testing of the water for aggressiveness to concrete.

The staff described its review and inspection of the applicant's scoping, screening, and aging management programs; the program implementation at PNPS; and resolution of the open items. The staff also confirmed that the applicant has committed to follow the Generic Aging Lessons Learned Report, without exceptions, regarding monitoring of the cumulative usage factor for environmentally assisted fatigue. The staff stated that it plans to issue a supplemental SER to document this commitment.

III. <u>Revisions to Standard Review Plan (SRP) Sections 19.0 and 19.2</u>

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

The Committee met with representatives of the NRC staff to discuss revisions to SRP Sections 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," and 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance."

SRP Section 19.0 is associated with Regulatory Guide 1.206 (Combined License Applications for Nuclear Power Plants) and provides guidance to NRC staff reviewers for evaluating the content of combined license (COL) applications. The staff stated that design certification (DC) and COL applicants are required to submit a "description of their PRA and its results." The staff outlined the scope and level of detail that a COL applicant's probabilistic risk assessment (PRA) must meet. The staff also summarized the requirements for updating and upgrading COL holders' PRAs. The staff elaborated on what the description of an applicant's PRA should include and what results the staff expects to see in an applicant's submittal. The staff mentioned that additional guidance in several areas related to PRA is needed and it plans to issue interim staff guidance (ISG) to convey this additional guidance to industry.

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IV. Proposed Recommendation for Resolving Generic Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment"

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

The Committee met with representatives of the NRC staff to discuss the proposed recommendation for resolving Generic Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment." The staff described the history of the issue, its prioritization through the Generic Issues Program, the specific investigations performed for Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs), and the outcome of these analyses. This issue was relevant to reactors that were designed and licensed prior to the issuance of the first SRP. Investigations narrowed the focus of the possible effects of pipe whip and jet impingement inside containment to the possible breach of the containment shell in BWR Mark 1 plants and the possible failure of instrumentation and control systems in PWRs. More detailed and quantitative analyses showed that for the 51 plants that originally fell within the scope of this issue, the designs were satisfactory and that no further actions were required on

V. Status of NRR Activities in the Fire Protection Area

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

The Committee met with representatives of the NRR staff to discuss the ongoing NRC activities in the fire protection area. The staff described several major activities, including those associated with implementation of the National Fire Protection Association (NFPA) Standard 805, "Fire-Induced Multiple Spurious Actuations and Manual Operator Actions." The staff discussed the plants that are transitioning to the NFPA 805 Standard and the lessons learned, as well as the status of industry guidance development for fire modeling. The staff also provided its views on the Nuclear Energy Institute multiple spurious actuation methodology and recent interaction with industry in addressing this issue. In addition, the staff discussed post-fire manual operator actions and recent staff guidance for addressing this issue. The staff also provided an update on the Hemyc and MT fire barrier issue, and the industry progress in addressing Generic Letter 2006-03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations."

VI. Subcommittee Report on Plant License Renewal

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

The Chairman of the Plant License Renewal Subcommittee provided a report to the Committee summarizing the results of the September 5, 2007, meeting with the NRC staff and representatives of Entergy to review the draft SER with Open Items related to the license renewal application for the James A. Fitzpatrick Nuclear Power Plant. The current operating license expires on October 17, 2014. Entergy submitted the license renewal application on July 31, 2006. The staff's draft SER was issued on July 31, 2006, and contains two open items and no confirmatory items. The two open items are related to reactor vessel neutron fluence and environmentally assisted fatigue. For determining reactor vessel neutron fluence, the staff finds that the projected fluence values are unacceptable. Entergy stated that it will submit a new fluence calculation to the staff for review. Entergy also stated that it will demonstrate that cumulative usage factors (CUF) of the most fatigue sensitive locations are less than 1.0 throughout the license renewal period, and it will submit the results of the CUF calculations to the staff for review and approval. Other discussion topics included drywell and torus monitoring, and torus repair.

VII. Draft Report on Quality Assessment of Selected NRC Research Projects

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

The Committee was briefed by the members of the ACRS panels regarding the results of their assessment of the quality of the NRC research projects on Cable Response to Live Fire (CAROLFIRE) Testing, Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping, and Technical Review of the Online Monitoring Techniques for Performance Assessment.

VIII. Draft ACRS Report on the NRC Safety Research Program

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

The ACRS provides the Commission a biennial report that presents the Committee's observations and recommendations concerning the overall NRC Safety Research Program. During the September 2007 meeting, the Committee was briefed by the lead members of ACRS regarding the status of their evaluation of research activities in specific technical disciplines.

IX. Executive Session (Open)

[Note: Mr. Frank P. Gillespie was the Designated Federal Official for this portion of the meeting.]

A. RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

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

- The Committee considered the EDO's response of August 20, 2007, to comments and conclusions included in the July 24, 2007, ACRS report concerning the staff's approach to verifying the closure of inspections, tests, analyses, and acceptance criteria (ITAAC) through a sample-based inspection program. The Committee decided that it was satisfied with the EDO's response.

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  - **The modifications identified during ACRS discussions associated with anticipated transients without scram (ATWS) instability will be included in the final Safety Evaluation prepared by the staff.**

  - **The Committee will be provided the opportunity to review the first few plant-specific MELLLA+ applications, and any significant changes in the final Safety Evaluation prepared by the staff, including any changes to the limitations. The Committee will be provided the opportunity to review any future significant changes to the limitations currently applied to the safety limit already defined in the minimum critical power ratio (MCPR), the operating limit MCPR, and on bypass voiding.**

- The Committee considered the EDO's response of August 29, 2007, to conclusions and recommendations included in the July 27, 2007, ACRS letter on draft NUREG/CR, titled, "Review of NUREG-0654, Supplement 3, 'Protective Action Recommendations for Severe Accidents'." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of July 23, 2007, to comments and recommendations included in the June 18, 2007, ACRS letter concerning the final draft NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of July 11, 2007, to comments and recommendations included in the May 23, 2007, ACRS letter regarding proposed technical basis for the revision to 10 CFR 50.46 LOCA embrittlement criteria for fuel cladding materials. The Committee decided to discuss this matter during a future meeting.

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

The following Subcommittee meetings were held during the period from July 13, 2007, through September 5, 2007:

- <u>Plant Operations</u> – August 14 – 16, 2007

The Subcommittee visited NRC Region IV offices and the San Onofre Nuclear Generating Station (SONGS) to discuss plant operations issues.

- Planning and Procedures — September 5, 2007

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

- Plant License Renewal - September 5, 2007

The Subcommittee reviewed the license renewal application and the associated NRC staff's Safety Evaluation Report with Open Items for the James A. Fitzpatrick Nuclear Power Plant.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee would like the opportunity to review the AP1000 technical reports related to PRA (and associated draft safety evaluation) that will form the basis, in part, for anticipated amendment to the AP1000 certified design.

- The Committee plans to discuss the final draft report on the results of its assessment of the quality of the selected NRC research projects during its October 2007 meeting.

- The Committee plans to continue discussion on its draft report on the NRC Safety Research Program during its October 2007 meeting.

- The Committee plans to discuss the final SER related to the license renewal application for the James A. Fitzpatrick Nuclear Power Plant during a future meeting.

- The Committee plans to discuss the report on the proposed technical basis for the revision to 10 CFR 50.46 LOCA embrittlement criteria for fuel cladding materials during a future meeting.

PROPOSED SCHEDULE FOR THE 546th ACRS MEETING

The Committee agreed to consider the following topics during the 546th ACRS meeting, to be held on October 4-6, 2007:

- Digital I&C Project Plan and Interim Staff Guidance

- Draft final Generic Letter 2007-XX, "Managing Gas Intrusion in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems"

- Dissimilar Metal Weld Issue

- Draft ACRS Report on the NRC Safety Research Program

- Draft final Report on Quality Assessment of Selected NRC Research Projects

- Meeting with NEI, EPRI, and INPO to discuss Industry Activities

B. Report on the Meeting of the Planning and Procedures Subcommittee Held on September 5, 2007

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through December 2007 was discussed. The objectives are to:

Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
Manage the members' workload for these meetings
Plan and schedule items for ACRS discussion of topical and emerging issues

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

Operating Plan, Self-Assessment, and Letter Matrix

The ACRS staff is in the process of preparing the ACRS/ ACNW&M Operating Plan for 2008. This is in three parts, 2008 operations, resources, and annual self-assessment. Contained within the annual self-assessment is the traditional letter matrix. The current due date to the Commission is November 1, 2007. An early draft was provided to the Planning and Procedures Subcommittee members on September 5, 2007 for information and comment as appropriate. A draft was sent to all ACRS members on September 28, 2007. The information is similar to last year's plan reformatted to eliminate material wherever possible.

Quadripartite Working Group Meeting

France's Groupe Permanent Réacteurs (GPR) will host the second Quadripartite Working Group (WG) meeting in France on the general topic of "EPR". The proposed dates are as follows:

October 9-10, 2008 OR
October 16-17, 2008 OR
October 23-24, 2008

GPR is asking for specific items/topics that the Committee would like to discuss at this WG meeting. Dr. Powers, Chairman of the EPR Subcommittee, proposes the following topics:

PRA
Digital I&C
Fire Risk
Quality Assurance

In addition, Dr. Powers recommends that the Committee authorize him, Dr. Bonaca, and Mr. Stetkar to attend this WG meeting.

Proposed ACRS Meeting Dates for CY 2008

Proposed ACRS meeting dates from CY 2008 are summarized below. This was provided to the members during the September meeting for comment. We have received no comments.

Meeting No.	Dates
***	January 2008 (No Meeting)
549	February 7 – 9, 2008
550	March 6 - 8, 2008
551	April 3 - 5, 2008
552	May 8 – 10, 2008
553	June 4 – 6, 2008 (Wed – Fri)
554	July 9 – 11, 2008 (Wed – Fri)
***	August, (No Meeting)
555	September 4 – 6, 2008
556	October 2 – 4, 2008
557	November 6 – 8, 2008
558	December 4 – 6, 2008

Proposed List of Research Projects for Quality Assessment in FY 2008

A list of research projects proposed by RES for quality assessment in FY 2008 was discussed. In view of the anticipated heavy workload, the Committee should select a maximum of 2 topics for quality assessment. Dr. Powers has selected the following two projects and an alternate:

FRAPCON/FRAPTRAN Code Work at PNNL (Dr. Powers, Panel Chair)
NUREG-6943, "Study of Remote Visual Methods to Detect Cracking in Reactor Components" (Dr. Armijo, Panel Chair)

Alternate: Baseline Risk Index for Initiating Events (BRIE) as documented in NUREG/CR 6932, June 2007.

Proposed Assignments for Reviewing Revisions to Regulatory Guides

During the September 2007 ACRS meeting, the Committee was informed of the RES staff's plan to update, as necessary, all NRC Regulatory Guides by December 2009. These updates will be performed in three phases:

- Phase 1, involving revisions to Regulatory Guides applicable to future plant licensing, was completed in March 2007.
- Phases 2 and 3 Regulatory Guides updates will be completed in December 2008 and December 2009, respectively.

At the September meeting, the ACRS staff committed to provide a list of proposed assignments for reviewing Phase 2 Regulatory Guides for consideration by the Subcommittee and the full Committee during their October meetings. These assignments may be changed, as needed, to balance the workload among the members.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the **546th** ACRS Meeting, **October 4-6, 2007.**

The **545th** ACRS meeting was adjourned at **12:00 PM, September 8, 2007.**

and (a)(2)(B)(II.B.) (shift in production to a foreign country) have not been met.

TA–W–61,687; The GSI Group, Inc., Vandalia, IL.

The investigation revealed that criteria (a)(2)(A)(I.C.) (increased imports) and (a)(2)(B)(II.B.) (shift in production to a foreign country) have not been met.

TA–W–61,742; Sypris Technologies, Inc., A Subsidiary of Sypris Solutions, Kenton, OH.

TA–W–61,845; NYC American, Inc., Brooklyn, NY.

The workers' firm does not produce an article as required for certification under Section 222 of the Trade Act of 1974.

TA–W–61,662; Metso Paper USA, Inc., Roll Service Shop, Appleton, WI.

TA–W–61,778; Integrated Brands, Inc., Divisional Coolbrands International, Ronkonkoma, NY.

TA–W–61,790; State Farm Insurance, Regional Claims Office, Wheelersburg, OH.

The investigation revealed that criteria of Section 222(b)(2) has not been met. The workers' firm (or subdivision) is not a supplier to or a downstream producer for a firm whose workers were certified eligible to apply for TAA.

None.

I hereby certify that the aforementioned determinations were issued during the period of July 30 through August 3, 2007. Copies of these determinations are available for inspection in Room C–5311, U.S. Department of Labor, 200 Constitution Avenue, NW., Washington, DC 20210 during normal business hours or will be mailed to persons who write to the above address.

Dated: August 8, 2007.

Linda G. Poole,

Certifying Officer, Division of Trade Adjustment Assistance.

[FR Doc. E7–15848 Filed 8–13–07; 8:45 am]

BILLING CODE 4510–FN–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 6–8, 2007, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Wednesday, November 15, 2006 (71 FR 66561).

Thursday, September 6, 2007, Conference Room T–2b3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10:30 a.m.: Final Review of the License Renewal application for the Pilgrim Nuclear Power Station (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Entergy Nuclear Operations, Inc. regarding the license renewal application for the Pilgrim Nuclear Power Station and the associated NRC staff's final Safety Evaluation Report.

10:45 a.m.–12:15 p.m.: Revisions to Standard Review Plan (SRP) Sections 19.0 and 19.2 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding revisions to SRP Sections 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," and 19.2, "Review of Risk Information Used to Support Permanent Plant Specific Changes to the Licensing Basis: General Guidance."

1:30 p.m.–3 p.m.: Proposed Recommendations for Resolving Generic Safety Issue (GSI) 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the recommendations proposed by the staff for resolving GSI–156.6.1, and related matters.

3:15 p.m.–4:45 p.m.: Status of NRR Activities in the Fire Protection Area (Open)—The Committee will hear presentations by and hold discussions with representatives of the Office of Nuclear Reactor Regulation (NRR) regarding the status of ongoing and proposed NRR activities associated with fire protection.

5 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 ACRS report on Technology-Neutral Framework for Future Plant Licensing.

Friday, September 7, 2007, 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.–9:30 a.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

9:30 a.m.–9:45 a.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

9:45 a.m.–10 a.m.: Subcommittee Report (Open)—The Committee will hear a report by and hold discussions with the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim review of the license renewal application for the Fitzpatrick Nuclear Plant.

10:15 a.m.–11:45 a.m.: Draft Report on Quality Assessment of Selected NRC Research Projects (Open)—The Committee will discuss a draft ACRS report on the results of the quality assessment of the NRC research projects on: Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping; Cable Response to Live Fire (CAROLFIRE) Testing; and Technical Review of On-Line Monitoring Techniques for Performance Assessment.

12:45 p.m.–2:45 p.m.: Draft ACRS Report on the NRC Safety Research Program (Open)—The Committee will discuss a draft ACRS report on the NRC Safety Research Program.

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

Saturday, September 8, 2007, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

12:30 p.m.–1 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 2, 2006 (71 FR 58015). 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 p.m., (ET). ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr@nrc.gov*, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/* (ACRS & ACNW Mtg schedules/agendas).

Video teleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., (ET), at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the video teleconferencing link. The availability of video teleconferencing services is not guaranteed.

Dated: August 8, 2007.

J. Samuel Walker,

Acting Secretary of the Commission.

[FR Doc. E7–15887 Filed 8–13–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

The ACRS Subcommittee on Planning and Procedures will hold a meeting on September 5, 2007, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

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

The agenda for the subject meeting shall be as follows:

Wednesday, September 5, 2007, 8:30 a.m.–10 a.m.

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

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

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

Dated: August 7, 2007.

Cayetano Santos,

Branch Chief, ACRS.

[FR Doc. E7–15889 Filed 8–13–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS); Meeting of the Subcommittee on Plant License Renewal; Notice of Meeting

The ACRS Subcommittee on Plant License Renewal will hold a meeting on September 5, 2007, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, September 5, 2007—10:30 a.m. until 5 p.m.

The Subcommittee will discuss Fitzpatrick license renewal application and the associated Safety Evaluation Report (SER) prepared by the NRR staff. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, Entergy Nuclear Northeast, and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Charles G. Hammer (telephone 301/415–7363) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 6:45 a.m. and 3:30 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: August 7, 2007.

Cayetano Santos,

Branch Chief, ACRS.

[FR Doc. E7–15890 Filed 8–13–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Sunshine Act Meetings

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATE: Weeks of August 13, 20, 27, September 3, 10, 17, 2007.

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

Strike out version App 2

August 6, 2007

SCHEDULE AND OUTLINE FOR DISCUSSION
545th ACRS MEETING
SEPTEMBER 6-8, 2007

THURSDAY, SEPTEMBER 6, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - ~~10:30~~ A.M. 10:11 Final Review of the License Renewal Application for the Pilgrim Nuclear Power Station (Open) (OLM/MB)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Entergy Nuclear Operations, Inc. regarding the license renewal application for the Pilgrim Nuclear Power Station and the associated NRC staff's final Safety Evaluation Report.

   Members of the public may provide their views, as appropriate.

~~10:30 - 10:45~~ A.M. 10:11 – 10:47 ***BREAK***

3) ~~10:45~~ - ~~12:15~~ P.M. 10:47 -12:25 Revisions to Standard Review Plan (SRP) Sections 19.0 and 19.2 (Open) (GEA/DCF)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding revisions to SRP Sections 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," and 19.2, "Review of Risk Information Used to Support Permanent Plant Specific Changes to the Licensing Basis: General Guidance."

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

~~12:15~~ - 1:30 P.M. 12:25 ***LUNCH***

4) 1:30 - 3:00 P.M. Proposed Recommendations for Resolving Generic Safety Issue (GSI) 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment" (Open) (WJS/DB)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff regarding the recommendations proposed by the staff for resolving GSI-156.6.1, and related matters.

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

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

5) 3:15 - 4:45 P.M. Status of NRR Activities in the Fire Protection Area (Open) (OLM/CGH)

5.1) Remarks by the Subcommittee chairman

5.2) Briefing by and discussions with representatives of the Office of Nuclear Reactor Regulation (NRR) regarding the status of ongoing and proposed NRR activities associated with fire protection.

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

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

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

Discussion of proposed ACRS reports on:

6.1) License Renewal Application for the Pilgrim Nuclear Power Station (OLM/MB)

6.2) Revisions to Standard Review Plan Sections 19.0 and 19.2 (Tentative) (GEA/DCF)

6.3) Proposed Recommendations for Resolving Generic Safety Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment" (WJS/DB)

6.4) Technology-Neutral Framework for Future Plant Licensing (WJS/DCF)

FRIDAY, SEPTEMBER 7, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - 9:30 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (WJS/FPG/SD)

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

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

9) 9:30 - 9:45 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (WJS, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

10) 9:45 - 10:00 A.M. Subcommittee Report (Open) (MVB/CGH)
Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding interim review of the license renewal application for the Fitzpatrick Nuclear Plant.

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

11) 10:15 - 11:45 A.M. Draft Report on Quality Assessment of Selected NRC Research Projects (Open) (DAP/HPN)
11.1) Remarks by the Subcommittee Chairman
11.2) Discussion of a draft ACRS report on the results of the quality assessment of the NRC research projects on: Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping; Cable Response to Live Fire (CAROLFIRE) Testing; and Technical Review of On-Line Monitoring Techniques for Performance Assessment.

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

12) 12:45 – 2:45 P.M. Draft ACRS Report on the NRC Safety Research Program (Open) (DAP, et.al/HPN, et.al)
12.1) Remarks by the Subcommittee Chairman
12.2) Discussion of the draft ACRS report on the NRC Safety Research Program.

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

13) 3:00 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
13.1) License Renewal Application for the Pilgrim Nuclear Power Station (OLM/MB)
13.2) Revisions to Standard Review Plan Sections 19.0 and 19.2 (Tentative) (GEA/DCF)
13.3) Proposed Recommendations for Resolving Generic Safety Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment" (WJS/DB)
13.4) Technology-Neutral Framework for Future Plant Licensing (WJS/DCF)

SATURDAY, SEPTEMBER 8, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 12:30 P.M. **(10:30-10:45 BREAK)**
Preparation of ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 13.

15) 12:30 - 1:00 P.M.
Miscellaneous (Open) (WJS/FPG)
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.

ACRS	ACRS	ACRS	Filed: CM-180
MA	SD/bjw	CS	
08/ /07	08/ /07	08/ /07	

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545th FULL COMMITTEE MEETING

September 6-8, 2007

#3 Attendees

PLEASE PRINT CLEARLY

NRC Attendees
TODAY'S DATE: September 6, 2007

	NAME	NRC ORGANIZATION
1	Martin Stutzke	RES/DRASP
2	Gerry Gulla	NRO/DSRA
3	Lyn Mrowca	NRO/DSRA
4	Mark Rubin	NRR/DRA
5	G. Parry	NRR/DRA
6	Ram Sulbaratha	NRO/DNRL
7	Theresa Clark	NRO/DSRA
8	Malcolm Patterson	NRO/DSRA
9	Sud Basu	RES/DRASP
10	Donnie Harrison	NRR/DSS
11	John Lai	NRO/DSRA
12	Harold Vander Molen	RES/DRASP
13	John Kaufman	RES/DRASP
14	Jack Foster	RES
15	Farouk Eltawila	RES/DRASP
16	Muhdi Reisi Fard	RES/DRASP
17	Alex Klein	NRR/DRA
18	Daniel Frumkin	NRR/DRA
19	Chuck Moulton	NRR/DRA
20	Harry Barrett	NRR/DRA
21	Pete Barbadoro	NRR/DRA
22	Paul Loin	NRR/DRA
23	Sunil Weerakody	NRR/DRA
24	Ray Gallucci	NRR/DRA
25	Naeem Iqbal	NRR/DRA
26	Perry Buckberg	NRR/DLR
27	James Medoff	NRR/DLR
28	Dan Hoang	NRR/DLR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545th FULL COMMITTEE MEETING

September 6-8, 2007

PLEASE PRINT CLEARLY

NRC Attendees
TODAY'S DATE: September 6, 2007

	NAME	NRC ORGANIZATION
29	D. Gettys	NRC
30	Dave Wrona	NRR/DLR
31	Jim Davis	NRR/RLR
32	Lambros Lois	NRR/DSS
33	Glenn Meyer	Region I
34	Donnie Ashley	NRR/RLRA
35	Ken Chang	NRR/DLR
36	Raj Auluck	NRR/DLR
37	Girija Shuckly	ACRS
38	Kim Green	NRR/DLR
39	Bill Rogers	NRR/DLR
40	Angelo Stubbs	NRO/DSRA
41	Steve Hoffman	NRR/DLR
42	Barry Elliott	NRR/DCI
43	Samson Lee	NRR/DLR
44	Farideh Saba	NRR/DLR
45	Zuhan Xi	NRR/DLR
46	Yeon-Ki Chung	NRR/DLR
47	Jonathan Rowley	NRR/DLR
48	Tommy Le	NRR/DLR
49	Chris Sydnor	NRR/DCI
50	Maurice Heath	NRR/DLR
51	Jon Thompson	NRR/DPR
52	Don Dube	NRO/DSRA
53	Ronaldo Jenkins	RES/DRASP
54		
55		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545th FULL COMMITTEE MEETING

September 6-8, 2007

PLEASE PRINT CLEARLY

Visitors

TODAY'S DATE: September 6, 2007

	NAME	ORGANIZATION
1	Reza Ahrable	Entergy
2	Steve Bethay	Entergy
3	Mike stroud	Entergy
4	Ed Sanchez	Entergy
5	David J. Lack	Entergy
6	Ted Ivy	Entergy
7	Franze-Josef Ulm	MIT for Entergy
8	Raymond Place	Entergy
9	John Dyckman	Entergy
10	Fred Mogolesko	Entergy
11	Andrew Taylor	Entergy
12	Alan Cox	Entergy
13	Garry G. Young	Entergy
14	Brian R. Sullivan	Entergy
15	David Mannae	Entergy
16	Rick Ploss	Entergy
17	Bryan Ford	Entergy
18	Jim Costerlio	Entergy
19	Jon Woodfield	ISL, Inc.
20	John Hufnagel	Exelon
21	John McCann	Entergy
22	Chalmer Myer	SNC
23	Terry Garrett	WCNOC
24	Paul Crawley	Stars/APS
25	Alan Levin	AREVA
26	Spyros Tramforos	LINK

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545th FULL COMMITTEE MEETING

September 6-8, 2007

PLEASE PRINT CLEARLY

Visitors

TODAY'S DATE: September 6, 2007

	NAME	ORGANIZATION
27	Jim Riley	NEI
28	Raj Tripalki	self

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545th FULL COMMITTEE MEETING

September 6-8, 2007

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NRC Attendees
TODAY'S DATE: September 7, 2007

	NAME	ORGANIZATION
1	Girija Shukla	ACRS
2	Jocelyn Mitchell	RES
3	Don Helton	RES
4	Perry Buckberg	NRR/DLR
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545TH FULL COMMITTEE MEETING

SEPTEMBER 6-8, 2007

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TODAY'S DATE: September 6, 2007

	NAME	NRC ORGANIZATION
1	MARTIN STUTZKE	RES/DRASP/OERA
2	Gerry Gulla	NRO/DSRA/SPLA
3	Ram Subbaratnam	NRO/DNRL/NGRA
4	Lynn Mrowca	NRO/DSRA/SPLA
5	Mark Rubin	NRR/DRA/APLA
6	Gareth Parry	NRR/DRA
7	Theresa Clark	NRO/DSRA/SPLA
8	MALCOLM PATTERSON	NRO/DSRA/SPLA
9	Sud Basu	RES/DRASP/NRCA
10	Don Helton	RES/DRASP/NRCA
11	Donnie Harrison	NRR/DSS/SBPB
12	John Lai	NRO/DSRA/SPLB
13	Harold Vandermolen	RES/DRASP/OEGIB
14	John V KAUFFMAN	RES/DRASP/OEGIB
15	Jack Foster	" "
16	FAROUK ELTAWILA	RES/DRASP
17	Mehdi Reisi Fard	RES/DRASP/OEGIB
18	Alex Klein	NRR/DRA/AFPB
19	Daniel Frumkin	NRR/DRA/AFPB
20	Chuck Moulton	NRR/DRA/AFPB
21	Harry Barrett	NRR/DRA/AFPB
22	Pete Barbadoro	NRR/DRA/AFPB
23	Paul Lain	NRR/DRA/AFPB
24	Sunil Weerakkody	NRR/DRA/
25	Ray Gallucci	NRR/DRA/FPB
26	Naeem IQBAL	NRR/DRA/AFPB
27		
28		

(NRC STAFF)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545TH FULL COMMITTEE MEETING

SEPTEMBER 6-8, 2007

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TODAY'S DATE: September 6, 2007

	NAME	NRC ORGANIZATION
1	Perry Buckberg	NRR/DLR
2	JAMES MEDOFF	NRR/DLR/RLRC
3	DAN HOANG	NRR/DLR/RLRC
4	E Gettys	NRC
5	DAVE WRONA	NRC/NRR/DLR/RLRA
6	Jim Davis	NRC/NRR/DLR/RLRC
7	Lambros Lois	NRR/DSS/SRXB
8	Glenn Meyer	NRC/Region I
9	Donnie Ashley	NRR/RLRA
10	Ken Chang	NRR/DLR
11	Raj Auluck	NRR/DLR/RLRB
12	GIRIJA SHUKLA	ACRS
13	Kim Green	NRR/DLR
14	Bill Rogers	NRR/DLR
15	ANGELO STUBBS	NRO/DSRA/SBPA
16	Steve Hoffman	NRR/DLR
17	BARRY ELLIOT	NRR/DCI/CVIB
18	SAMSON LEE	NRR/DLR
19	Farideh Saba	NRR/DLR
20	Zuhan Xi	NRR/DLR/RLRC
21	Yeon-Ki Chung	NRR/DLR
22	Jonathan Rowley	NRR/DLR
23	Tommy Le	NRR/DLR
24	Chris Sydnor	NRR/DCI/CVIB
25	MAURICE HEATH	NRR/DLR
26	Jon Thompson	NRR/DPR/PSPB
27	DON DUBE	NRO/DSRA
28	RONALDO JENKINS	RES/DRASP/PRASB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545TH FULL COMMITTEE MEETING

SEPTEMBER 6-8, 2007

PLEASE PRINT

TODAY'S DATE: September 6, 2007

	NAME	ORGANIZATION
1	REZA AHRABLI	ENTERGY
2	Steve Bethay	Entergy
3	MIKE STROUD	ENTERGY
4	Ed Sanchez	Entergy
5	David J. Lach	Entergy
6	Ted Ivy	Entergy
7	FRANZ-JOSEF ULM	MIT for ENTERGY
8	Raymond Pace	Entergy
9	John Dyckman	"
10	FRED MOGOLESKO	Entergy
11	ANDREW TAYLOR	ENTERGY
12	ALAN COX	ENTERGY
13	GARRY G. YOUNG	ENTERGY
14	Brian R. Sullivan	Entergy
15	DAVID MANNAI	ENTERGY
16	Rick Moss	Entergy
17	Bryan Ford	Entergy
18	Jim Costello	Entergy
19	Jon Woodfield	ISG, Inc.
20	~~E. Gettys~~	~~NRC~~
21	John Hufnagel	Exelon
22	John McCann	Entergy
23	Chalmer Myer	SNC
24	Terry Garrett	WCNOC
25	Paul Crawley	STARS/APS
26	~~[illegible]~~	~~[illegible]~~
27	Alan Levin	AREVA
28	SPYROS TRAMFOROS	LINK

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545TH FULL COMMITTEE MEETING

SEPTEMBER 6-8, 2007

PLEASE PRINT

TODAY'S DATE: September 6, 2007

	NAME	ORGANIZATION
1	Jim Riley	NEI
2	Raji Tripathi	Self
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545TH FULL COMMITTEE MEETING

SEPTEMBER 6-8, 2007

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TODAY'S DATE: September 7, 2007

	NAME	NRC ORGANIZATION
1	GIRIJA SHUKLA	ACRS
2	Jocelyn Mitchell	RES
3	Don Helton	RES
4	Perry Buckberg	NRR/DLR
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September 13, 2007

SCHEDULE AND OUTLINE FOR DISCUSSION
546th ACRS MEETING
OCTOBER 4-6, 2007

THURSDAY, OCTOBER 4, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:30 A.M. Digital Instrumentation and Controls (I&C) Project Plan and Interim Staff Guidance (Open) (GEA/GSS)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding Digital I&C interim staff guidance on Cyber Security, Diversity & Defense in Depth, Highly Integrated Control Room – Communications, and Highly Integrated Control Room – Human Factors, as well as the Digital I&C Project Plan.

   Members of the public may provide their views, as appropriate.

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

3) 10:45 - 12:15 P.M. Draft Generic Letter 2007-XX, "Managing Gas Intrusion in ECCS, Decay Heat Removal, and Containment Spray Systems" (Open) (SAK/DB)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding the Draft Generic Letter 2007-XX, "Managing Gas Intrusion in ECCS, Decay Heat Removal, and Containment Spray Systems."

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

12:15 - 1:30 P.M. ***LUNCH***

4) 1:30 - 3:00 P.M. Dissimilar Metal Weld Issue (Open) (WJS/CGH)

- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff and nuclear industry regarding the advanced finite element analysis performed by the industry to provide basis for leak-before-break and the associated NRC staff's evaluation.

Members of the public may provide their views, as appropriate.

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

5) 3:15 - 5:15 P.M. Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN)

- 5.1) Remarks by the Subcommittee Chairman
- 5.2) Discussion of the draft ACRS report on the NRC Safety Research Program.

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

6) 5:30 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

- 6.1) Digital I&C Interim Staff Guidance (GEA/GSS)
- 6.2) Draft Generic Letter 2007-XX, "Managing Gas Intrusion in ECCS, Decay Heat Removal, and Containment Spray Systems" (SAK/DB)
- 6.3) Dissimilar Metal Weld Issue (WJS/CGH)

FRIDAY, OCTOBER 5, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - 11:00 A.M. Meeting with NEI, EPRI, and INPO to Discuss Industry Activities (Open) (OLM/MB)

- 8.1) Remarks by the Subcommittee chairman
- 8.2) Briefing by and discussions with representatives of NEI, Electric Power Research Institute (EPRI), and Institute of Nuclear Power Operations (INPO) regarding industry activities.

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

9) 11:15 - 12:15 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (WJS/FPG/SD)

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

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

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

10) 1:15 - 1:30 P.M. <u>Reconciliation of ACRS Comments and Recommendations</u> (Open) (WJS, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

11) 1:30 - 2:15 P.M. <u>Draft Final Report on Quality Assessment of Selected NRC Research Projects</u> (Open) (DAP/HPN)
11.1) Remarks by the Subcommittee Chairman
11.2) Discussion of the draft final ACRS report on the results of the quality assessment of the NRC research projects on: Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping; Cable Response to Live Fire (CAROLFIRE) Testing; and Technical Review of On-Line Monitoring Techniques for Performance Assessment.

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

12) 2:30 - 7:00 P.M. <u>Preparation of ACRS Reports</u> (Open)
Discussion of proposed ACRS reports on:
12.1) Digital I&C Interim Staff Guidance (GEA/GSS)
12.2) Draft Generic Letter 2007-XX, "Managing Gas Intrusion in ECCS, Decay Heat Removal, and Containment Spray Systems" (SAK/DB)
12.3) Dissimilar Metal Weld Issue (WJS/CGH)
12.4) Draft ACRS Report on the NRC Safety Research Program (DAP/HPN)

<u>SATURDAY, OCTOBER 6, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

13) 8:30 - 12:00 P.M. **(10:30-10:45 A.M. BREAK)** <u>Preparation of ACRS Reports</u> (Open)
Continue discussion of proposed ACRS reports listed under Item 12.

14) 12:00 - 12:30 P.M. <u>Miscellaneous</u> (Open) (WJS/FPG)
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.

ACRS	ACRS	ACRS	Filed: CM-180
GSS/bjw	MA	CS	
09/ /07	09/ /07	09/ /07	

Appendix 5 List

LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
545th ACRS MEETING
September 6-8, 2007

MEETING HANDOUTS

AGENDA ITEM #	DOCUMENTS/HANDOUTS LISTED IN ORDER
1.	Opening Remarks by the ACRS Chairman 1. 1.Items of Interest
2.	Final Review of the License Renewal Application for the Pilgrim Nuclear Power Station 2. Pilgrim Nuclear Power Station License Renewal Safety Evaluation Report (Slides from Perry Buckberg, NRC/NRR) 3. Pilgrim Nuclear Power Station License Renewal (Slides from Entergy)
3.	Revisions to Standard Review Plan (SRP) Sections 19.0 and 19.2 4. Standard Review Plan (SRP) Sections 19.0 and 19.2 (Slides from NRC/NRO/DSRA
4.	Proposed Recommendations for Resolving Generic Safety Issue (GSI) 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment" 5. Policy Issue, Notation Vote dated May 19, 1993. Subject: Recommendation on Large Release Definition. (Memo from James M. Taylor, Executive Director for Operations, to The Commissioners.) 6. Information Bridge/Calculations in Support of a Potential Definition 7. An Approach For Estimating the Frequencies of Various Containment Failure Models and Bypass Events (NUREG/CR-6595, Rev 1) 8. GI-156.6.1, "Pipe Break Effects on Systems and Components Inside Containment (Slides from Harold Vander Molen (RES) & Abdul Sheikh (RES)
5.	Status of the NRR Activities in the Fire Protection Area 9. Fire Protection Program Briefing for the ACRS (Slides from NRR)
6.	Preperation of ACRS Reportt

[Note: Some documents listed herein may have been provided or prepared for the Committee use only. These documents must be reviewed prior to release to the public.]

8. Future ACRS Activities/Report of the Planning and Procedures Subcommittee
    10. Planning & Procedures Handout

9. Reconciliation of ACRS Comments and Recommendations
    11. Reconciliation of ACRS Comments and Recommendations Handout

10. Subcommittee Report

11. Draft Report on Quality Assessment of Selected NRC Research Projects

12. Draft ACRS Report on the NRC Safety Research Program

13. Preperation of ACRS Reports

**Copies of most of the handouts can be obtained through the transcript copy found in the Agency Document Management System (ADAMS) or a complete set can be requested by calling the ACRS office of the NRC.

[Note: Some documents listed herein may have been provided or prepared for the Committee use only. These documents must be reviewed prior to release to the public.]

ITEMS OF INTEREST

545th ACRS MEETING

SEPTEMBER 6-8, 2007

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
545th MEETING
September 6-8, 2007

CONGRESSIONAL CORRESPONDENCE

MEMORANDUM AND ORDERS

RECENTLY ISSUED SIGNIFICANT REACTOR ENFORCEMENT ACTIONS

INSIDE NRC

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NR[illegible] Yellow Announcement

UNITED STATES
NUCLEAR REGULATORY COMMISSION

Announcement No. 092

Date: September 4, 2007

To: **All NRC Employees**

SUBJECT: In Memoriam - Edward McGaffigan, Jr.

It is with deep regret that I must inform you of the passing of Commissioner Edward McGaffigan Jr.

After a long battle with melanoma, Commissioner McGaffigan passed away on Sunday, September 2, 2007. He was 58. All of us at the NRC extend our prayers and profoundest sympathies to Ed's family. He is survived by his Mother, Margaret; Brother, Brien; Sister, Kathleen; Son, Edward; and Daughter, Margaret. His wife of 18 years, Peggy, passed away in 2000.

E[illegible] first appointed to the Commission by President Clinton in 1996, and began an unprecedented third term of service in O[illegible] of 2005. He lived to mark the 11th anniversary of his first swearing-in at the end of August. But while many people know that Ed was the longest-serving Commissioner in the agency's history, only those of us who were privileged to work alongside him can appreciate how devoted he was to his work and the value of public service... how much he believed in the mission of the NRC... and how much he admired and respected the men and women who work here.

With respect to funeral arrangements, the viewing will be from 6:00 to 8:00 p.m. on Wednesday, September 5, 2007, at Murphy's Funeral Home, 4510 Wilson Boulevard, Arlington, Virginia. A Mass of Christian Burial will be celebrated at 10:30 a.m. on Thursday, September 6, 2007, at St. Agnes Catholic Church, 1914 N. Randolph Street, Arlington, Virginia, followed by burial at Columbia Gardens Cemetery, 3411 Arlington Boulevard, Arlington, Virginia. In lieu of flowers, the family has requested that donations be made to Huntington's Disease Society of America, Washington Metro Area Chapter, 8303 Arlington Boulevard, Suite 210, Fairfax, VA 22031 (703-204-4634). Cards and letters may be addressed to: The Family of Edward McGaffigan Jr., 4818 North 37th Street, Arlington, VA 22207-2912.

Agency employees, with the consent of their supervisors, are authorized excused absence to attend the funeral services. Please note that a memorial service for agency employees and other professional colleagues will be held at or near the NRC headquarters in a few weeks. Details will be sent out in a future announcement.

/RA/

Dale E. Klein
Chairman

NRC [illegible]w Announcements Index

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs Telephone: 301/415-8200

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

Web Site: http://www.nrc.gov

No. S-07-040 August 29, 2007

The Human Factor in Nuclear Safety

Dr. Peter B. Lyons, Commissioner
U.S. Nuclear Regulatory Commission

Joint IEEE Conference on Human Factors and Power Plants and
Workshop on Human Performance - Root Cause - Trending -
Operating Experience - Self Assessment

Monterey, California

August 27, 2007

Introduction

I welcome all of you to this first joint meeting between IEEE's conference on human factors and power plants and the human performance, root cause, trending, operating experience, and self-assessment workshop (HPRCT). There is clearly an extensive range of topics being explored this week, but the two groups that have joined together for this meeting also appear to share many similar objectives. I commend all of you for taking this mutual step, for I strongly believe that opportunities such as this can promote great synergism in information exchanges and help us all to better achieve nuclear plant safety now and into the future. As usual, I must preface my remarks today with the statement that they represent my personal thoughts and not necessarily those of the Commission.

Optimizing the Human Factor

The title of my speech today begins with "The Human Factor," and is meant to emphasize the importance of the human element. I did not use the term "Human Factors" since that term is often used as a label for a long list of interrelated research areas. I am taking this approach today because of a fascinating discussion I had recently with one of the operators who was at the controls of Three Mile Island Unit 2 on the night of the accident that became a defining event for this industry and for the NRC. Although I had already read with great interest the official reports of that event, hearing it and re-living it through the personal story of one of the actual operators made me profoundly aware of the nature of the relationship between operator and machine. That relationship seems to be defined by a constant tension between what might be called "oneness" and "separateness."

For example, there is often a certain sense of connection between operator and machine. The former TMI operator discussed his need to feel this connection through the controls and instruments of the plant, and he expressed his own concern that such a feeling could be lost in an all-digital, control-room environment. I think most of us can relate to such a feeling, for example, as we drive our cars and "feel" that connection through all of our senses as well as the instrument panel. I can imagine that race car drivers and airplane pilots are even more attuned to this feeling of connection to their machines and the environment in which they operate.

Popular movies often examine the degree of connection between computers and humans as the means to explore our relationship with machines. Such stories can be entertaining, but nuclear professionals need to deal with the reality of the Human Factor, that is, the question of "what is the optimal degree of connectedness, both physical and cognitive, that should exist between operators and the plants they operate?" I suspect that many of you here today have made professional careers out of answering this and associated questions.

For the designers and regulators of nuclear power plants, I submit that the "optimal" degree of connectedness should relate predominantly to overall plant safety. I probably wouldn't get any argument on that point, but I also acknowledge that the devil is always in the details. I believe that digging into those details means staying well-grounded in real operational environments. Designers, engineers, researchers, scientists, and regulators must stay grounded in the details of reality by putting themselves in the shoes of real operators and in their real environments and experience their world through their senses. Or, as one Human Factors pundit once said, "Humans are infinitely creative when it comes to making mistakes."

Although I am using cognitive and physical operator interfaces as the example here, this question of what is the optimal degree of human connectedness might also apply more broadly to areas such as safety culture, organizational performance, root cause analysis, fitness for duty, and knowledge management. In these cases, the answer might be different than for the control room example. For example, in the case of safety culture, the concept of optimum human connectedness might have far more to do with optimizing the questioning attitude of plant personnel. My point is that these broader areas of interest are also vitally important to understand, and my message to you is the same: you must stay well grounded with the real people, doing the real jobs, in their real environments.

A Human Factor Taxonomy

The history of our technological advances is replete with examples of how the Human Factor has contributed to countless events and accidents, both large and small, serious and minor. Stories and anecdotes can help convey the sense of its potential importance, but we still need a taxonomy of some kind to systematically organize and sort all the aspects of the Human Factor that should be considered in the design of a machine, a technical enterprise, an industrial facility, or a nuclear power plant. Those of you who are specialists in human performance and reliability or in organizational factors may already use various taxonomies in your work.

However, let me offer a simple one - one that applies very broadly to all human enterprise in general. It has only two categories: first is the ways in which accidents have happened before, and second is the ways in which accidents could happen in the future but that have not happened yet. The first category provides us with lessons that we must apply so those particular failures and accidents do not happen again. The second category requires both creative and systematic thinking about how things can possibly go wrong that have never happened before. Complicating this is the relentless advance of technology. How we go about understanding what went wrong in the past and predicting what could go wrong in the future must constantly change as the underlying technologies change and evolve. The advent of computer-based safety systems and highly integrated control rooms is a clear example.

Category 1 - Learning Lessons

Let me briefly examine each of these two categories of my Human Factor taxonomy in more detail, starting with the first category involving learning from experience. Broadly sharing and using operating experience is really the only means to address this category and to avoid repeat problems. To help accomplish this, the NRC is working with other international regulatory bodies in countries in which highly advanced computerized control rooms were put into operation during a time when the U.S. was experiencing a hiatus of new nuclear plant construction. However, during this same time, several U.S. vendors developed digital I&C systems for use abroad, so our industrial expertise in this area was clearly advancing, even if the systems weren't in use here.

As the U.S. now prepares for potential new plant construction, we are fully leveraging this international experience to help gain the safety benefits we seek. The NRC's research in this area also looks to industries beyond nuclear power. Specifically, we have been seeking insights in areas such as aerospace, transportation, petrochemical applications, medical devices, and the military. In utilizing these insights, we are being careful to fully understand the differences in their safety functions and the degree to which they are relied upon to control hazards.

The U.S. stands to significantly benefit from international experience and, to the extent that advanced nuclear plant designs are licensed, we will also be providing increasing contributions to the international knowledge base. The infrastructure for managing this sharing of experience is already beginning to take form, and we must be careful to capture the most useful information and not to duplicate efforts. For example, the international Organization for Economic Co-operation and Development (OECD), through its Nuclear Energy Agency (or NEA) recently became the Secretariat for an initiative originated by NRC. Known as the Multi-National Design Evaluation Program (or MDEP), 10 countries are currently participating in this initiative to standardize worldwide nuclear power plant designs, regulatory reviews, and quality assurance standards, to improve regulatory efficiency, and to promote international safety and security. A first stage effort is for NRC to collaborate with the Finnish and French regulators on reviews of the AREVA EPR design. The NRC is actively engaged in discussions with the Finnish regulator on its reviews of the digital I&C system for the Olkiluoto Unit 3 currently under construction. Although the MDEP participants include only regulators, interactions with industry are planned as an important aspect of this project.

Also, I'm very pleased with the NEA's development of a new database, named Computer Systems Important to Safety, or COMPSIS, to collect digital system operating experience information to support improved operation and regulation of digital systems and its continued sponsorship of workshops on human and organizational factors. The NRC encourages and supports these efforts.

Examples

As I've noted, specific examples of past problems can be useful as anecdotes that remind us of the importance of the Human Factor. It is in that light that I would like to offer the following examples based on my recent readings that have struck me as particularly noteworthy. They range from the amusing to the deadly serious. My purpose in presenting them is not to minimize the significant improvement in safe operations that have resulted from carefully designed systems, but to call attention to the pitfalls that await anyone who does not thoroughly confront the challenges of designing human-machine interfaces and digital controls.
One example of bad human interface design was the cockpit control panels of the B-17 bombers in WWII. It was cheaper and faster to design and build the panels using a series of closely spaced toggle switches. Unfortunately, two of these adjacent switches were the flaps and the landing gear. When they were initially deployed, it was not uncommon for a just-landed and taxiing B-17 to suddenly belly-flop onto the concrete when the pilot mistakenly hit the landing gear toggle instead of the one for the flaps.

Another example of poor human interface design was the modification made to many U.S. police cars in the 1990s that coupled the brake lights to the roof flashing lights so that the brake lights would flash on and off with the roof lights. Unfortunately, in many vehicle models the brake lights were part of the interlock circuit that prevents the shift lever from moving out of park unless the brakes are engaged. This, of course, is intended to be a safety interlock. However, on these modified police cars this safety interlock was actually turning on and off with the flashing lights. This came to light in 1999 only in an accident investigation for a tragedy in which a parked police car was shifted into gear at full throttle, hitting several parade-goers. This can serve as an example of the problems that can happen from connecting safety systems together, either inadvertently or by design, without careful analysis of all the implications.

These are just two of many examples in a fascinating book I recently read entitled "Inviting Disaster - Lessons from the Edge of Technology," by James Chiles. I encourage you to read it, as it is one of the most informative that I have seen on the subject.

In the medical field, the NRC noted that through the 1980s to mid-1990s the number of misadministrations from computerized radiation therapy machines was increasing. Its review determined that nearly half of the events studied involved interface deficiencies that included cryptic or misleading error messages and problems in the data entry routines. One of the most thoroughly studied of medical misadministration events was the THERAC-25 radiation therapy machine that caused significant overdoses of radiation in six known accidents in the late 1980s. These accidents involved serious injuries and death. There were a number of contributing factors involving software design flaws. In one of these, depending on the sequence and timing of the operator's data entry at the keyboard, the software could incorrectly set the intensity of the beam, without any indication to the operator.

Similar data entry problems have caused lock-ups and failures of computer-based systems at nuclear power plants. These have included multiple instances of loss of control room alarm functions and another instance involving the failure of an ATWS mitigation system. Such issues highlight the importance of careful design of human-machine interfaces to minimize potential data entry issues.

Examples of digital system failures continue to come across my desk. For example, last summer a scram at Browns Ferry Unit 3 occurred when a digital network controlling the reactor recirculation pumps experienced a 'data storm' of excessive traffic due to malfunction of one of the components on the network. It seems there was no 'limiter' designed into the network to ensure that the data flow remained within the physical capability of the network.

Then earlier this summer, the Honeywell uranium hexafluoride conversion plant digital control system power supply failed and placed plant components into a start-up configuration while the plant was operating. Operators were able to bypass the failed power supply and restore power to the work stations and communications network. However, when communications were re-established with the plant controllers, the controllers reinitialized as designed. This reconfigured the production equipment for a "cold start," which shut a number of valves. However, because the plant was operating and 'hot,' the valve closure caused some of the process tanks to begin increasing their pressure. The operators noted the increasing pressures and shut the plant down safely.

Although these last two problems were more design-related and not operator interface issues, it remains true that the root cause was the Human Factor.

Category 2 - Predicting Problems

Turning now to the second category of my Human Factor taxonomy, which involves failures that have never happened before, but could. In my view, research is one of the best ways to address this second category, that is, to identify and anticipate the possible problems that have not yet actually occurred.

Early in my term as a Commissioner, I visited the OECD Halden Reactor Project and observed the digital I&C and human-machine interface research being done there. The NRC contributes support for much of this work, which is aimed at addressing challenges that include the impact of rapidly changing technology, increasing complexity, new failure modes, system and human reliability metrics, new concepts of operation, and the need for updating regulatory acceptance criteria and review procedures. Halden is helping to provide us with a growing technical basis for more realistic safety decisions related to the software and hardware of digital systems, the humans that operate and maintain them, and information to enhance human reliability analyses.

The NRC sponsors domestic research predominately through individual contractor arrangements in a case-by-case fashion. However, to improve our ability to make regulatory improvements that keep up with rapidly advancing digital technology and the science of human-machine interfaces, the NRC will begin a public dialog on the potential benefits and challenges of a research, test, and evaluation facility in the U.S for digital safety system and advanced control room applications. My hope is that such an integrated facility, if approved by the Commission, would create synergies and efficiencies not evident in our current approach. Also, I believe this could better attract new graduates and experienced professionals in this highly competitive field. Possibilities include the participation of other government agencies and industries in examining issues, such as hardware and software configuration, system requirements, maintenance approaches, normal and adverse environmental conditions, faulted condition performance, and a variety of human-machine interaction approaches, all evaluated under controlled conditions representative of those in nuclear facilities and in other safety-related applications. I am pleased to announce that this dialog will begin with a public workshop to be held in Atlanta, Georgia, on Sept. 6 and 7, 2007, and continue in Rockville, Md. on Sept. 11. More information is available from our NRC website at www.nrc.gov. I hope you will consider attending or advising your colleagues about it.

All of our research in the digital system area is integrated within our NRC Digital System Research Plan that aims to address many related technical regulatory needs. This publically available plan organizes our digital system safety research into six categories: system characteristics, software quality assurance, risk assessment, cyber-security, emerging new technologies, and advanced reactor I&C and control room designs. In its recent periodic review of the NRC safety research program, the Advisory Committee on Reactor Safeguards (ACRS) gave this plan good marks.

Near-Term NRC Challenges For Review of New Plant Applications

In addition, earlier this year the NRC formed a senior management steering group and several specific task working groups with industry to focus on specific problems related to our upcoming reviews of digital I&C systems in new power plant applications and replacement systems for existing plants, as well as certain materials licensees. The NRC expects to receive up to seven applications for new plants later this year, with up to 11 more next year. The working groups have held over 25 public meetings to develop near-term interim regulatory guidance to provide greater clarity and predictability to our reviews of these expected applications. Specific areas of focus include diversity and defense-in-depth, highly integrated control room communications and human factors, cyber-security, risk-informed approaches, and the licensing process. Most of the interim staff guidance is due to be finished later this year, but work will continue to further refine and capture this guidance into formal regulatory guides and standards, for instance, IEEE standards developed by the subcommittee that sponsors this conference.

A significant challenge moving forward into the future will be to keep regulatory guidance current with the pace of digital technology progress. Rulemaking cannot always keep to that pace - so we need to rely on guidance documents that can. I see no other answer than for the staff, nuclear research community, and the nuclear industry to maintain a joint and active engagement with the larger multi-industry, technical community for this rapidly evolving technology as you are doing this week.

Human Resources and Technical Expertise

From all indications, we see a coming surge of new plant applications, and the NRC is getting ready to meet this significant new challenge. I see a need for both NRC and industry to attract new people to reemerging work in nuclear power in order to build and maintain the necessary pool of talent to be successful in an environment of growth, without compromising the safety performance of existing plants. One of the most significant of these challenges is that we are competing for digital system and human factors technical expertise with many other industries in a very competitive job market. At the NRC, I believe the solution will be a balance of attracting and building in-house expertise, combined with close links to the expertise at our national laboratories and with programs and facilities that are part of the larger technical infrastructure and communities-of-practice for digital systems across all the industries that use these systems for safety or critical functions. By maintaining our connection with this larger infrastructure and utilizing organizations with broad expertise among many industries, we would expect to efficiently access the most applicable and relevant national and global work being done on safety-critical digital systems.

Another perspective on this same point is that the move toward state-of-the-art I&C systems and human-machine interfaces in our power reactors will certainly enhance the interest and recruitment of the next generation of students to the nuclear industry. But unfortunately, as I visit university research reactors throughout the U.S., I am struck by our national failure to upgrade the instrumentation and controls at our research reactor facilities to state-of-the-art capabilities and the negative impact this must have on our ability to attract new students.

A final perspective on this topic is the need for NRC to stay current in training its own staff on digital system technology, human factors, advanced control rooms, and regulatory requirements. Part of this will have to be accomplished through strong knowledge management programs, since so many of the NRC and industry staff are nearing retirement age.

Closing

In my travels, I've visited several facilities that incorporate advanced control room and computer-based safety and control systems from the plants at Palo Verde, San Onofre, and Waterford that use relatively simple core protection calculators designed in the 1970s, to the Advanced BWR Kashiwazaki-Kariwa Units 6 and 7 in Japan that uses fully computerized control rooms. I've also seen the advanced control room digital retrofit at Oskarshamn Unit 1 in Sweden, the computerized control room of the Civeaux N4 reactor in France with its impressive human-machine interface, and the fully modern digital systems of the research reactors at the OPAL facility in Australia and at Tsinghua University in China. I was also extremely impressed with the digital I&C systems of the newest reactors in the U.S. naval nuclear propulsion program, a program renowned for its rigorous standards and impeccable safety record. Finally, I'm certainly aware of other operating commercial power reactors around the world using digital safety systems with advanced control room designs and I hope to be able to visit some of these in the future.

As a technical person and a safety regulator, I'm drawn to the potential safety benefits of computer-based technology, but I'm also sobered by the challenge of the many failure possibilities that must be addressed for its intended safety-related uses. Nevertheless, I am an optimist that we can achieve improved human-machine interfaces and overall safety performance, provided that the failure vulnerabilities are thoroughly identified, understood, and mitigated. As a regulator, the potential for enhanced safety motivates NRC's ongoing efforts to refine the regulatory requirements that enable such enhancements.

Building on a wealth of experience from other industries as well as the nuclear power industry, the NRC is considering human information gathering and cognitive processes to a greater extent than ever before in the design of advanced and highly integrated nuclear plant control rooms, aided by ongoing and extensive research.

In closing, I will reemphasize my key point: Digital I&C and safety systems offer the potential for improved human-machine interfaces and safety performance, provided that the Human Factor and other failure vulnerabilities are thoroughly identified, understood, and mitigated. Achieving this potential will require industry, the research community, and the NRC to work through new and complex technical issues systematically and thoroughly, with the constant mutual goal of ensuring overall plant safety. Further, to accomplish this efficiently, we must all seek to fully leverage the experience of others in the international community who have moved ahead in applying digital systems to nuclear power plants.

Lastly, although we have an ever-expanding set of new tools to create digital I&C systems that function in more and more complex ways, like the 'brain and nervous system' of a nuclear plant, I believe that we must constantly remind ourselves that increasing complexity will exponentially increase the cost of demonstrating and maintaining safety and also the difficulty in detecting and correcting problems.

I am encouraged by the ongoing dialogue between NRC staff and the industry to tackle topics such as improving the methods to achieve defense-in-depth and diversity, cyber-security, and advanced control room design. As we continue this dialogue and move forward, I think it is useful to remind ourselves that the greatest difficulties will reside in the multitude of details that must be considered. Therefore, success will require a great and constant discipline to master the complexity to ensure it serves only the cause of safety.

Thank you.

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Contributions of Structural Mechanics to the Science of Nuclear Regulation
Dr. Peter B. Lyons, Commissioner
U.S. Nuclear Regulatory Commission

at the
19th International Conference on Structural Mechanics
in Reactor Technology

August 13, 2007

It is an honor to speak to you during the 19th International Conference on Structural Mechanics in Reactor Technology (SMiRT-19). I am extremely pleased to share my perspectives on the role of this conference in the renewed global interest in nuclear energy and to discuss some of the U.S. Nuclear Regulatory Commission's (NRC) future challenges. I especially want to recognize the efforts of Vernon Matzen, conference chairman, and his committee in the planning and execution of this conference.

SMiRT-19 is taking place at a time of significant change in the global outlook of the industry. The technical issues related to design, manufacturing, and construction are becoming more important, similar to the situation in the early 1970s. These conferences, which bring together the world's experts from the structural mechanics community who are involved in the design, construction, and operational phases of nuclear power, have a significant role to play in readiness of this industry and its regulators.

The need for global cooperation on nuclear safety is an urgent matter, because nuclear energy can no longer be regarded as a strictly domestic matter for any individual country. Nuclear power is now a truly international industry, from the mining of the uranium ore, through nearly all the following steps of the fuel cycle. Furthermore, the regulatory and industrial infrastructures are now very different from those of the early 1970s, including the use of new materials, new construction and fabrication methods, and the associated new structural mechanics challenges.

Based on lessons from our past licensing and regulatory experiences, we have a new, improved, licensing process. The combination of the standardized design certification, early site permit, and combined construction and operating license has contributed significantly to the interest in and feasibility of new nuclear projects in the United States. The NRC is continuing to improve our licensing regulations. Recent changes to our Part 52 regulations will further enhance our effectiveness and efficiency.

The new regulatory scheme has undergone its first tests, with the review of early site permits at four locations. We have issued early site permits for Clinton and Grand Gulf, and are working on an early site permit for North Anna. Four reactor designs are certified, with three more in various stages of consideration. Later this year and for the first time in 30 years, the NRC expects to receive up to seven license applications to build and operate new nuclear plants. Eleven additional applications are expected in 2008. To date, we have received letters of interest from several potential applicants, which indicate that NRC may expect that first plant completion to be followed by as many as 30 others. We have even received part of the first combined operating license to be filed. These numbers change frequently, so stay tuned for further developments.

The U.S. manufacturing and industrial capacity to support new construction has been significantly diminished since the 1970s and 1980s. The number of U.S. companies certified by the American Society of Mechanical Engineers (ASME) to produce N-stamped parts has dropped by almost a factor of five since 1980. We also face a challenge in ensuring the quality of the thousands of smaller parts and materials that are manufactured in other parts of the world. The construction of a commercial nuclear plant today involves pumps, valves, motors, fans, pipes... and even bolts... that may be produced by any number of companies—both private and state-owned—around the world. The close scrutiny that regulatory agencies can enforce on major manufacturers to assure that quality components are produced is challenging to achieve for a vastly greater number of sub-vendors that supply parts and materials to the manufacturers.

The International Boiler and Pressure Vessel Code 2007 version was just released and establishes rules of safety governing the design, fabrication, and inspection of boilers, pressure vessels, and nuclear power plant components during construction. A section also provides requirements for (1) containment systems and transport packagings for spent fuel and high-level radioactive waste; and (2) concrete reactor vessels and containment. Some of you attending this conference probably participated in that recent and very important work.

The issue of constructing an advanced reactor around the world raises the importance of international communication and collaboration to a new height. This communication is necessary at regulatory, operational, and supply chain levels. A good example of international regulatory cooperation is the Multinational Design Evaluation Program, or MDEP. The MDEP is an initiative to enhance regulatory cooperation and, where feasible and desirable, to converge on common regulatory requirements and review practices associated with the design reviews of new reactors. Conferences like SMiRT enhance a common understanding of technical issues and facilitate communication and resolution, such that a design can be safely constructed at many locations under different regulatory requirements. In this regard, a common understanding of regulatory practices in different countries is important.

The issue of aircraft impact has obviously taken on new visibility in the post-9/11 world. While aircraft impact was considered in earlier designs in the context of accidental accidents, the explicit consideration of sabotage in designs raises a significant challenge for us all. Sharing of technical knowledge is vital to guard against such threats; however, it is also important that the security of sensitive information is maintained. In April 2007, in support of this issue, the NRC unveiled the third in a series of major steps to enhance the post-9/11 security of nuclear power plants. The agency proposed a rule that would require each applicant for a new reactor design to assess how the design, to the extent practicable, has greater built-in protections to avoid or mitigate the effects of a large commercial aircraft impact, making them less reliant on operator actions than existing plants. That approach allows designers to evaluate potential competing technical factors, such as the response to earthquakes and passive safety systems, while at the same time addressing aircraft impacts. These assessments should look at areas such as core cooling capability, containment integrity, and spent-fuel-pool integrity.

The Commission emphasized that seeking security assessments and examining how designs can be improved is consistent with the traditional approach the NRC has taken to so-called "beyond-design-basis-events," which are considered to have such low probability of occurrence that design features to address them can meet realistic analysis criteria. These are events with conditions exceeding the stresses imposed by the "design-basis-event" conditions for which plants are required to be analyzed according to strict and prescriptive rules. Design-basis-event conditions include large pipe breaks, fires, earthquakes, hurricanes, tornados, and floods. Assessing a new reactor in the early design stages can enable modifications to reduce the need for operator mitigation actions in the event of an airplane crash.

In an August 1985, NRC Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," the NRC said it expected future reactor designers to build in more safety features to cope with so-called severe accidents that went beyond the design basis. However, it did not require specific features, leaving that to plant designers. In the subsequent decades, reactor designs submitted to and approved by the Commission have achieved substantial safety improvements to address such beyond-design-basis-accidents.

To quote NRC Chairman Dale Klein's comment on issuing the proposed rule for public comment, "This is the most recent step in a broad, proactive effort to improve the security of reactors initiated by the NRC after Sept. 11, 2001. We need more technical analysis to understand how to address this." In my view, this proposed rule will give us the opportunity to assess and make changes to new reactor designs early in the design process. I should note that many of the challenges that will be reviewed in these assessments fall within the scope of the structural mechanics issues explored in this conference.

Along with the challenges associated with anticipated construction of new reactors of advanced designs, the prospect of the next generation of nuclear power plants involving technologies such as high-temperature and liquid-metal reactors, derived from the Next Generation Nuclear Plant and the Global Nuclear Energy Partnership initiatives, raises a different set of challenges to this community. The designs will involve new materials and different operational and accident conditions. In recognition of strong programs in other countries related to these technologies, codes and standards will have to be developed with an international perspective.

Despite the nuclear renaissance, the most important issue is still the safety of operating reactors. This conference will help us maintain this focus. Our experiences have shown that the understanding of aging and degradation mechanisms, timely detection through inspection technologies, and implementation of effective remedial measures are vital to maintain safety throughout the operating life. Operation beyond the current 60-year, license-renewal periods may also be sought and would challenge our knowledge of aging phenomena.

Other initiatives also use structural mechanics, such as modification to 10 CFR 50.46a, regarding improved safety through a more risk-informed approach for addressing double guillotine breaks of the largest reactor coolant pipes, which can allow better utilization of water supplies and optimization of safety systems to better cope with more likely events than the large loss of cooling accident. If a new version of 50.46a is approved, it will depend heavily on our ability to maintain very low likelihood of breaks in pipes greater in diameter than the so-called transition break size and on our understanding of and ability to detect flaws and degradation in large pipes.

The incorporation of risk perspectives also raises challenges in realistically characterizing the performance of structures, systems, and components when subjected to beyond-design-basis environments. It is particularly difficult to characterize failure modes of passive components that can experience beyond-design-basis conditions for which the failure data can not be realistically obtained. This community will plan a significant role in establishing realistic assessments of passive component performance to enhance our progress toward risk-informed regulation. The recent NRC experiences, related to risk-informing the pressurized thermal shock rule to assure reactor pressure vessel integrity, highlight the benefit of risk-informed considerations and probabilistic methods.

Natural hazards are another area in which knowledge continues to evolve, and we continue to learn from each significant event worldwide. The December 2005 tsunami is a case in point. It is leading to rapid development in the state-of-the-art of prediction, propagation, and early warning systems. The implementation of performance-based seismic siting approaches in a recent early site permit also reflects a substantial change from the deterministic perspective of early years. The recent earthquake in Japan will provide important data to the entire nuclear community. SMiRT is a forum for both understanding and analyzing external hazards and developing safe designs to resist these hazards.

Let me now switch to the subject of human capital. Both the NRC and the industry are facing critical shortages of experienced staff. No nuclear reactor can operate without trained and dedicated people who have made safety a priority. Regulatory bodies must also have trained and knowledgeable staff. The global growth in nuclear power compels all of us to focus on training the next generation of construction workers, electricians, welders, engineers, operators, managers and regulators.

You may be aware that the NRC is engaged in strenuous efforts to increase our staff by a net of 600 people to handle the increased workload of new plant applications and other nuclear regulatory business. Obviously, we cannot simply hire people off the street and send them out to be nuclear power plant regulators the next day. Even when hiring people with substantial experience in industry, we have found that it takes 6 months to a year of training before they begin thinking and acting like regulators. For recent university graduates, it takes one to two years.

Perhaps one of the most important roles that conferences like SMiRT can play is in the area of knowledge management. The SMiRT conference planners may even consider accepting this as one of their challenges. These conferences, which began at the time of the design and construction of the current generation of plants, can provide historical perspectives on technical issues and lessons learned. Knowledge management is viewed as critical in the United States, and both the NRC and U.S. industry are exploring and implementing strategies for effective knowledge management programs. Your conference also affords opportunities for this professional growth and networking that are vital components of knowledge management. This is particularly important to the NRC, as we assimilate many engineers who are new to the nuclear field and strive to create a new generation of regulatory experts.

As I've indicated, the NRC considers participation in conferences such as SMiRT to be vital for many reasons. Among these reasons, it is consistent with agency policy to have effective outreach efforts with our diverse stakeholders. It is also important that we share information related to our research and regulatory initiatives, get feedback on them, and receive new perspectives from research conducted around the world. Our interest is evident from the diverse NRC staff presentations at this conference. The topics presented cover issues related to operating reactors, licensing of new reactors, and waste disposal facilities. One common thread in these presentations is consideration of risk-informed and performance-based approaches.

I challenge all participants of this conference to move beyond knowledge sharing and to promote common understanding of issues among stakeholders with diverse perspectives, researchers, regulators, operators, and designers. This will facilitate development of universal implementation strategies, which could encourage the use of standardized designs worldwide and help to enable consensus and improved approaches to address safety issues.

Thank you for your attention this morning, I will be happy to take questions.

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NRC Chairman Dale E. Klein

Remarks as Prepared for Delivery

ANS Utility Working Conference
Amelia Island, FL

August 6, 2007

I am delighted to be here this morning to share a few thoughts with you.

The theme of this year's conference is "The Future Begins Now," while the title of his particular session is "Staying Focused." I hope that I will be able to confine my remarks within the tight boundaries of these very specific and concrete descriptions! Actually, I am always impressed by the ingenuity of conference organizers who are able to come up with these topics and themes... which are wonderfully general and rather vague, yet still manage to convey some important concept or message to provide a unifying thread for the conference. And I am pleased to say that the themes, "The Future Begins Now" and "Staying Focused" do capture very well the subjects I want to address this morning.

A few weeks ago I spoke at the U.S. Women in Nuclear Meeting, which was held at Disneyland in Anaheim, California. And just to be impartial, independent and fair—which is what we strive for at the NRC—I decided to come to the other coast, and visit the other Disney resort, here in Florida.

Now, I know that we are actually 200 miles from Orlando, but I ask you to indulge me so that I can make a point that I made in Anaheim, which is that, in my capacity as the Chairman of the Nuclear Regulatory Commission, I regard Disney as my competitor.

That may seem like a somewhat cryptic remark, so let me explain it with a brief story. Bill Gates was once asked who Microsoft's biggest competitor was. He responded, rather surprisingly, that his biggest competitor was Goldman Sachs. Gates explained that both companies were in what he called "the IQ business." "Microsoft must win the IQ war," he said, "or we won't have a future. I don't worry about Lotus or IBM, because the smartest guys would rather come to work for Microsoft. Our competitors for IQ are investment banks such as Goldman Sachs and Morgan Stanley."

So what does that story have to do with nuclear energy and Disneyland? I think it comes down to similar commonality: safety.

I don't mean to discount IQ, of course. But since we have a lot of engineers present, I think we can take that as a given. And, as you know, the NRC is not a Mickey Mouse operation. But what Disney and the nuclear energy business have in common is that they both depend for their success, for their existence, on an absolute commitment to safety.

That focus is key to a successful in-house engineering operation, with a comprehensive education, training and development program. I think you will find this kind of program in companies that understand, as Disney appears to, that without their customers' trust, nothing else matters.

So if the theme of this panel is "Staying Focused," I would urge everyone in this room to remember that the object of our focus—and this includes industry and regulators alike—is safety. This focus must be paramount in the design, construction, operation and oversight of nuclear plants at every stage, of course. But let me elaborate on one area that is of particular concern to me: the ability of the global manufacturing sector to meet the growing demand for high quality nuclear components in a timely way.

I should mention that the relatively small number of firms producing major components at least makes it relatively easier to oversee the quality and authenticity of these components. We face a different challenge in ensuring the quality of the thousands of smaller parts and materials that are manufactured in other parts of the world.

The construction of a commercial nuclear plant today involves pumps, valves, motors, fans, pipes... and even screws... that may be produced by any number of companies—both private and state-owned—around the world. And the close scrutiny that regulatory agencies can bring to bear on major manufacturers to assure that quality components are produced does not always apply with the same intensity to the sub-vendors that supply parts and materials to the manufacturers.

To address this, I have suggested in meetings with regulators from other nations that we establish more extensive channels of communication to share information about any components or equipment that may be substandard, counterfeit, inadequate or inappropriate to a nuclear power plant. Regulatory agencies and industry would benefit from sharing this data under normal circumstances, but it seems to me even more critical during the current worldwide push to build new plants.

Now, it may be remotely possible that some of you have heard me mention this topic before. But I think it is important to reiterate the key point that the NRC depends on industry to be the first line of safety.

It is a well-known adage around my office that if there is something amiss at a commercial nuclear plant, the plant owners and operators should find it first. If they don't find it, INPO should. If INPO doesn't find it—and it falls to the NRC resident inspectors to find it—well, then industry has, in a sense, failed. So what I am trying to do when I revisit these themes is to avert problems before they come to our attention as a regulator.

In addition, there are two other reasons I keep coming back to this theme. First, according to data compiled by the American Society of Mechanical Engineers, the number of ASME Nuclear Certificates held by companies fell worldwide from nearly 600 in 1980, to under 200 this year. More strikingly, the decline was due almost entirely to the loss of nuclear certificates among American companies. The number of certificates held by other nations has remained fairly steady—around [illegible]0—since 1980, but the number of American certificate holders today is one-fifth of what it was 27 years ago. Clearly, this must be a consideration as we contemplate the anticipated growth in demand for parts.

The second point is more anecdotal, but I am sure it is something you have been following in the news. I am referring to the problems with regard to quality control over both food products and manufactured items that are bought and sold on the global market.

This is bad enough when it concerns contaminated consumer products—which is certainly very serious. But it is a matter of even greater concern when supposedly high quality machine components are substandard or counterfeit, particularly when such defective or fraudulent parts could find their way into a commercial nuclear reactor. That has not happened. And I am confident that it will not happen, as long as we remember that at the end of the day, nuclear power plants are really in the safety business.

That covers what I would like to say on "Staying Focused." Now let me say a word about "The Future Begins Now," and then I would be happy to take some questions.

If the much-discussed Nuclear Renaissance is in fact happening—as appears to the be the case—its success may well depend ultimately on public trust. On my visit to various nuclear facilities in Japan earlier this year, I was struck by how much effort the Japanese put into making their commercial nuclear reactors accessible to the public— through viewing areas and visitors centers. I would suggest that this is something plant designers might keep in mind for new reactors here in the U.S. It doesn't take all that much ingenuity to allow people to view the turbines and other parts of plants without putting on slippers or dosimeters, and, in fact, without providing any access points into the reactor. Plant owners and operators can cite safety statistics until, as they say, the cows come home. But, in the end, people tend to trust what they have seen with their own eyes.

After 9/11, access to nuclear facilities was significantly reduced. But now we need to plan for the future. Without compromising security considerations, I think industry needs to re-evaluate its public education policies—including tours.

Another consideration is that the more an industry or business represents the public, and reflects the diversity of society at large, the more likely that it will be able to generate public trust. I think that is an important lesson to keep in mind, as you expand your efforts in workforce development. We all know that one of the challenges facing both industry and regulators is the need to prepare the next generation of engineers, as well as electricians, welders, and other skilled crafts people.

I believe that NEI has just come out with an updated edition of its workforce survey, which contains a great deal of useful information. I look forward to seeing it, and I hope there may be some good news. But even if things do look brighter, that would hardly mean that the challenge is solved. So, to a greater or lesser degree, we still have the same task before us.

I have said before that none of our interests is going to be well served if we spend our time and money chasing after a limited number of candidates. Instead of bidding against each other, all of us – industry and government alike – must focus on an intensive nationwide effort to expand the base of qualified people. And reaching out to people who have not been traditionally well-represented in this business is one of the best ways we can do that.

There is also the simple fact of self-interest: ensuring full access for all potential employees vastly increases the talent pool. And both government and industry are going to need all the talent we can get. In fact, there are now more women in college than men—so if industry wants to build a future with the best and brightest young talent, it needs to attract and encourage people of both genders and all races. I think we can safely say that we no longer live in that time when the phrase "nuclear engineer" referred more or less exclusively to people who look like... well, me!

That brings me to the final point I would like to make. At the NRC we have some very energetic programs for reaching out to small, minority-owned and disadvantaged businesses; and I think that some of these efforts may even be instructive for your own procurement policies. Now, I realize that the notion that government ever operates according to sound business practices strikes many people as a contradiction in terms; and the idea that there may be government initiatives worth imitating makes even less sense! But I do think that what we are doing is at least moving in the right direction: toward greater diversity, more active and inclusive engagement with all levels of society, and therefore greater public support for our activities.

Now, these efforts, while ambitious, are still evolving—so there are some targets we have not yet been able to reach. One of them is an area many of you may not even be very familiar with, but it is a very important category, and one that the government is putting a lot of effort into. I am talking about combat-disabled veteran-owned businesses.

I know from direct experience while I was the Pentagon that our men and women in uniform are highly dedicated and professional. And regardless of the differing opinions people may have on various political questions, I think we can all agree that America owes a great debt of gratitude to those who have been disabled while serving their nation. So I think that if the commercial nuclear power industry is really interested in seeking out the best and brightest, you cannot afford to overlook the nation's disabled veterans.

Let me be clear that in saying all this, I am speaking not as a regulator but simply as someone who cares about this issue and thinks that it is important. So, as "Dale" rather than as "Chairman Klein," I would take this opportunity to remind you that just as the Nuclear Renaissance cannot afford to leave any stone unturned as we seek to expand the talent pool for hiring, in both industry and government, we should not overlook the possibilities for mentoring small businesses. This will not only increase goodwill with the public, it will also enhance the supplier base, and help the industry prepare for the future in an increasingly diverse society.

If you are interested in finding out more information, there are many government resources available. The best place to begin is the Small Business Administration office in your local area; which is staffed by people who possess a great deal of knowledge and experience with these programs, and they can provide some very helpful guidance. Beyond that, please feel free to contact the NRC's Office of Civil Rights and Small Business Office, which will be glad to steer you in the right direction.

Winston Churchill once said that "The future is imminent, though obscure." What that means in plain language is that we don't know what tomorrow, let alone next year, will look like. What we do know, however, is that with great change comes great opportunity. So in this time of momentous change for the commercial nuclear power industry, I ask that you seize the opportunity to do great—and good—things. The future of the industry will be more engaged, fairer, and more representative of society at large, but only if you work to make it so. And the future—as the organizers of this conference wisely remind us—begins today.

Thank you for your attention. I would be happy to take some questions.

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NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs Telephone: 301/415-8200

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

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No. S-07-037

"The NRC and the 'Safety Business'"

Remarks Prepared for
NRC Chairman Dale E. Klein

Women in Nuclear Conference
Anaheim, CA

July 16, 2007

Thank you.

I am very glad to see such a large turnout at this meeting.

Nuclear engineering is much different today than when I first got into the field years ago. Both the academic and business aspects are changing, more rapidly all the time it seems—and not just in the U.S. but around the world. So today, for example, when I travel I find more and more often that my counterpart is a woman... such as Judith Melin in Sweden, or Linda Keen in Canada.

And we are quite proud of the increasingly important role women play at the NRC. If I may brag for a moment, Undine Shoop, an engineer who works in the NRC's Communication's Office, is on the cover of the current issue of *Woman Engineer*.

Now, the changing face of the nuclear power business is a positive development for a number of reasons including basic fairness and greater diversity. There is also the simple fact of self-interest: ensuring full access for women essentially doubles the talent pool. And both government and industry are going to need all the talent we can get. In fact, there are now more women in college than men—so if industry wants to build a future with the best and brightest young talent, it needs to attract and encourage people of both genders and all races.

Another benefit is tied to the fact that the success of the Nuclear Renaissance—if it is to happen—will depend ultimately on public trust. That's a theme I will come back to in a minute. But my point right now is that the more an organization looks like the public, the more likely it is to earn that trust. After all, people are more comfortable with what is familiar. So the more the organization represents society at large, the more likely that it will be trusted by society at large.

This is all the more important today, because one of the potential bottlenecks I see for future growth is workforce development. One of the challenges facing both industry and regulators is the need to prepare the next generation of engineers, as well as electricians, welders, and other skilled crafts people. I have said before that none of our interests is going to be well served if we spend our time and money chasing after a limited number of candidates. Instead of bidding against each other, all of us – industry and government alike – must focus on an intensive nationwide effort to expand the base of qualified people. And reaching out to people who were not traditionally well-represented in this business is one of the best ways we can do that. So I am very pleased to see how large and strong a force WIN has become for expanding the industry's talent pool.

On the subject of workforce development, I want to commend you for your decision to hold this event at Disneyland. I assume you chose this venue so that we could all check out the competition… because I know that I regard Disney as my competitor.

That may seem like a somewhat cryptic remark, so let me explain what I mean with a brief story. Bill Gates was once asked who Microsoft's biggest competitor was. He responded, rather surprisingly, that his biggest competitor was Goldman Sachs. Gates explained that both companies were in what he called "the IQ business." "Microsoft must win the IQ war," he said, "or we won't have a future. I don't worry about Lotus or IBM, because the smartest guys would rather come to work for Microsoft. Our competitors for IQ are investment banks such as Goldman Sachs and Morgan Stanley."

So what does that story have to do with nuclear energy and Disneyland? I think it comes down to a similar commonality: safety.

I don't mean to discount IQ, of course. But since we are in a room full of nuclear engineers, I think we can take high IQ's in the nuclear energy business as a given. And, as you know, the NRC is not a Mickey Mouse operation. But what Disney and the nuclear energy business have in common is that they both depend for their success, for their existence, on an absolute commitment to safety.

That focus is key to a successful in-house engineering operation, with a comprehensive education, training and development program. I think you will find this kind of program in companies that understand, as Disney appears to, that without their customers' trust, nothing else matters.

What does it mean for us to recognize that we are in the safety business? And I say "we" because, of course, the safe operation of commercial nuclear plants is a joint responsibility that requires the active cooperation of the utilities and the NRC. What is required for us to fulfill our separate but complementary responsibilities? I am glad you asked, because that brings me to the theme of our panel, "Rebuilding the Nuclear Industrial Infrastructure."

As I have remarked several times over the last few months, I firmly believe that while the NRC faces significant challenges in the near future, we will not be a roadblock to the anticipated growth in nuclear power, if we receive high-quality applications. But I have also pointed out that there seems to be two other areas that may present bottlenecks. I have already talked about workforce development.

The second potential bottleneck is the ability of the global manufacturing sector to meet the growing demand for high-quality nuclear components in a timely way.

I should mention that the relatively small number of firms producing major components at least makes it relatively easier to oversee the quality and authenticity of these components. But we—and I mean here both the utilities and the regulatory community—face a different challenge in ensuring the quality of the thousands of smaller parts and materials that are manufactured in other parts of the world.

The construction of a commercial nuclear plant today involves pumps, valves, motors, fans, pipes… and even bolts… that may be produced by any number of companies—both private and state-owned—around the world. And the close scrutiny that regulatory agencies and nuclear customers can bring to bear on major manufacturers to assure that quality components are produced does not always apply with the same intensity to the sub-vendors that supply parts and materials to the manufacturers.

To address this, I have suggested in meetings with regulators from other nations that we establish more extensive channels of communication to share information about any components or equipment that may be substandard, counterfeit, inadequate or inappropriate to a nuclear power plant. Regulatory agencies and industry would benefit from sharing this data under normal circumstances, but it seems to me even more critical during the current worldwide push to build new plants.

Whether it involves major components, smaller parts, nuclear plant designs, or the actual construction and operation of power plants, we all have an interest in encouraging high levels of safety, and strong safeguards in every country that participates in the fuel cycle.

Now, when I address the need for industry to join with us in being vigilant in this area it is not intended to cast aspersions. It is simply meant to recognize that the NRC depends on industry to be the first line of safety.

It is a well-known adage around my office that if there is something amiss at a commercial nuclear plant, the plant owners and operators should find it first. If they don't find it, INPO should. If INPO doesn't find it… and it falls to the NRC resident inspector to find it… well, then industry has, in a sense, failed. So what I am trying to do when I revisit these themes is to avert problems before they come to our attention as a regulator.

Still, some people wonder why I keep coming back to this theme. So let me offer two examples of why I think this issue is important, and why I want to ensure that all of us are putting sufficient effort into addressing it. Because I have always said that ensuring high-quality components is a challenge we must address together.

Here is the first example: According to data compiled by the American Society of Mechanical Engineers, the number of ASME Nuclear Certificates holders fell worldwide from nearly 600 in 1980, to under 200 this year. More strikingly, the decline was due almost entirely to the loss of nuclear certificates among American companies. The number of certificates held by other nations has remained fairly steady—around 100—since 1980, but the number of American certificate holders today is one-fifth of what it was 27 years ago. Clearly, this must be a consideration as we contemplate the anticipated growth in the demand for parts.

The second point is more anecdotal, but I am sure it is something you have been following in the news… and it seems to be happening more and more often. I am referring to the problems with regard to quality control over both food products and manufactured items that are bought and sold on the global market.

This is bad enough when it concerns contaminated consumer products—which is certainly very serious. But it is a matter of even greater concern when supposedly high-quality machine components are substandard or counterfeit... particularly when such defective or fraudulent parts could find their way into a commercial nuclear reactor. That has not happened. And I am confident that it will not happen… as long as we remember that at the end of the day, nuclear power plants are really in the safety business.

So the question I would leave you with is: Are we being vigilant enough? Is industry doing enough to:

Establish more rigorous safeguards and oversight in procurement?
Find quality vendors and ensure that they maintain high standards?
Make quality assurance a top priority?
That is my charge to you today.

To come back to the point I made at the beginning—it clearly appears Disney understands this. There is a reason they are known and loved and trusted around the world… because this park is a global enterprise, too. No ride, no attraction will be enticing enough if people don't feel safe bringing their families here.

That same confidence, that trust, should be the ultimate goal of those of you trying to bring about the revival of commercial nuclear power.

Thank you.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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Web Site: http://www.nrc.gov

No. S-07-036

"Perspective on Preparedness for Radiological Terrorism"

Prepared Remarks for

The Honorable Gregory B. Jaczko
Commissioner
U.S. Nuclear Regulatory Commission

at the

52nd Annual Meeting of the Health Physics Society
Portland, Oregon

July 10, 2007

I admire the work that you do and I am glad to have the opportunity to open your session entitled "Community Preparedness for Radiological Terrorism." You will be engaging in technical discussions about radioactive dispersal devices, community response plans, and first responders throughout today's session. The work health physicists perform, whether for a radioactive materials licensee, a nuclear power plant, a university, or a government agency, is designed to ensure protection from the health affects of radioactive materials. Health Physicists have unique training and expertise.

Before I get much further, I want to discuss something that I 1believe is vital to good public policy. To best accomplish its mission of protecting public health and safety and the environment, the Nuclear Regulatory Commission (NRC) should be as open with information as possible and transparent in explaining the processes the Commission uses to make decisions. To be successful we need two things: policy based on sound science, regulatory, and technical decisions, AND public confidence in those decisions. We only get that public confidence when we engage a diverse group of stakeholders, to hear their concerns and ideas, and to make them active participants in our decision making processes.

No where is this more important than in the emergency preparedness arena. This is one of the few issues in which the NRC interacts not only with licensees, but directly with the public. Emergency preparedness is an emotional issue which makes communicating accurate scientific information even more important and challenging. The public must also follow recommendations we make for those protective actions to be effective. Looking at preparedness planning through the public's eyes gives us a much fuller appreciation for what we need to accomplish.

The Nuclear Regulatory Commission has an advantage in that our agency and our licensees have been working with the local communities around nuclear power plants for decades. We have a good understanding of the risks these facilities pose, the time frames in which an accident is likely to occur, and the investments that have been made to develop preparedness plans for the ten-mile emergency planning zones and the fifty-mile ingestion pathways.

That is not to say the agency is not continuing to evolve. The Commission has proposed modifying its regulatory requirements for radiological emergency preparedness in several different areas. The first involves additional requirements for our licensees to provide better information more quickly. Based on advances in technology and emergency management over the last quarter century, they deal with such issues as requiring licensees to have a backup capability to notify government and the public of an incident at a plant, and performing periodic reviews and updates of evaçuation time estimates to better assist in making protective action recommendations. We have also funded evacuation and protective action studies1 at Sandia National Laboratory over the past few years. The preliminary results of these studies show that in certain emergencies resulting in releases of radiological materials – such as short duration or "puff" releases and/or in communities with longer evacuation times, it may be better for people to shelter in place rather than attempt to evacuate. There is a widespread perception that radiological emergency preparedness is equivalent to evacuation. So making a dramatic change such as this would require good communication and stakeholder confidence.

The second area involves the inclusion of security-based drills and exercises. We have a sophisticated exercise program at nuclear power plants but are planning to add more realism into these training opportunities. These exercises may include a spectrum of simulated releases to better familiarize responders with different timing, duration, and severity of events.

Finally, we are exploring a new way of regulating emergency preparedness – a performance based approach which focuses on results as the primary basis for regulatory decision-making. This would allow the agency to more effectively define what "adequate public health and safety" means in an emergency preparedness context. It would involve a broad stakeholder discussion about what the "adequate protection" standard should be, the protection that emergency preparedness plans and procedures should result in, and new more objective and measurable regulations.

Now, none of that is easy or short term, but the NRC has talented and dedicated career staff who constantly work to implement and improve our programs. For the nuclear power plants we regulate, we have sixty-five distinct sites to worry about. Concerns about the potential of radiological terrorism have no defined boundaries and could affect areas that do not have as established of an emergency preparedness infrastructure. This is the challenge and the opportunity that the government faces.

Our role as radiation experts and regulators is to help communicate with the public and educate them about the risks of the materials we know. This benefits the work of emergency responders during planning for the initial and intermediate phases of an incident, but it is also invaluable to providing the "sound science" portion of decision-making about long-term recovery efforts.

The NRC has been involved in revising the Environmental Protection Agency's 1992 Protective Actions Guidelines, and the Department of Homeland Security's addition of a section on radiological dispersal devices and improvised nuclear devices. These recommendations for when communities should take protective actions are vital and the most important aspect of them will involve moving back to communities that may be affected. The interagency community has appropriately not yet defined a standard for long term recovery, instead relying on the principle of "optimization" for determining the appropriate radiation threshold after a radiological event. I agree that one size might not fit all and that there are public benefits to being able to return to communities as quickly as possible. But optimization is rather loosely defined and it is important to remember that the public would look to you for help to make decisions about whether and when it is safe for them to go home.

This type of situation would pose a challenge similar to the anthrax attacks that occurred at the U.S. Capitol when I worked there in 2001. There was no play book for the challenging set of circumstances that confronted decision makers following the discovery of anthrax spores in a Senate office mailroom. There were lengthy debates about how clean was "clean," and officials were required to make difficult judgments about how to remediate and when to reopen Senate office buildings. Similarly, your work is and would be crucial if a radiological incident were to occur in the U.S. The public would need your expertise and you would need the confidence of the public to succeed.

In conclusion, I want to thank you for the valuable work you do to provide the sound science that serves as a foundation for policy making and public protection. I appreciate the opportunity to be here and I look forward to hearing the results of today's session.

I would welcome any questions or comments you may have.

August 22, 2007

MEMORANDUM TO: Margaret M. Doane, Director
Office of International Programs

Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-07-0118 - STATUS REPORT ON THE PROGRESS OF THE G-8 COUNTRIES IN IMPLEMENTING THE IAEA CODE OF CONDUCT (SRM-M050602A)

The Commission has approved the staff's recommendation to discontinue the periodic updates on the G-8 countries' implementation of the IAEA Code of Conduct on the Safety and Security of Radioactive Sources. The Commission believes it would be useful to understand international practices and international benchmarks with regard to its decisions next year on expanding the National Source Tracking System to Category 3 sources, and whether to expand specific licensing to Category 3.5 sources. For this reason, the Office of International Programs should support the Office of State and Federal Materials and Environmental Management Program's efforts to improve NRC's byproduct materials licensing and oversight programs by obtaining information on other nations' Code implementation to serve as benchmarks for the NRC effort.

Previously, in the Staff Requirements Memorandum for SECY-06-0171, "Analysis of 10 CFR Part 110, Appendix P Implementation Issues," the staff was directed to keep the Commission informed of "methods and opportunities to achieve greater and more uniform adoption of the provisions of the Code of Conduct by other countries." The Commission appreciates the staff's continuing efforts to provide the Commission with information on other countries' implementation of the Code of Conduct, and looks forward to any recommendations the staff may have for enhancing the agency's approach to implementing the same.

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

August 15, 2007

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - COMSECY-07-0025 - SEMIANNUAL UPDATES OF THE LESSONS-LEARNED PROGRAM

The Commission has approved the Lessons-Learned Oversight Board's recommendation for staff to provide annual updates of the Lessons-Learned Program (LLP) beginning in August 2008. Efforts should be made to increase staff awareness of the LLP, and the LLP should include discussions of successes and not just failures.

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

August 10, 2007

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-07-0082 - RULEMAKING TO MAKE RISK-INFORMED CHANGES TO LOSS-OF-COOLANT ACCIDENT TECHNICAL REQUIREMENTS; 10 CFR 50.46A, "ALTERNATIVE ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER NUCLEAR POWER REACTORS"

The Commission has approved a modified Option 3. The Commission agrees with the staff recommendation that the rulemaking be given a medium priority, but that the 50.46a and 50.46b rulemakings should be given a higher priority than the Pressurized Thermal Shock (PTS) rulemaking effort, and that the LOOP-LOCA rulemaking priority should be lower than the one for PTS. The Commission expects the staff to make progress on the 50.46 rules and to apply resources to the effort in FY 2008. Additionally:

1. The final rule should require licensees to justify that the generic results in the revised NUREG-1829, "Estimating Loss-of-Coolant Accident Frequencies Through the Elicitation Process," are applicable to their individual plants. The staff should develop regulatory guidance that will provide a method for establishing this justification.

2. If a 50.46a rule is completed prior to revising 50.46b and license amendment applications using 50.46a are submitted, staff should ensure an appropriate safety margin for fuel clad integrity that will have high assurance of meeting the final acceptance criteria.

3. The staff should strengthen the assurance of defense-in-depth for breaks beyond the transition break size (TBS). In particular, the rule, in consonance with other applicable regulations, should not permit removal of existing mitigation equipment or alteration of mitigation capability without prior staff approval. However, equipment whose only function is to mitigate beyond TBS LOCAs should be permitted reclassification as non-safety. Other requirements to strengthen the assurance of defense in depth may include those already proposed by the ACRS, or alternatives developed by the staff including specific limits on the availability of equipment used to mitigate breaks beyond the TBS.

4. The revision of NUREG-1829 should be completed before the rule is finalized. The staff should incorporate, as appropriate, the changes resulting from the resolution of public comments to the final rule.

5. In order to more closely follow the approach presented in Regulatory Guide 1.174, the staff should modify the proposed rule to ensure that any changes under this rule be further restricted to very small risk increases, notwithstanding the fact that they would otherwise be permitted under 50.59. Therefore, staff should add the word "very" before the word "small" in section (f)(1)(i) so that it reads "...the total increase in core damage frequency and large early release frequency are very small and the overall risk remains small..." or make other changes as appropriate to achieve the above objective.

6. The staff should evaluate various approaches for enhancing the rule with requirements for improved leak detection methods and thorough diagnosis of observed defects for generic implications. Regulatory guidance for this rule should ensure that potentially significant defects are fully evaluated for generic implications. The staff should interact with the appropriate ASME codes and standards committees to explore ways to improve regulatory confidence that greater-than-TBS LOCAs remain an insignificant contributor to risk. Also, the staff should continue research activities that lead toward an improved understanding of aging mechanisms and fracture mechanics in support of more accurately predicting the likelihood of pipe breaks.

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

July 25, 2007

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-07-0096 - POSSIBLE REACTIVATION OF CONSTRUCTION AND LICENSING ACTIVITIES FOR THE WATTS BAR NUCLEAR PLANT UNIT 2

The Commission has approved the staff's recommendations for the licensing and inspection program approach to be used for Watts Bar, Unit 2 if the Tennessee Valley Authority (TVA) decides to reactivate Unit 2 construction, subject to the comments noted below.

Should TVA decide to reactivate its Operating License application for Watts Bar Unit 2, the staff should issue a further notice of opportunity for hearing on the application.

The staff should keep the Commission informed of any significant issues that arise concerning the implementation of the Commission policy on deferred plants.

The Commission supports a licensing review approach that employs the current licensing basis for Unit 1 as the reference basis for the review and licensing of Unit 2. Further, TVA and the NRC staff should review any exemptions, reliefs, and other actions which were specifically granted for Unit 1 to determine whether the same allowance is appropriate for Unit 2. Significant changes to that licensing approach would be allowed where the existing backfit rule would be met or as necessary to support dual unit operation. The staff should encourage the licensee to adopt updated standards for Unit 2 where it would not significantly detract from design and operational consistency between Units 1 and 2.

There are current generic safety issues at the resolution stage, such as GSI-191 or security issues, that will be much easier to resolve before plant operation. The staff and TVA should, during the licensing period, look for opportunities to resolve such issues where the unirradiated state of Watts Bar 2 makes the issue easier to resolve than at Watts Bar 1.

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

August 22, 2007

The Honorable James M. Inhofe
Ranking Member, Committee on
Environment and Public Works
United States Senate
Washington, D.C. 20510

Dear Senator Inhofe:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter of July 13, 2007, regarding your concerns about the agency's policy of withholding information from the public, specifically in regard to a significant event at Nuclear Fuel Services (NFS). The Commission is currently reconsidering its policy and criteria for withholding information from the public that is related to regulatory activities at the BWX Technologies (BWXT) and NFS nuclear facilities.

The Commission's goal is to strike an appropriate balance between a regulatory process that is open to the public and the protection from disclosure of sensitive information which would be helpful to potential adversaries. As an initial step, on July 19, 2007, the agency made publicly available a Confirmatory Order issued to NFS on February 21, 2007, that was a result of an Enforcement Alternative Dispute Resolution process. This Order will be published in the *Federal Register* to provide an opportunity for members of the public to request a hearing. The Commission has also published on the NRC website a redacted transcript and staff requirements memorandum from a closed meeting held on May 30, 2007, between the Commission and NFS management.

I want to assure you that maintaining the public's confidence in the NRC's safety role is of the highest priority to the Commission. The NRC staff is working very hard to protect public health and safety while keeping agency processes as transparent as possible.

I appreciate the opportunity to respond to your letter. Please let me know if you wish to discuss this matter further.

Sincerely,

/RA/

Dale E. Klein

August 22, 2007

The Honorable Joe Barton
Ranking Member, Committee on
Energy and Commerce
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Barton:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter, dated July 3, 2007, regarding your concern that the NRC's August 2004 policy concerning withholding information about Nuclear Fuel Services' (NFS') licensed activities from the public has resulted in withholding many documents of no security significance. You also expressed concern that persons who may be affected by a February 21, 2007 Confirmatory Order that amended the NFS license were not provided an opportunity to request a hearing because the Order was withheld under the policy.

The NRC, working with the Department of Energy's (DOE's) Office of Naval Reactors, is reconsidering its August 2004 policy and the criteria used for withholding from public disclosure information deemed to be security related for those fuel cycle facilities where NRC and DOE's Office of Naval Reactors have a role, which includes NFS. Consistent with this effort, the NRC has made publicly available a number of recent documents related to NFS, including the February 21, 2007 Confirmatory Order amending NFS' license, which was issued as a result of an Enforcement Alternate Dispute Resolution process. The Confirmatory Order was made publicly available on July 19, 2007, and is being published in the *Federal Register* to offer persons adversely affected by the Order an opportunity to request a hearing.

The NRC staff will update Mr. Dwight Cates of the Minority Committee staff on the status of these actions, as you requested.

Sincerely,

/RA/

Dale E. Klein

cc: The Honorable John D. Dingell

August 22, 2007

The Honorable Ed Whitfield
Ranking Member, Subcommittee on
Oversight and Investigations
Committee on Energy and Commerce
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Whitfield:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter, dated July 3, 2007, regarding your concern that the NRC's August 2004 policy concerning withholding information about Nuclear Fuel Services' (NFS') licensed activities from the public has resulted in withholding many documents of no security significance. You also expressed concern that persons who may be affected by a February 21, 2007 Confirmatory Order that amended the NFS license were not provided an opportunity to request a hearing because the Order was withheld under the policy.

The NRC, working with the Department of Energy's (DOE's) Office of Naval Reactors, is reconsidering its August 2004 policy and the criteria used for withholding from public disclosure information deemed to be security related for those fuel cycle facilities where NRC and DOE's Office of Naval Reactors have a role, which includes NFS. Consistent with this effort, the NRC has made publicly available a number of recent documents related to NFS, including the February 21, 2007 Confirmatory Order amending NFS' license, which was issued as a result of an Enforcement Alternate Dispute Resolution process. The Confirmatory Order was made publicly available on July 19, 2007, and is being published in the *Federal Register* to offer persons adversely affected by the Order an opportunity to request a hearing.

The NRC staff will update Mr. Dwight Cates of the Minority Committee staff on the status of these actions, as you requested.

Sincerely,

/RA/

Dale E. Klein

cc: The Honorable Bart Stupak

July 20, 2007

The Honorable Bennie G. Thompson
Chairman, Committee on Homeland Security
United States House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter of May 15, 2007, regarding the cyber-security posture of the Nation's nuclear power plants. Your letter expressed concerns regarding several issues, including the adequacy of NRC regulations to impose adequate cyber-security protections.

The NRC is committed to ensuring the continued protection of the public health and safety, the environment, and the secure use and management of radioactive materials. Shortly after the events of September 11, 2001, the NRC issued a number of new security requirements that, in part, required power reactor operators to implement measures to enhance their cyber-security. Recently, the NRC completed a rulemaking to amend nuclear power plant security requirements to include a cyber attack as a threat attribute against which nuclear power plants must defend. In addition, the NRC is currently developing a final rule which will require all nuclear power plants to establish cyber-security programs to protect any system that could adversely impact safety, security, or emergency preparedness of a facility.

Your letter also raised several specific concerns, based on Information Notice 2007-15, regarding the effects of a malfunction of non-safety-related controls connected to an internal plant data network. On August 19, 2006, the Browns Ferry Nuclear Plant Unit 3 was manually shutdown from 40 percent power as a result of the unplanned loss of both recirculation pumps. The loss of the non-safety-related recirculation pumps was due to a control system failure. The plant is designed to ensure that reactor safety systems remain capable of addressing component failures regardless of the cause. For this event, despite the non-safety component failure, all safety-related systems performed as designed and the operator followed established procedures to safely shutdown the reactor.

The licensee determined that the cause of the event was a malfunction of the recirculation pump variable frequency drive (VFD) microprocessor-based controller. The controller failure was attributed to excessive traffic on the internal control network. Since the control network is physically and electrically independent of networks that interface outside the plant, the NRC is confident that the failure was not the result of a cyber attack. As stated in the Information Notice, the licensee subsequently installed firewalls to prevent similar control network traffic from affecting the VFD controller. The NRC determined that these actions are appropriate to ensure plant safety and consistent with the agency's policy and requirements for digital instrumentation and control (I&C) systems.

The NRC issued Information Notice 2007-15 to notify nuclear plant operators about the Browns Ferry incident. The NRC expects that licensees will examine the information for applicability at their facilities and consider actions, as appropriate, to avoid similar problems. Additionally, the NRC's Office of Nuclear Security and Incident Response has been coordinating with other Federal partners and several National Laboratories to examine existing cyber-security vulnerabilities, specifically in the digital I&C area. NRC also works in close collaboration with the Department of Homeland Security and other Federal agencies to identify any emerging cyber-security threats. Information on the nature of an emerging threat and appropriate mitigative measures is readily conveyed to any NRC licensees that may be vulnerable to such emerging threats. For additional information regarding digital I&C, refer to http://www.nrc.gov/about-nrc/regulatory/research/digital.html. Similar steps are being taken to incorporate the lessons learned from this event and rulemaking efforts into the licensing of the next generation of reactor designs.

More specific responses to your questions are provided in the enclosure. If you have further questions or would like to arrange a briefing, please contact me.

Sincerely,

/RA/

Dale E. Klein

Enclosure:
As stated

Identical letter sent to:

The Honorable Bennie G. Thompson
Chairman, Committee on Homeland Security
United States House of Representatives
Washington, D.C. 20515

The Honorable James R. Langevin
Chairman, Subcommittee on Emerging Threats, Cybersecurity, and Science and Technology
Committee on Homeland Security
United States House of Representatives
Washington, D.C. 20515

NRC Response to May 15, 2007, Questions Regarding the Cyber-Security Posture of the Nation's Nuclear Power Plants

QUESTION 1: Has the NRC conclusively determined the source of the data storm described in Information Notice 2007-15?

RESPONSE: No. In this case, a review of plant data found that the safety systems functioned as designed, the cause identified was credible and was corrected, and no evidence of external influences was observed. The consideration of an external source for the failure was specifically reviewed. Based on the lack of remote access to the controllers, and the configuration and failure modes of the failed components, it was determined by the licensee and the NRC staff that the failure was not due to outside sources.

QUESTION 2: Does the NRC plan to exercise its authority under 10 C.F.R 50.65 to conduct an investigation of the incident at Browns Ferry?

RESPONSE: Yes. The NRC's "Maintenance Rule," 10 CFR 50.65, requires NRC nuclear power plant licensees to monitor the overall continuing effectiveness of their maintenance programs and take corrective action as necessary. The NRC routinely conducts inspections to verify that licensees comply with this rule. A detailed review of the August 2006 event at Browns Ferry was conducted by the NRC. After confirming that the reactor was shut down and that all safety systems performed as designed, the on-site NRC resident inspection staff provided details regarding plant status and performance of equipment and personnel to NRC management, event review staff, and risk analysts. These details were used to determine the level of agency response.

As part of the routine inspection program, the NRC will review and assess the adequacy of the licensee's corrective actions and provide a determination as to whether any additional NRC action for Browns Ferry is warranted.

QUESTION 3: In reviewing the incident, will the NRC determine what cyber-security policies and procedures the site followed, and what cyber-security assessments were performed?

RESPONSE: Yes. As part of the event follow-up, the NRC inspectors reviewed event assessments and corrective actions completed by the Browns Ferry staff to ensure that appropriate actions were taken consistent with NRC requirements and site procedures. The NRC's review of the August 2006 event found that the actions taken by Browns Ferry ensured that the safety and security of the facility were maintained.

Enclosure

QUESTION 4: How will future NRC regulations address the cyber-security interdependencies of non-safety and safety systems? What specific features will these systems contain?

RESPONSE: The NRC has already established a design requirement in 10 CFR Part 50 for safety systems to function in the event of a non-safety system failure. Section 50.55a(h) of 10 CFR requires that the safety system design shall be such that credible failures in, and consequential actions by, other systems shall not prevent the safety systems from performing their safety functions.

As stated previously, new security regulations in 10 CFR Part 73 will specifically identify cyber attack as a threat that must be protected against.

Lastly, plans are in place for NRC reviews of new reactor designs and safety systems to include security features against cyber threats, as well as physical threats.

QUESTION 5: Non-safety systems are not the only networked operations within a nuclear plant. As time passes, more and more safety systems will be networked and accessible online. How will future NRC regulations address the rise of networked safety systems?

RESPONSE: Regulations for new reactors and safety systems already exist and require that the safety system designs, which include safety-related networks, shall be such that credible failures in, and consequential actions by, other systems will not prevent them from performing their safety functions.

Our response to the previous question (Question 4) addresses the safety and security of networked safety systems. For deployment of digital systems in nuclear facilities (e.g., new reactors), upgrades to operating reactors and fuel cycle facilities, the NRC will evaluate these networked safety systems to determine whether they comply with the relevant regulations. This will include NRC requirements for the separation of safety and non-safety systems as well as the requirement that failure of other systems will not impair the ability of safety systems to perform their safety functions, as mentioned above.

Once the proposed rule for cyber-security requirements is published in final form and regulatory guidance has been issued, the NRC will incorporate the requirements into the baseline inspection program to help ensure that all nuclear power plants maintain adequate protection against external sources.

The current regulatory review plan for existing and future nuclear safety-related systems is to ensure that no modem or other means of connectivity provides access to safety systems via external networks.

QUESTION 6: How has the NRC reached out to the non-nuclear control system community to solicit feedback to the proposed rulemaking? What role have these experts played in assisting the NRC in developing regulations for nuclear plants?

RESPONSE: The NRC's Design Basis Threat rule (10 CFR 73.1), which requires power reactor licensees to be able to defend against a cyber attack with high assurance, was published for public comment in the *Federal Register* from November 7, 2005 through February 22, 2006. The NRC received over 900 comments from the public regarding that rulemaking, including a comment on cyber-security. In addition, the NRC also published a proposed rule that would revise the physical security requirements for all nuclear power plants and includes provisions related to cyber-security. That proposed rule was published in the *Federal Register* for public comment from October 26, 2006 through March 26, 2007, and three public meetings to solicit comments and resolve questions were held between November 2006 and February 2007. The NRC has received, and is reviewing, numerous comments on the proposed cyber-security requirements.

Our efforts also include outreach to our Federal partners, including the Department of Homeland Security, to gain additional insights on cyber-security best practices as related to control and safety systems. As a result, the NRC intends to integrate these insights into the current rulemaking on cyber-security requirements and to develop associated regulatory guidance to support rule implementation.

Recently, the NRC has established a Digital Information & Control Steering Committee. One of the functions of the Digital Information & Control Steering Committee has been to facilitate consistent resolution of issues across multiple NRC organizations and to ensure timely resolution of strategic and policy issues by forming task working groups assigned to individual key issues. These task working group meetings, as well as the Steering Committee meetings, are public meetings which solicit and involve participation by nuclear as well as commercial control system designers and users.

QUESTION 7: The NRC concluded that the remote access capability of the VFD controllers at the Browns Ferry plant was removed prior to the incident. However, it would seem that there exists a strong possibility that other plants are utilizing remotely accessible controllers, making them vulnerable to remote exploitation. Has the NRC conducted a review of plants to determine which ones are using remotely accessible controllers?

RESPONSE: Yes. Following the events of September 11, 2001, the NRC issued two orders, EA-02-026 *Interim Compensatory Measures*, and EA-03-086

Revised Design Basis Threat, which included requirements for power reactor licensees to implement measures to enhance cyber-security. In addition, the NRC has published NRC NUREG/CR-6847, *Cyber-security Self-Assessment Method for U.S. Nuclear Power Plants*. The NUREG forms the basis for an industry-generated guidance document that is being voluntarily implemented by plant owners to assess comprehensively their facilities for cyber-security vulnerabilities and to establish mitigation strategies. In May 2007, all plants completed site-specific self-assessments. The staff will review the self-assessments to identify if and how remotely accessible controllers are being utilized.

In 2004, the NRC completed a cyber-security assessment at four nuclear plants. The results of this assessment are summarized in NUREG/CR-6852, *An Examination of Cyber-security at Several U.S. Nuclear Power Plants*.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

Dale E. Klein, Chairman
Edward McGaffigan, Jr.
Gregory B. Jaczko
Peter B. Lyons

DOCKETED 08/30/07
SERVED 08/30/07

In the Matter of SOUTHERN NUCLEAR OPERATING CO. (Early Site Permit for Vogtle ESP Site)	Docket No. 52-011-ESP

CLI-07-24

MEMORANDUM AND ORDER

The Commission delegated this early site permit (ESP) application proceeding to the Licensing Board to conduct the mandatory hearing and make the findings required under 10 C.F.R. § 2.104(b).[1] Subsequently, the Commission accepted a proposal from the Combined License Review Task Force that the Commission itself conduct the mandatory hearings for combined operating license applications.[2] In view of this Commission decision, the Board certified the following question to the Commission, pursuant to 10 C.F.R. § 2.319(l):

> Does the Commission wish this Licensing Board to conduct the Vogtle ESP mandatory hearing?[3]

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[1] *See Southern Nuclear Operating Company; Notice of Hearing and Opportunity to Petition for Leave to Intervene on an Early Site Permit for the Vogtle ESP Site*, 71 Fed. Reg. 60,195, 60,195-96 (Oct. 12, 2006).

[2] *See* Memorandum from Annette L. Vietti-Cook, Secretary, to Luis A. Reyes, Executive Director for Operations, et al., Staff Requirements - COMDEK-07-0001/COMJSM-07-0001 - Report of the Combined License Review Task Force (June 22, 2007) at 1.

[3] *Memorandum (Certifying Question Regarding Conduct of Mandatory Hearing)* (July 12, 2007) at 3, unpublished Licensing Board decision.

In response to this certified question, the Commission affirms its original delegation to the Board and asks the Board to conduct the mandatory hearing in this proceeding, as originally planned.

IT IS SO ORDERED.

For the Commission

/RA/

Annette L. Vietti-Cook
Secretary of the Commission

Dated at Rockville, Maryland,
this 30th day of August, 2007

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

Dale E. Klein, Chairman
Edward McGaffigan, Jr.
Gregory B. Jaczko
Peter B. Lyons

DOCKETED
USNRC

August 2, 2007 (1:33pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

SERVED August 2, 2007

In the Matter of	)	
DOMINION NUCLEAR NORTH ANNA, LLC	)	Docket No. 52-008-ESP
(Early Site Permit for North Anna ESP Site)	)	

CLI-07-23

ORDER

On June 29, 2007, a split Atomic Safety and Licensing Board issued its Initial Decision, LBP-07-9, in the "mandatory hearing" portion of this adjudication addressing Dominion Nuclear North Anna, LLC's application for an Early Site Permit (ESP). "Before the Early Site Permit . . . can be made effective, the Commission must review and approve the Licensing Board's Initial Decision authorizing its issuance."[1] Here, the majority of the Board approved issuance of the North Anna ESP, while the dissenting judge would have denied the ESP due to insufficiencies in the NRC Staff's and Dominion's examinations of alternative sites and alternative design features related to water conservation. The Initial Decision recommended that the Commission consider the following issues:[2]

> (i) Did the Staff's environmental justice analysis in the FEIS follow the "greater detail" guidance set forth in the Commission's Environmental Justice Policy Statement?

[1] *System Energy Resources, Inc.* (Early Site Permit for Grand Gulf ESP Site), CLI-07-7, 65 NRC 122 (2007) (citing 10 C.F.R. § 2.340(f)).

[2] *See* slip op. at 91-107.

(ii) How do the NRC's multiple radiation protection standards (and the ALARA concept) apply to new reactors that are proposed to be added at a site with pre-existing nuclear reactors and radiological effluents?

(iii) How should the Commission apply its statement prohibiting partial ESPs and ESPs where adequate information is not available to a situation where significant elements of the plant parameter envelope for the ESP are missing and numerous siting issues are unresolved due to lack of information?

We invite the NRC Staff and Dominion to submit initial and reply briefs addressing the questions above, the issues of alternative sites and alternative design features raised in Judge Karlin's dissent, the suggestions in LBP-07-9 regarding perceived deficiencies in the NRC Staff's and Dominion's evidence and arguments,[3] and any other issues that, in the parties' view, warrant comment. Each initial brief shall be no longer than 40 pages (exclusive of title page, table of contents, and table of authorities) and shall be filed within 21 calendar days of the date of this order. Each reply brief shall be no longer than 20 pages and shall be filed within 14 days thereafter.[4]

IT IS SO ORDERED.

For the Commission

Kenneth R. Hart
Acting Secretary of the Commission

Dated at Rockville, Maryland,
this 2nd day of August, 2007.

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[3] *See* slip op. at 28-36 (hydrology), 45-46 (tritium), 56-61 (alternative sites); dissenting slip op. at 2-11 (alternative sites), 1 & 11-12 (alternative design criteria).

[4] Due to the potentially large number of issues requiring discussion, the Commission will entertain motions to expand these page limits if good cause can be shown. We urge the parties, however, to keep their briefs as short as possible, consistent with providing meaningful responses to our inquiry.

Commissioner Gregory B. Jaczko respectfully concurring:

I approve of this order and the request for briefs on these difficult and important questions. I offer a concurring opinion because I believe the Commission should have also specifically requested amicus briefs on these issues. The answers to these questions will impact the early site permit process for future applicants and participants. Thus, I believe the ultimate Commission decision would be better informed with a wider variety of interested stakeholder perspectives on these issues to aid the Commission in better understanding how best to improve the ESP process.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
DOMINION NUCLEAR NORTH ANNA, LLC	)	Docket No. 52-008-ESP
	)	
(Early Site Permit for North Anna ESP Site)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing COMMISSION ORDER (CLI-07-23) have been served upon the following persons by electronic mail this date, followed by deposit of paper copies in the U.S. mail, first class, and NRC internal mail.

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Docket No. 52-008-ESP
COMMISSION ORDER (CLI-07-23)

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Evangeline S. Ngbea
Office of the Secretary of the Commission

Dated at Rockville, Maryland,
this 2nd day of August 2007

Recently Issued Significant Reactor Enforcement Actions

Nebraska Public Power District (Cooper Nuclear Station) EA-07-090

On August 17, 2007, a Notice of Violation was issued for a violation associated with a White Significance Determination Finding involving a violation 10 CFR Part 50, Appendix B, Criterion XVI. The licensee failed to establish measures to promptly identify and correct a significant condition adverse to quality (SCAQ) and failed to assure that the cause of a SCAQ was determined and corrective action taken to preclude repetition. Specifically, the licensee's inadequate procedural guidance for evaluating the suitability of parts used in safety related applications presented an opportunity in which the licensee failed to promptly identify a defective voltage regulator circuit board used in Emergency Diesel Generator (EDG) 2 prior to its installation. Following installation of the defective EDG 2 voltage regulator circuit board, the licensee failed to determine the cause of two high voltage conditions, and failed to take corrective action to preclude repetition. As a result, an additional high voltage condition occurred resulting in a failure of EDG 2.

Southern Nuclear Operating Company (Joseph M. Farley Nuclear Plant) EA-07-155

On August 17, 2007, parallel White finding was issued to Southern Nuclear Operating Company as a result of inspections at the Joseph M. Farley Nuclear Plant. The parallel White finding was identified during a supplemental inspection to assess the licensee's evaluation associated with unreliability and unavailability reporting in the Support Cooling Water Systems Performance Indicator (PI) within the Mitigating Systems Performance Index (MSPI). Failures of the licensee's existing safety-related breakers associated with this PI predominantly contributed to the indicator crossing the threshold to White in the second quarter of 2006. This PI was subsequently reported Green in the 3rd quarter of 2006. The supplemental inspection for the White PI identified significant weaknesses related to the thoroughness and quality of several root cause evaluations that challenged the licensee's ability to implement effective overall corrective actions. The licensee's evaluations of the individual failures that contributed to the White PI did not effectively review for systemic aspects of circuit breaker failures. In addition, more recent problems were identified concerning the thoroughness of design reviews for the installation of new breakers. Based on these NRC-identified weaknesses, a parallel PI inspection finding (White) was opened to allow the NRC to continue to monitor activities in this area.

FirstEnergy Nuclear Operating Company (Davis-Besse Nuclear Plant, Perry Nuclear Power Plant, Beaver Valley Nuclear Plant, Units 1 and 2) EA-07-199

On August 15, 2007, a Confirmatory Order (Effective Immediately) was issued to FirstEnergy Nuclear Operating Company (FENOC) to formalize commitments made by FENOC following the NRC staff's issuance of a Demand for Information (DFI) on May 14, 2007. The DFI was issued in response to the information provided by FENOC relative to its re-analysis of the time line and root causes for the 2002 Davis-Besse reactor pressure vessel head degradation event following its receipt of a report prepared by Exponent Failure Analysis Associates and Altran Solutions Corporation (Exponent). On June 13, 2007, FENOC provided its response to the DFI. On July 16, 2007, FENOC provided a supplemental response to the DFI which provided additional detail regarding the planned implementation of commitments established in its June 13, 2007, response to the DFI.

Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 & 3) EA-07-189

On July 30, 2007, an Immediately Effective Order was issued to Entergy Nuclear Operations, Inc., to ensure compliance with the regulations and implementation of the requirements of the Energy Policy Act of 2005. The Order supplemented the requirements of NRC Confirmatory Order (EA-05-190) which required, in part, a backup power system for the Emergency Notification System (ENS). The Confirmatory Order was issued to the Indian Point Nuclear Generating Unit Nos. 2 and 3 on January 31, 2006, and was amended by the NRC on January 23, 2007, extending the implementation date for the required ENS backup power system to April 15, 2007. The requirements of the January 31, 2006, Confirmatory Order remain in effect except as specifically modified or supplemented by this Order.

Inside NRC

Volume 29 / Number 18 / September 3, 2007

Hyperion seeks to gain license for portable reactor by 2015

Hyperion Power Generation Inc. officials are aiming to obtain a manufacturing license by 2015 for a small, portable power reactor fueled by uranium hydride.

The company plans to request an NRC preapplication review beginning in fiscal 2009 for a manufacturing license, company representatives told agency staff at an August 22 meeting.

The new reactor, based on a concept developed at Los Alamos National Laboratory and licensed to the Santa Fe, New Mexico company, would consist of a small, portable "sealed module" containing a reactor core of uranium hydride powder about 1.5 meters (4.9 feet) in diameter, Otis "Pete" Peterson of HPG, a retired LANL scientist, said in his presentation.

Peterson said that he had been "peddling" his idea for a uranium hydride reactor among his colleagues at Los Alamos for some time, telling them to "shoot this down" with technical objections. But "after about an hour of debate, they finally gave up, so I kept peddling," he said. Fission in the reactor would be moderated by variations in the amount of hydrogen gas, generated by the uranium hydride, present in the reactor core, with hydrogen moving back and forth between the core and external storage trays of depleted uranium, Peterson said. Heat would be removed from the core by about 1,600 steel pipes containing potassium and could be utilized directly as process heat for industrial applications or in a conventional steam turbine to generate electricity, he said.

Higher reactor temperatures would cause more hydrogen to move to the storage trays, and lower temperatures would draw the hydrogen back in, creating a "natural relaxation oscillator" which would moderate fission in the reactor, he said. Such a design is "completely self-controlling, self-regulating, and safe," requiring "no mechanical moving parts" and "minimum human oversight," Peterson said in his presentation. Preliminary calculations show that "transients damp out in a few minutes," he said. Even "order of magnitude" changes in the rate of reactivity "do not affect stability" of the reactor, Peterson said. Hydride fuels are "safe from temperature over-excursions," and power pulses from 1 MW to 20 GW would be "self-terminated" in 10 to 20 milliseconds, he said. The safety of uranium hydride fuel has been "demonstrated at over 60 installations," including Triga research reactors, he said.

HPG Vice President Deborah Blackwell said at the meeting that each small reactor would be buried in an underground vault, and could generate 67 megawatts of thermal power (27 MW of electricity) using uranium hydride enriched to about 5% for an operational life of five years. The reactor could then be dug up and returned to the factory for refueling, which "greatly hampers proliferation attempts," Blackwell said.

The company estimates capital costs of $20 million per unit for purely thermal applications and $37 million (about $1,380/kW) for electricity generation, making its economics "very attractive," Blackwell said. She foresees potential applications in recovering oil from tar and sands, energy resources that she said are currently uneconomic due to the amount of energy required to recover them.
The HPG reactor could produce heat at a cost of about $3 per million BTU, compared with $7 to $14 per million BTU for natural gas, resulting in five-year savings of approximately $1 billion-$2 billion per heavy oil field using the reactor, Blackwell said. Other promising applications include the production of clean water in developing countries, she said.

Staff questions
Don Helton of NRC's Office of Nuclear Regulatory Research told company officials that whether HPG's design is "formally aligned with a customer" when it applies for its manufacturing license "may affect review priority" given to reviewing the application. John "Grizz" Read, HPG's CEO, said that the company had already received expressions of interest but did not name any potential customers.

HPG would need to receive both a manufacturing license and design certification from NRC, Helton said at the meeting. NRC has previously issued only one manufacturing license. In the early days of the agency, a vendor received a license to manufacture floating nuclear power plants, but that license was later withdrawn when the potential customer backed out of the project, NRC staffers said.

Agency staff must consider what priority to give to review of the design relative to other activities, such as DOE's Next Generation Nuclear Plant and Global Nuclear Energy Partnership initiatives and NRC's preapplication review of a pebble bed modular reactor, Helton said. "Availability of technical reviewers is an important constraint," and "resources for non-LWR efforts [are] expected to decrease in [the] next few years," he said.

Other NRC staffers at the meeting asked about a wide variety of safety, transportation, and security issues, but said they recognized that the design was still at too conceptual a stage for HPG to provide definitive answers. "At this point, detailed analysis of materials, thermodynamic, and chemical issues," as well as accident analysis, "has not been performed," but "this analysis, and related experimentation (as necessary), is planned" by HPG, Helton said in an August 29 written meeting summary.

"What we don't know is enormous," and "we don't know if we're ever going to actually produce one of these" reactors, Read acknowledged at the meeting. The issues raised by NRC staff "are the kind of things that could stop us from going to market," so the opportunity to discuss them early is appreciated, Read said.

—Steven Dolley, Washington

Licensees are not to exceed licensed power level, NRC says

After documenting several instances in which licensees intentionally exceeded the maximum power level specified in their plant's operating license, NRC issued August 27 a generic communications reminding licensees that they must adhere to the limits in their license.

The regulatory issue summary, RIS 2007-21, focuses on whether a licensee purposely exceeds its licensed limit. NRC staffers said that problems it has documented, going back to 1989, stem from licensees adopting internal NRC enforcement guidance into plant operating procedures. The guidance to NRC inspectors was issued in an August 22, 1980 memorandum from Edward Jordan, then the assistant director for technical programs in the reactor operations inspection division.

In the memo, Jordan said regulators had been having discussions with plant operators since at least the early 1970s about what constitutes a "full, steady-state licensed power level." For the purpose of enforcing a plant's maximum licensed power, Jordan said, plant operators should average the thermal power flow over an eight hour period to determine whether the licensed power level was exceeded.
The "Jordan memo," as it became known, was used by inspectors to decide when to pursue enforcement. The memo was later incorporated into NRC inspection procedure 61706. The guidance instructs inspectors to check that a plant's "average power level" over an eight-hour shift doesn't exceed the "full, steady-state licensed power level" or similar wording in a plant's license.

The inspection manual incorporated language from the Jordan memo as guidance on reviewing a licensee's core thermal power evaluation. "It is permissible to briefly exceed the 'full, steady-state licensed power level' by as much as 2% for as long as 15 minutes," the inspection guidance states. "In no case should 102% power be exceeded, but lesser power 'excursions' for longer periods should be allowed," it said.

As an example, the guidance said that inspectors would not need to take action if there was a "1% excess for 30 minutes and a 1/2% for 1 hour." Moreover, there were no restrictions on the "excursions" as long as the eight-hour average showed no "abuse of this allowance."

Mary Ann Ashley, an enforcement coordinator in the NRC Office of Nuclear Reactor Regulation, told industry representatives at an August 22 meeting that guidance was solely to help inspectors determine when they should — and should not — take enforcement action because it is widely recognized that maintaining 100% power is "virtually impossible."

But inspectors began finding instances where the enforcement guidance had been incorporated into operating procedures at sites. "What it did was set up a dynamic where operating procedures said, 'It's permissible for you to exceed your maximum power level.' That was in direct conflict with the operating license for maximum power level," Ashley said.

Ashley said RIS 2007-21 will supersede the Jordan memo. "We have found the current ROP [reactor oversight process] tools are adequate to identify issues and to correctly disposition instances where the maximum power level may be exceeded."

The RIS states that NRC never intended the inspector guidance to become a "staff position." It said the staff position is that licensees must follow the maximum thermal power limit stated in the operating license.

Ashley said NRC inspectors identified four instances in which the agency's enforcement guidance had been incorporated into plant operational guidance: at Sequoyah in 1989; Kewaunee in 2005; Dresden in 2006; and Wolf Creek in 2007. The RIS does not have a description of the Wolf Creek incident.

But in the cases of Kewaunee and Dresden, NRC inspectors found the maximum power level was "only marginally exceeded" and the "safety significance was very low," the RIS said. However, it said, both licensees were in violation of their facility operating license. At Sequoyah, operators had "intentionally" run the reactors "above the licensed limit of rated thermal power (i.e., over 100 percent) for brief periods," the RIS said.

Ashley said that exceeding the maximum power level for short periods of time was not a safety significant issue. But it does raise a compliance issue, she said.

"It's not that the guidance is going away," she told industry officials at last month's meeting. "What we are trying to do is recognize that a particular set of guidance that was provided for the use by inspectors has somehow made its way into the licensee arena, and they are using it and using it inappropriately."

She said NRC wants to remove the guidance from public use. NRC's message is that the industry must stick to its licensing basis, she said. But she also said the staff will be looking at whether there is a way to establish new guidance for inspectors to monitor whether licensees are intentionally exceeding the maximum power limits in their license.
—Jenny Weil, Washington

NRC staff extends deadline for weld inspections at nine PWRs

NRC last month told the operators of nine PWRs that they can wait until their spring 2008 refueling outages to conduct inspections and mitigations of dissimilar-metal welds in their pressurizers, according to an NRC staffer. Without the extensions, the nine units — Braidwood-2, Comanche Peak-2, Diablo Canyon-2, Palo Verde-2, Seabrook, South Texas-1, Summer, Vogtle-1, and Waterford-3 — would have had to shut down by December 31 to do the required work.

The decision — conveyed to the operators of the nine units in an August 23 conference call — settles a long-running question about the appropriate inspection regime for Alloy 82/182 welds between dissimilar metals. However, comments last month by industry and NRC officials suggested that a remaining detail of the follow-up could be divisive.

The year-end deadline originally was set by the Electric Power Research Institute's Materials Reliability Program, in a document known as MRP-139. But, under the industry's selfpolicing regime, some units applied for and received "deviations" from the industry requirement. NRC questioned the extensions, particularly after the discovery last fall of large circumferential cracks in pressurizer welds at Wolf Creek. In March, the regulators sent out confirmatory letters establishing the December 31 deadline (INRC, 19 March, 12). But they left open the possibility of extending the deadline if a detailed technical study, known as a finite element analysis, that a group of industry experts was undertaking showed that the assumptions on which the agency had based its decisions were overly conservative.

One of NRC's major concerns was that, under certain scenarios, there would be only a short interval from beginning of leakage to rupture, giving operators little warning of potential rupture. But last month, in the completed study and at an August 9 meeting, MRP officials presented data to address the NRC's concerns.

The regulators then carried out their own safety assessment. The written safety assessment is expected to be issued in the coming weeks, the NRC staffer said.

The agency's favorable decision did not come as a surprise (INRC, 20 Aug., 10). At the August 9 meeting and elsewhere, NRC staffers had said the results of their confirmatory analyses had been consistent with the MRP's.

Jim Riley, the Nuclear Energy Institute's director of engineering, said August 30 that the project was a good example of NRC and the industry "maintaining coordination and communication" while carrying out independent work.

With that approach, potential differences could be worked at the beginning of the process rather than the end, he said. In an August 16 interview, John Grobe, NRC's associate director for engineering and safety systems, also said the cooperation had been "very positive." A slide in NRC's presentation at the August 9 meeting said that the recent work had "greatly increased" the understanding of primary water stress corrosion cracking in dissimilar-metal pressurizer welds.

However, one issue that surfaced during the long process of developing an inspection regime for the dissimilar-metal welds apparently remains unresolved. In the August 16 interview, Grobe said NRC did not anticipate that so many operators would request deviations from the industry imposed deadline. He said he expected EPRI would now formalize a mechanism for communicating such deviations to the NRC. Riley said the Nuclear Strategic Issues Advisory Group — made up of chief nuclear officers and senior vendor officials — had discussed the issue but had not reached any conclusions.

A group of technical experts is going to study the issue, but it should be resolved fairly quickly, he said. However, he said, the issue is not just whether the reporting on the deviations to NRC should come from individual licensees or from an industry group, such as the MRP or NEI, but whether the deviations are reported at all. In August 16 and 30 interviews, he said such reporting is not required and goes beyond the American Society of Mechanical Engineers code.

The potential dispute could have broad implications, since the industry envisions applying the self-policing regime not just to the inspections of dissimilar-metal welds but to a much wider range of materials issues.

Some of the difference between NRC and the industry appears to come from the classification system the industry uses for recommendations under its regime, which was described in a document — NEI 03-08, "Guideline for the Management of Materials Issues" — issued four years ago. The highest level of recommendation is called "mandatory" and is "to be implemented at all plants where applicable," according to the document. Since then, industry officials have told NRC that deviations from mandatory recommendations have to be approved by a company executive and a third party.

In the August 16 interview, Grobe said the high number of MRP-139 deviations requested and approved "calls into question" the use of the term "mandatory." Riley said the choice of that term might have been "unfortunate." Recommendations in that category cover issues that are not absolute requirements but do deal with issues that constitute "the basic foundation" of what the industry guidelines are trying to accomplish and would not vary much from plant to plant, he said.

In the case of the pressurizer-weld inspections, industry officials wanted the work done quickly but realized that resources were limited, he said. They also realized that the outage schedules for some plants would lead them to ask for extensions of the December 31, 2007 deadline.

"That has been industry's position all along," Riley said. NEI has said previously that it does not plan to make MRP-139 into a "regulatory document," such as an NRC Inspection Procedure (INRC, 31 Oct. '05, 3).
—Daniel Horner, Washington

Staff and industry are 'converging' on operability determination criteria

The industry and NRC staff are converging on a mutually agreeable approach to determining the operability of certain classes of power reactor components that exhibit flaws and leakage. But several outstanding issues require clarification and further discussion, both sides said at a public meeting last week.

Industry told staff earlier this year that guidance provided in a regulatory issue summary, RIS-2005-20, was overly strict and could expose up to 37 units at 23 sites to the risk of "immediate" shutdown if leakage were detected in certain components (INRC, 28 May, 1). The agency issued new interim inspection guidance on the subject in July, specifying that "immediate determinations of operability should be based on a reasonable expectation of operability," and "prompt" operability determinations should be based on non-destructive examination, or NDE, measurements of the system (INRC, 9 July, 5).

Agency staff is now revising the inspection guidance. Draft revisions of Appendices C.11 and C.12 of the relevant chapter in NRC's inspection manual — which deal with flaw evaluation and operational leakage respectively — were issued in mid-August, and were the subject of an August 30 public meeting at NRC headquarters.

At the meeting, Michael Schoppman of the Nuclear Energy Institute said that licensees are now using the interim guidance and it has had "a positive impact." However, "NRC and industry need to isolate and reach consensus on key terms and definitions, walk through practical examples of field situations, [and] consider a formal 'process mapping' project to develop a flow chart and time line for compliance with regulatory requirements that apply to through-wall flaws," NEI said in talking points Schoppman tabled at the meeting.

Developing "consensus" definitions of such key phrases in the inspection guidance as "reasonable expectation of operation" is important but could prove difficult, Schoppman said.

Edmund Sullivan of NRC's Office of Nuclear Reactor Regulation said that one of main topics still to be resolved is how to deal with a situation where a particular component at a plant does not meet American Society of Mechanical Engineers, or ASME, code criteria but the operator's evaluation concludes that "structural integrity exists" and the component is still effectively "operable." In its comments on the draft inspection guidance, NEI said that "the inability to satisfy a code requirement requires an operability determination, but by itself may not constitute inoperability." Both sides acknowledged that for a component to not be in "conformance" with an ASME code criterion is not synonymous with its being "inoperable," but could not agree on criteria to determine operability in such situations.

One of the key issues in such situations is the amount of margin available for the component. Industry participants noted that ASME code requirements for some components provide a margin of a factor of two or three between the code requirement and calculated failure points. However, Sullivan responded, better understanding is needed of how much of the margin built into code requirements is there "for good safety and engineering reasons" before that margin is reduced by determinations that a component is out of conformance but still operable.

Another issue, Sullivan said in his presentation at the meeting, is that "substantial operating experience with pressure boundary leakage and degradation mechanisms in [a] leaking system [is] needed to establish a reasonable expectation of operability," but "this information is not generally available," and "detailed NDE data ... would take more time to obtain than [the] time allocated for an immediate operability determination." Mike Milton of NEI said that plant engineers have considerable experience with making such determinations, and both industry and NRC staff agreed that developing a set of case studies would help provide clarification.

Sullivan said he is "not comfortable" with acceptance criteria "being totally open ended," resulting in "case-by-case" standards for operability determinations. Common criteria should be developed, and perhaps implemented in industry guidance currently being developed by NEI. Revision two of an NEI white paper on the topic was sent to NRC in May, though it is "a vehicle for discussion at this point" rather than a document for which industry seeks formal NRC endorsement, Schoppman said.

Both sides agreed at the end of the meeting that a number of substantive issues remained but significant progress had been made. Another meeting, possibly a teleconference, will be scheduled this fall to continue the dialog. Sullivan said that NRC staff's goal is to finalize revisions to the inspection guidance by the end of the year.

Noting the complexity of ASME code requirements, one industry participant quipped: "This is harder than the IRS tax code."—***Steven Dolley, Washington***

Inside NRC

Volume 29 / Number 17 / August 20, 2007

Risk-informed initiatives to go forward, at measured pace

A majority of the NRC commissioners agreed this month to keep moving ahead on the development of three risk-informed initiatives and the revision to another. It told the staff to assign a "medium priority" for establishing a voluntary alternative approach to emergency core cooling requirements and for updating existing fuel cladding performance criteria. A lower priority for the staff should be revisions to the pressurized thermal shock, or PTS, rulemaking, and below that should be efforts to risk-inform the current requirements for consideration of a loss-of-coolant accident, or LOCA, coincident with a loss of offsite power.

Although the commissioners finished voting in June, before former Commissioner Jeffrey Merrifield's term ended, it took more than a month to reach an agreement on a plan of action for the initiatives. In the end, NRC Chairman Dale Klein and Commissioners Edward McGaffigan and Peter Lyons decided to modify the staff's preferred option by slowly making progress on the initiatives, but not totally shelving work over the next year. Under its recommended option, the staff would have determined the level of effort based on its rulemaking prioritization process.

One NRC staffer said that could have resulted in no work being done on the initiatives in 2008. "The commission said, 'We agree; it's not high priority.

But we don't want it to fall off the end of the table either. So call it a 'medium,' but make sure you spend some effort on it in 2008, and make some progress," the staffer said.

McGaffigan said August 17 that the vote means that work on the rulemakings "will not be done under any circumstance in fiscal year 2008, and at best, sometime in fiscal year 2009."

McGaffigan, who was initially the only commissioner to vote for the go-slow approach, said in his May 22 vote sheet that he had misgivings about what an optional rule with a redefined large-break LOCA design basis accident would achieve because of its heavy reliance on "beyond state-ofthe-art" probabilistic risk assessment, PRA, methods and data. He said the commission "has never found a way to require that all power reactor licensees have an up-to-date,. high quality internal- and external-initiating event all-mode PRA." While new plants will be required to have such PRAs, existing plants are not, he said.

McGaffigan said he believed it was "premature to rely upon calculations of risk significance to reduce or remove capabilities that provide defense-in-depth." He said he even considered voting to terminate the rule, an option favored by Commissioner Gregory Jaczko, who questioned whether the safety case has been made to "relax the current treatment of the large break loss-of-coolant accident."

A little more than a week before the commission finalized the approach for future risk-informed activities, the staff and industry representatives provided different views on how the effort should be prioritized. At the August 2 briefing, Klein asked for their list of top three issues the commission should focus on to move forward in the area of risk-informed regulation.

Both the industry and NRC staff prioritized implementation of risk-informed technical specifications for power reactors (INRC, 6 Aug., 1) and development of a final rule providing an alternative approach to core cooling requirements, known as 50.46(a) for its location in the Code of Federal Regulations. But they differed on timing for the rulemaking and other issues.

Anthony Pietrangelo, vice president for regulatory affairs at the Nuclear Energy Institute, said at the briefing that by risk-informing core cooling requirements, the 50.46(a) pro posed rule "underscores the whole risk-informed, performance-based concept" for regulation. Mark McBurnett, vice president at South Texas Project, said that 50.46(a) was at the top of his list of risk-informed regulatory priorities. The NRC staff is currently awaiting further guidance from the commission on how to proceed with 50.46(a) after the agency's Advisory Committee on Reactor Safeguards said last November that it could not approve issuance of the proposed rule without a number of modifications. Richard Dudley, NRC staff project manager for 50.46(a), said in March that it would take "several years" and substantial agency resources to address the issues raised by the ACRS (INRC, 19 March, 4).

The August 2 briefing provided a rare glimpse behind the curtain at commission deliberations on the proposed rule, and hinted at the central issue involved. McGaffigan said at the briefing that two commissioners had met with staff on 50.46(a) that morning, and that those staffers feel "very strongly" that a rulemaking on fuel cladding integrity, known as 50.46(b) (NuclearFuel, 26 March, 4), must be completed "before anyone's going to want to implement" 50.46(a).

"The position I was hearing this morning" from Office of Nuclear Reactor Regulation staffers, McGaffigan said, "is that it would be very useful to get 50.46(b) done approximately concurrently with 50.46(a)." This position was the primary assumption underlying Dudley's projection in March that revising 50.46(a) would take several years. NRR's preliminary 50.46(b) rulemaking schedule released last month projected that staff would provide a final rule to the commission in October 2009.

Lyons said he was "not sure we should be discussing this morning's meeting at all, so I think I will refrain from saying that I don't quite agree" with McGaffigan on 50.46(a). The earlier staff meeting was not public, but the commission briefing was.

"I don't mind talking about how we're trying to resolve [these issues] in public. These are hard issues. I almost would have had this morning's meeting down here," in the commission conference room, as a public meeting, McGaffigan replied.

At the meeting earlier that day with staff, Lyons said, "we also discussed ways around" the need to complete both rules simultaneously "that had some degree of support from those same people" in NRR. Gary Holahan, deputy director of NRC's Office of New Reactors, said at the commission briefing that treatment of fuel oxidation issues in 50.46(b) is "an important detail that we need to work out, but I don't think it has such a large effect" on core cooling requirements "that it needs to be settled as a precondition to moving forward on 50.46(a)."

"All the folks who love PRAs [probable risk analyses] and think they're perfect think we should plow ahead as we did" with 10 CFR 50.69, NRC's rule on risk-informed categorization and treatment of structures, systems and components, McGaffigan said at the briefing. However, three years after approval of 50.69, "we don't have any applications" and "won't have one until next year," he said, questioning the need to expend resources on 50.46(a) just to "make some sort of grand statement."

McGaffigan said, "My bet would be that if we make that grand statement on too fast a horizon, it'll be 2015 or 2025 before somebody comes in with an application. We did that once. We made that mistake once. We followed theoretical thinking and we got precisely nowhere in the last three years."

SDP changes debated
The industry representatives at the August 2 briefing said there was a need to reform the significance determination process, or SDP, of the agency's reactor oversight process. Industry has previously proposed using licensee PRAs to determine the risk significance of inspection findings, rather than relying solely on the agency's standardized plant analysis risk (SPAR) models. But this approach has met with resistance from agency staff (INRC, 25 June, 3).

Pietrangelo said in his presentation at the briefing that, rather than being "risk-informed," the SDP is a "risk-based process" which "consumes a lot of PRA resources on these de minimis risk evaluations at a time when resources are precious." The "impact on operating companies" of such evaluations is "disproportional to risk significance," and "process improvements are warranted," Pietrangelo said. "It's going to be a trouble-strewn path to find a way to improve that process and use common models" for SDP findings, Richard Rosenblum, senior vice president and chief nuclear officer of Southern California Edison, said at the briefing. But "there is a very, very high reward at the outcome if we can find a way to do that.

"The SDP process is fundamentally the sieve we use on both sides to separate the wheat from the chaff, to know what's a big deal and where we really have to focus senior management and other attention" regarding inspection findings, Rosenblum said. If the current SDP is "giving us the wrong answers, we're putting our attention in the wrong issues. That's a very big deal for the safety of our plants," he said.

Lyons said he was "very torn" on the issue, noting that it was "very, very logical" to have "greater use" of licensee PRAs in the regulatory process. However, Lyons said that he sees "a pretty substantial host of challenges if the NRC agreed tomorrow to use plant-specific PRAs" in the SDP, including assuring the continued availability of models that NRC staff can run on the agency's computer systems with "unfettered access to the codes."

Lyons said that "there's enough variability in the way the plant-specific PRAs are constructed that it would be quite difficult for our relatively small staff to embrace and understand all the variations in the plant-specific PRAs."

Pietrangelo said that industry is "open to suggestions" on how to approach the issue. "Quite frankly, I don't think this proposal is going any place fast. It was clear to me that the staff was not in favor of this approach," he said.

Jaczko said that NRC staff and industry are "arguing and taking so long with the SDPs" not because of issues with PRA models, but because the industry doesn't want to receive white findings under the ROP.

"I think the solution ... is to improve plant performance so that we're not in the margins on that white boundary or the white to yellow boundary," Jaczko said. And if a licensee's assessment of the risk associated with an inspection finding doesn't agree with that of NRC staff, "then we take the higher number or average them or whatever. I think that there's a way to accomplish that and to move the SDP process forward," he said.
—Steven Dolley and Jenny Weil, Washington

Pilgrim Nuclear Power Station
License Renewal Safety Evaluation Report

Staff Presentation to the ACRS

Perry Buckberg

Project Manager

Office of Nuclear Reactor Regulation

September 6, 2007

Introduction

- Overview
- Section 2: Scoping and Screening Review
- License Renewal Inspections
- Section 3: Aging Management Review Results
- Section 4: Time-Limited Aging Analyses (TLAAs)

Overview

- LRA Submitted by Letter, January 27, 2006
- GE BWR3 - MARK 1 Containment
- 2028 MWth, 690 MWe
- Op License DPR-35 Expires June 8, 2012
- Located in Plymouth, MA

Overview

- SER Issued June 28, 2007
- SSER to be Issued September, 2007
- Open Items (4) Have Been Closed
- Four (4) License Conditions
- 92 RAIs Issued, 329 Audit Questions
- ≈82% Consistent With GALL Report, Revision 1

Review Highlights

- AMP GALL Audit
  - May 22, 2006
- Scoping and Screening Methodology Audit
  - June 6 - June 9, 2006
- AMR GALL Audit
  - June 19, 2006
- AMP/AMR Status Briefing
  - July 17 - 19, 2006
- Regional Inspections
  - September 18 - 22, 2006
  - October 2 - 6, 2006
  - December 6 - 7, 2006

Section 2: Scoping and Screening Review

Section 2.1 - Scoping and Screening Methodology

- On-site Audit - June 6 – June 9, 2006
- Pilgrim included all system components in scope if any components were (a)2 – exceptions stated

Section 2.3

- 4 Additional Components Brought Into Scope

Section 2.2, 2.4, 2.5

- No Omissions

Section 2: Scoping and Screening Review

Section 2.3 – Mechanical Systems

- Open Item 2.3.3.6: Security Diesel
  - LRA Did not Include System Drawings
  - Referred to Regional Inspector to Determine System Components in Scope
  - Staff Considered the 3/9/2007 Inspector Input Adequate to Close the Open Item

Section 2: Scoping and Screening Summary

- The Applicant's Scoping Methodology Meets The Requirements Of 10 CFR Part 54.4

- Scoping And Screening Results, As Amended, Included All SSCs Within The Scope Of License Renewal And Subject To AMR

License Renewal Inspections

Glenn Meyer

Region I

Scoping and Screening

- 54.4(a)(2) - Non-safety SSCs Whose Failure Could Impact Safety SSCs
- Spatial and Structural Interactions
- LRA Drawings and Procedures Reviewed
- Plant Walkdowns Performed

Scoping and Screening Conclusions

- Spatial Interaction - Acceptable
- Structural Interaction – Corrected
- Scoping and Screening Acceptable for License Renewal

Aging Management

- Reviewed 26 AMP Programs
- Reviewed Programs, Evaluations, and Records
  - Program Procedures
  - Operational Experience Information
  - Prior Pilgrim Issues
- Performed Plant Walk Downs
- Interviewed Cognizant Personnel

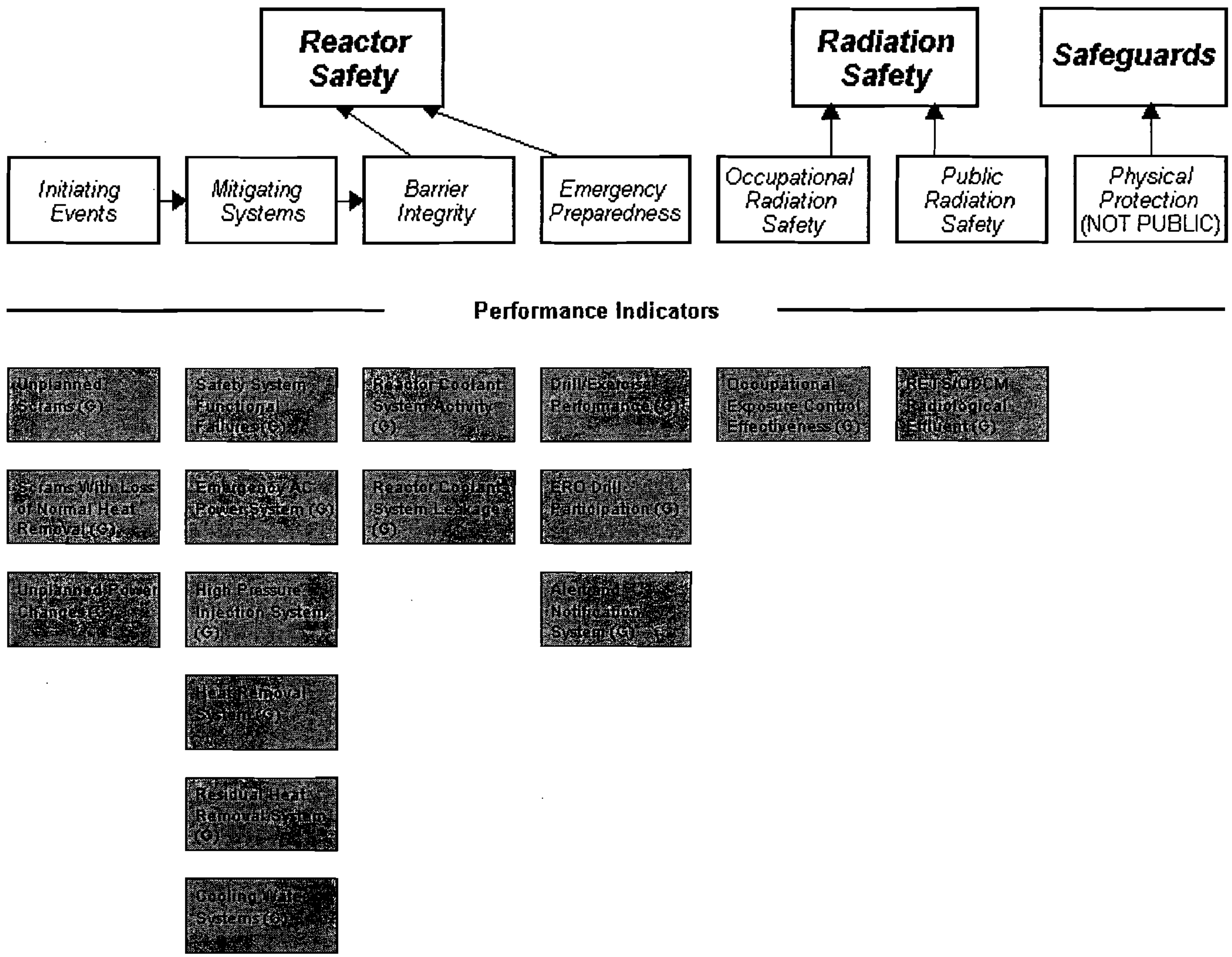
Inspection Conclusions

- Scoping and Aging Management Programs Support Conclusion That Aging Effects will be Managed
- Drywell Shell Monitoring

Current Performance

- Licensee Response Column (Column I) of the NRC's Action Matrix – Green PIs and Findings
- No Cross-cutting Issues
- Reactor Oversight Process Baseline Inspections

Performance Indicators



Inspection Findings

Reactor Safety: Initiating Events → Mitigating Systems → Barrier Integrity; Emergency Preparedness

Radiation Safety: Occupational Radiation Safety; Public Radiation Safety

Safeguards: Physical Protection/ Security

Most Significant Inspection Findings

	Initiating Events	Mitigating Systems	Barrier Integrity	Emergency Preparedness	Occupational Radiation Safety	Public Radiation Safety	Physical Protection/ Security
2Q/2007	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/2007	No findings this quarter	G	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	
4Q/2006	No findings this quarter	G	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	
3Q/2006	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	

Miscellaneous findings

Pilgrim Nuclear Power Station
Aging Management Review
Time Limited Aging Analysis
Open Items

Fire Protection Program (B.1.13.1)

- Open Item 3.0.3.2.10:
  - Applicant did not Adequately Address how to Manage the Aging Effects of Inaccessible Seals.
  - Applicant Stated (ACRS) and Documented (June 2007) That There are Actually No Inaccessible Seals at PNPS

Containment Inservice Inspection Program (B.1.16.1)

- <u>Open Item 3.0.3.3.2</u>:
  - Regional Inspection Documented:
    - Inoperative Bellows Rupture Drain Flow Switch
    - Drain Monitoring Inconclusive & Undocumented
    - Water on Torus Room Floor

Containment Inservice Inspection Program (B.1.16.1)

- Open Item 3.0.3.3.2:
  - Replace Switches Now and in 15 years
  - Identified Non-Aggressive Groundwater as Source of Water on Torus Room Floor
    - Tested November 2006 and June 2006
  - Provided Documentation of Drain Monitoring
    - Committed to Obtain Drywell UT Data

Containment Inservice Inspection Program (B.1.16.1)

- Open Item 3.0.3.3.2:
  - Torus Structure
    - Provided Evaluation of Effect on Torus Basemat
    - Commitments to Evaluate Groundwater/Torus Water
    - Commitment to Inspect Condition of Torus Hold Down Bolts and Grout

Section 4.2: Reactor Vessel Neutron Embrittlement

- Six TLAAs Affected by Neutron Fluence
  - Reactor Vessel Fluence
  - Pressure-Temperature Limits
  - Upper Shelf Energy
  - Adjusted Reference Temperature
  - Circumferential Weld Inspection Relief
  - Axial Weld Failure Probability

Section 4.2: Reactor Vessel Neutron Embrittlement

- <u>Open Item 4.2</u>
  - Pilgrim – The First BWR-3 to Use RAMA Methodology to Calculate Neutron Fluence
  - Dosimetry Data was not Available with Which to Benchmark the RAMA Calculated Results
  - Result - Fluence Calculation Not Acceptable Per Reg Guide 1.190

Section 4.2: Reactor Vessel Neutron Embrittlement

- Open Item 4.2
  - Applicant's Back Calculation of Limiting Fluence Values Considered Acceptable by the Staff
  - TLAA Identified Which Established the Limiting Fluence Value
    - Axial Welds @ RV Inner Surface - 3.37 x 10^{18} n/cm2 (E > 1.0 MeV)

Section 4.2: Reactor Vessel Neutron Embrittlement

- Open Item 4.2

  – License Condition 4.2.6: On or before June 8, 2010, the applicant (Entergy) will submit to the NRC correctly benchmarked RV neutron fluence calculations, consistent with RG 1.190, that will confirm that the neutron fluence for the lower intermediate shell axial welds, at the inner surface of the RV, will not reach the limiting value of 3.37×10^{18} n/cm2 (E > 1.0 MeV) by the end of the period of extended operation (54 EFPY).

Section 4.2: Reactor Vessel Neutron Embrittlement

- Open Item 4.2
  - Commitment 47: Submit to the NRC An Action Plan for Benchmarking the Reactor Pressure Vessel Fluence Evaluation.
  - Entergy Plan Submitted August 23, 2007.

Section 4.3: Metal Fatigue

- Reactor Water Environment
  - Removed Exception to Fatigue Monitoring Program regarding Environmentally Assisted Fatigue.
  - Combined FMP and EAF – FMP is Now Consistent with GALL.

Conclusions

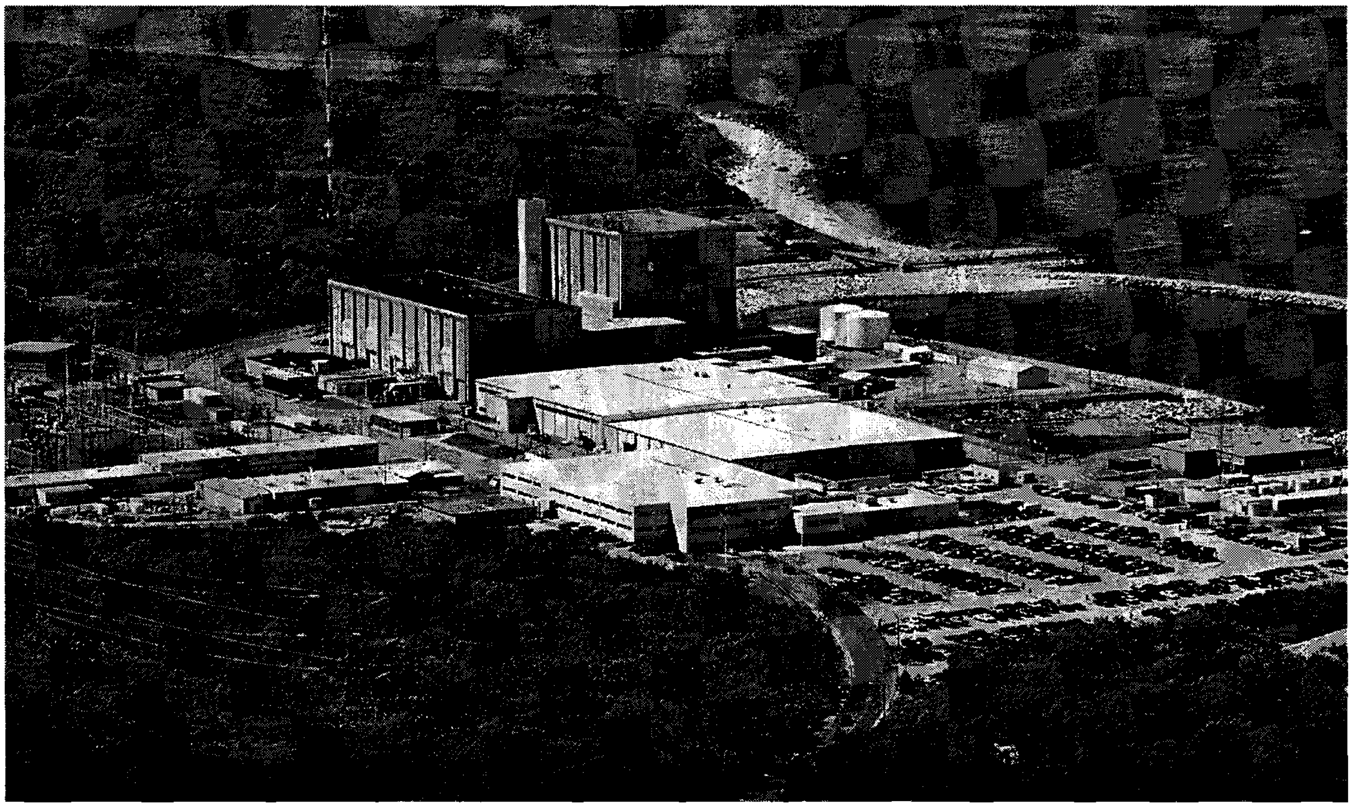
- On the basis of its review of the LRA, with the closing of Open Items 2.3.3.6, 3.0.3.2.10, 3.0.3.3.2 and 4.2, the staff determines that the requirements of 10 CFR 54.29(a) have been met.

Questions

3

Pilgrim Nuclear Power Station

License Renewal ACRS
September 6, 2007



Entergy

Pilgrim Personnel in Attendance

Kevin Bronson	Site Vice President
Steve Bethay	Director of Nuclear Safety Assurance
Brian Sullivan	Director of Engineering
Bryan Ford	Senior Manager NS&L
Alan Cox	Entergy LR Project Manager
Fred Mogolesko	Pilgrim LR Project Manager

Other support personnel

Agenda

- Description and Current Status
- Licensing History and Highlights
- License Renewal Project
- Draft SER (March 2007)
  - 4 Open Items
- Final SER (June 2007)
  - Open Items resolved
- Summary

Pilgrim Description

- Located in Plymouth, Massachusetts on Cape Cod Bay
- ~ 40 miles south of Boston
- Sited on 1600 Acres
- BWR-3
- Mark I Containment
- General Electric (NSSS), Bechtel (AE and Constructor)
- 2028 MWt Thermal Power; ~ 690 MWe
- Open Cycle Condenser Cooling
- Owned and Operated by Entergy
- Staff: ~ 650

Current Plant Status

- Completed RFO-16 May 9, 2007
- Operating at 100% power
- NRC PIs & Inspection Findings All Column 1
- Next Refueling Outage April/May 2009

Licensing History and Highlights

• Construction Permit	August 26, 1968
• Operating License	June 8, 1972
• Full Power License	September 15, 1972
• Commercial Operation	December 9, 1972
• License Transfer to Entergy	July 13, 1999
• Appendix K Power Uprate (1.5%)	May 8, 2003
• LR Application Submitted	January 25, 2006
• Operating License Expires	June 8, 2012

Licensing History and Highlights
(continued)

Significant design improvements

- 1977- Replaced Core Spray safe-ends and piping inside primary containment with IGSCC-resistant material
- 1978 -1982 Mark I containment modifications
- 1984 - Replaced recirculation piping to address IGSCC concerns
- 1986 -1989 Safety enhancement modifications (SSW-RHR cross-tie, Direct Torus Vent to Main Stack, Station Blackout Diesel Generator)

Licensing History and Highlights
(continued)

Significant design improvements

- 1991 - Hydrogen water chemistry
- 1995 - Replaced ECCS suction strainers
- 2007 - Implementation of Noble Metals

- Spent fuel pool capacity adequate through end of current operating license
- Dry cask storage project to be initiated in 2008

Entergy

License Renewal Project

- LRA prepared by experienced, multi-discipline Entergy team (corporate and on-site)
- Extensive training program provided to Engineering, Licensing, and QA
- Pilgrim and VY LRAs first applications submitted following issuance of Rev. 1, SRP and GALL
- Incorporated lessons learned from previous applications
- Peer review conducted (10 Utilities), all observations addressed
- LRA internal reviews (OSRC, SRC, QA)

License Renewal Project
(continued)

- Commitments in the LRA refined as needed during audit/inspection process (40 aging management programs)
- Commitments captured in the Pilgrim commitment tracking system
- Programs owned by site Engineering
  - 14 programs in place w/o enhancements
  - 16 programs require enhancement
  - 10 new programs

Safety Evaluation Report (SER)

- **Draft SER - 4 Open Items** (March 2007)
  - OI 2.3.3.6 Security Diesel Generator
  - OI 3.0.3.2.10 Fire Barrier Penetration Seals
  - OI 3.0.3.3.2 Containment Inservice Inspection
  - OI 4.2 Reactor Vessel Neutron Fluence
- **Final SER** (June 2007)
  - All open items resolved

Security Diesel Generator

OI 2.3.3.6

- Region 1 Confirmatory Item to determine if security diesel components are within the scope of license renewal
- Requested support provided

Fire Barrier Penetration Seals

OI 3.0.3.2.10

- Concern on aging management of inaccessible seals
- All penetration seals are included in the inspection program

Entergy

Containment Inservice Inspection
OI 3.0.3.3.2

- Potential for corrosion in the inaccessible area of the steel containment shell, base mat and sand pocket region

Entergy

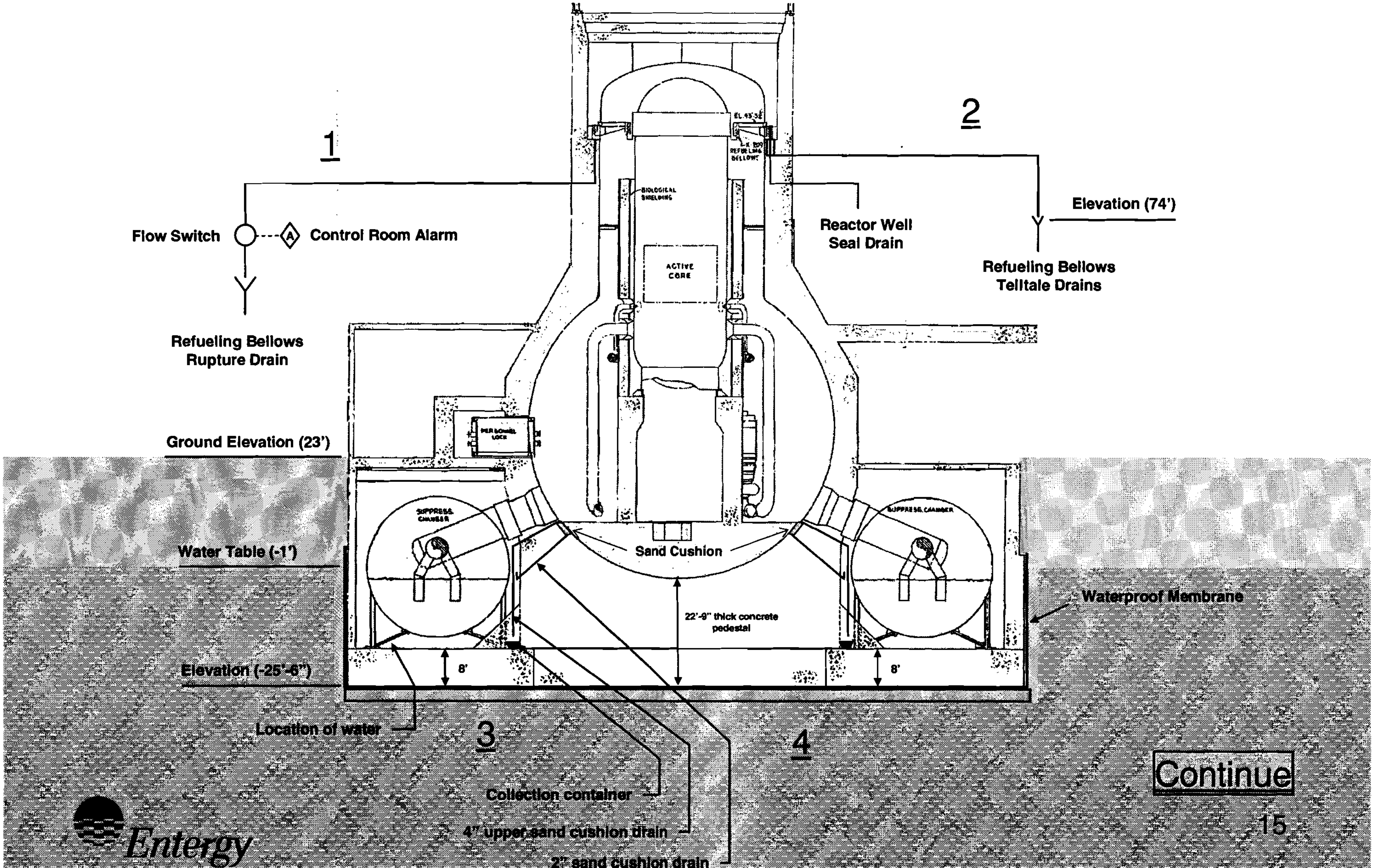
Containment Inservice Inspection

OI 3.0.3.3.2

Drywell Shell Condition and Monitoring

- Defense in depth design minimizes potential for undetected water intrusion
- Diverse methods of prevention and identification of potential water leakage into air gap
- No refueling bellows leakage
- No water intrusion into drywell air gap
- No drywell shell degradation
- Confirmatory inspections planned and performed

Containment Inservice Inspection
Drywell Shell Condition and Monitoring



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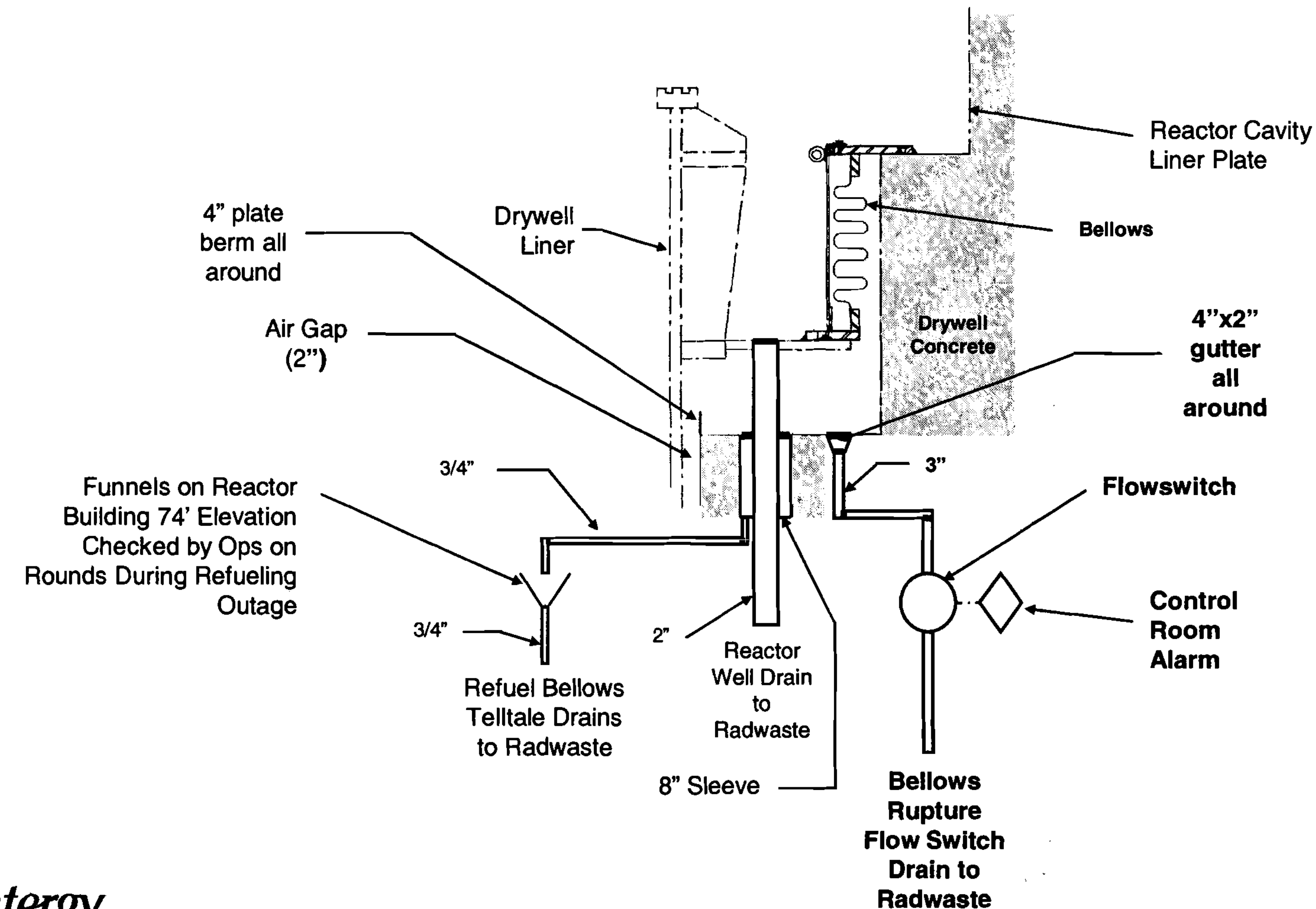
Containment Inservice Inspection
Drywell Shell Condition and Monitoring

3" instrumented drain line alarms in control room



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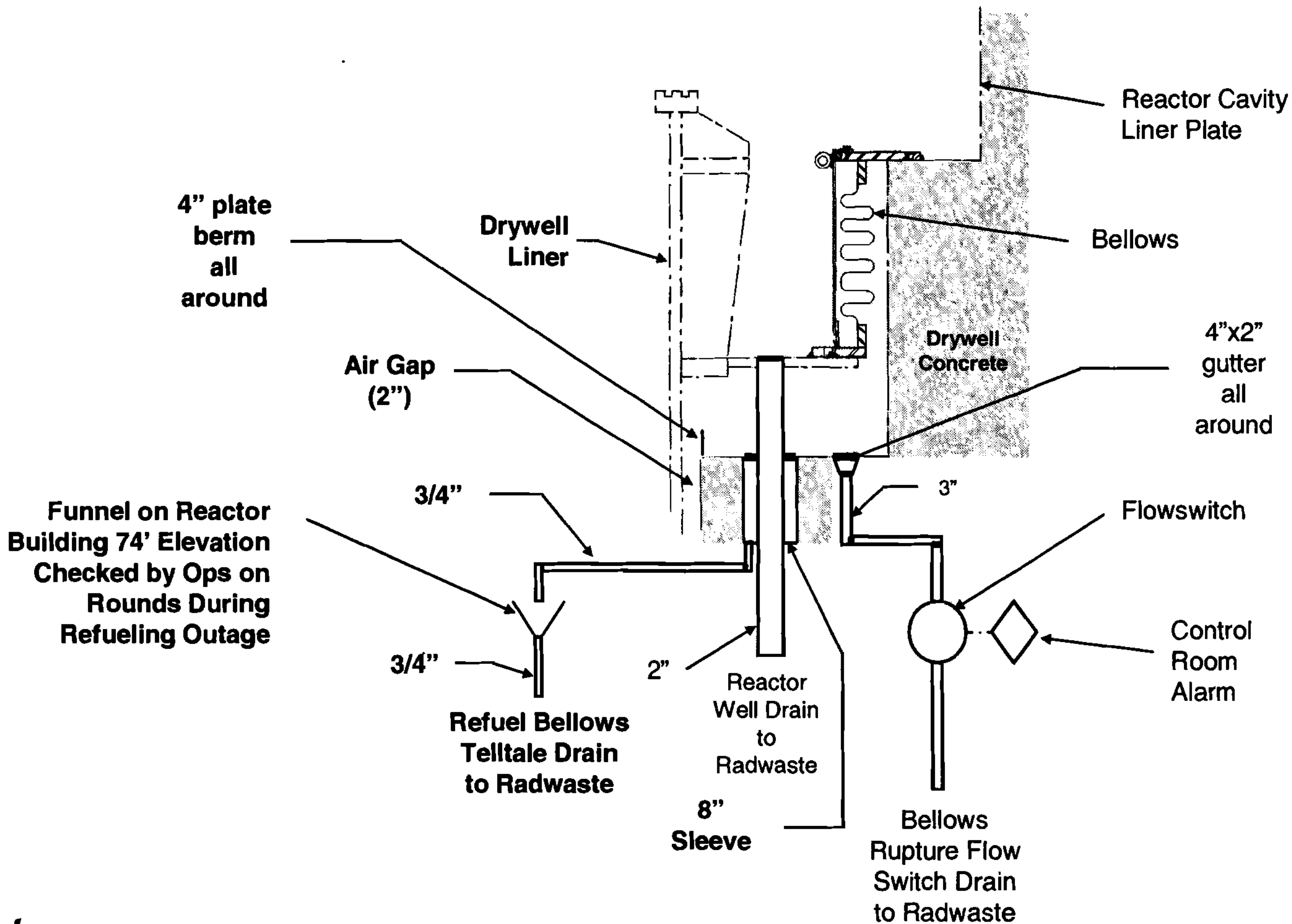
Containment Inservice Inspection

Drywell Shell Condition and Monitoring

Four ¾" drain lines which exit to 74' checked during operator tours



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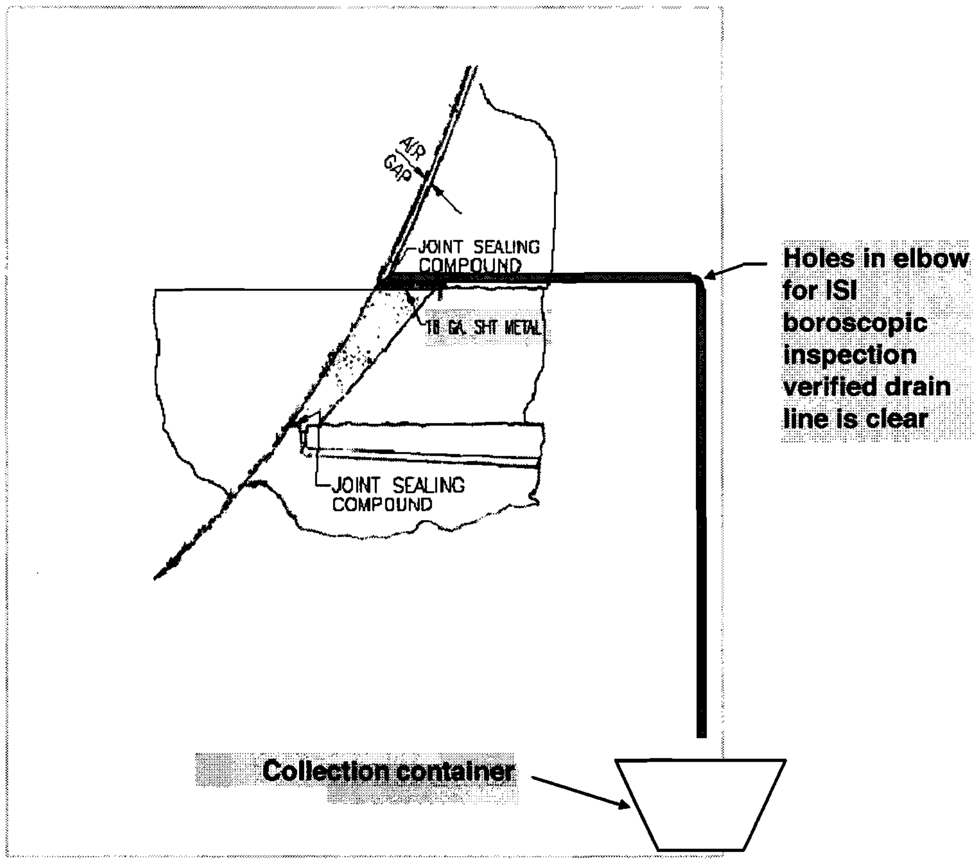
Containment Inservice Inspection
Drywell Shell Condition and Monitoring

Four 4" upper sand cushion drains drain into collection devices and are monitored at beginning and end of each RFO



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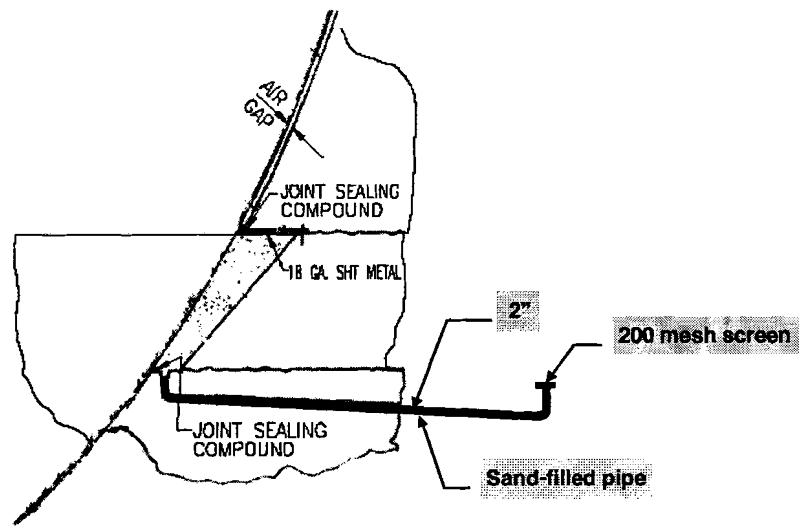
Containment Inservice Inspection
Drywell Shell Condition and Monitoring

Four sand cushion drains provide further detection capabilities



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Containment Inservice Inspection

Drywell Shell Condition and Monitoring

Past Inspections

- Limited confirmatory examinations
  - UT at twelve locations at 9'-2" elevation
  - UT at four locations at 9'-1" elevation
    - Concrete chipped out to a depth of 1"
  - UT at six locations at 72' and 83' elevations
- Verified upper sand cushion drains unobstructed and dry
- All inspections identified no corrosion

Containment Inservice Inspection

Drywell Shell Condition and Monitoring

Future Examinations

- UT at 12 locations at 9'-2" elevation
  - Prior to Period of Extended Operation
  - Once within first 10 years
- UT at 4 locations at 9'-1" elevation
  - Prior to Period of Extended Operation
  - Once within first 10 years
- UT at 72' elevation adjacent to SFP
  - Conducted every 40 months by IWE

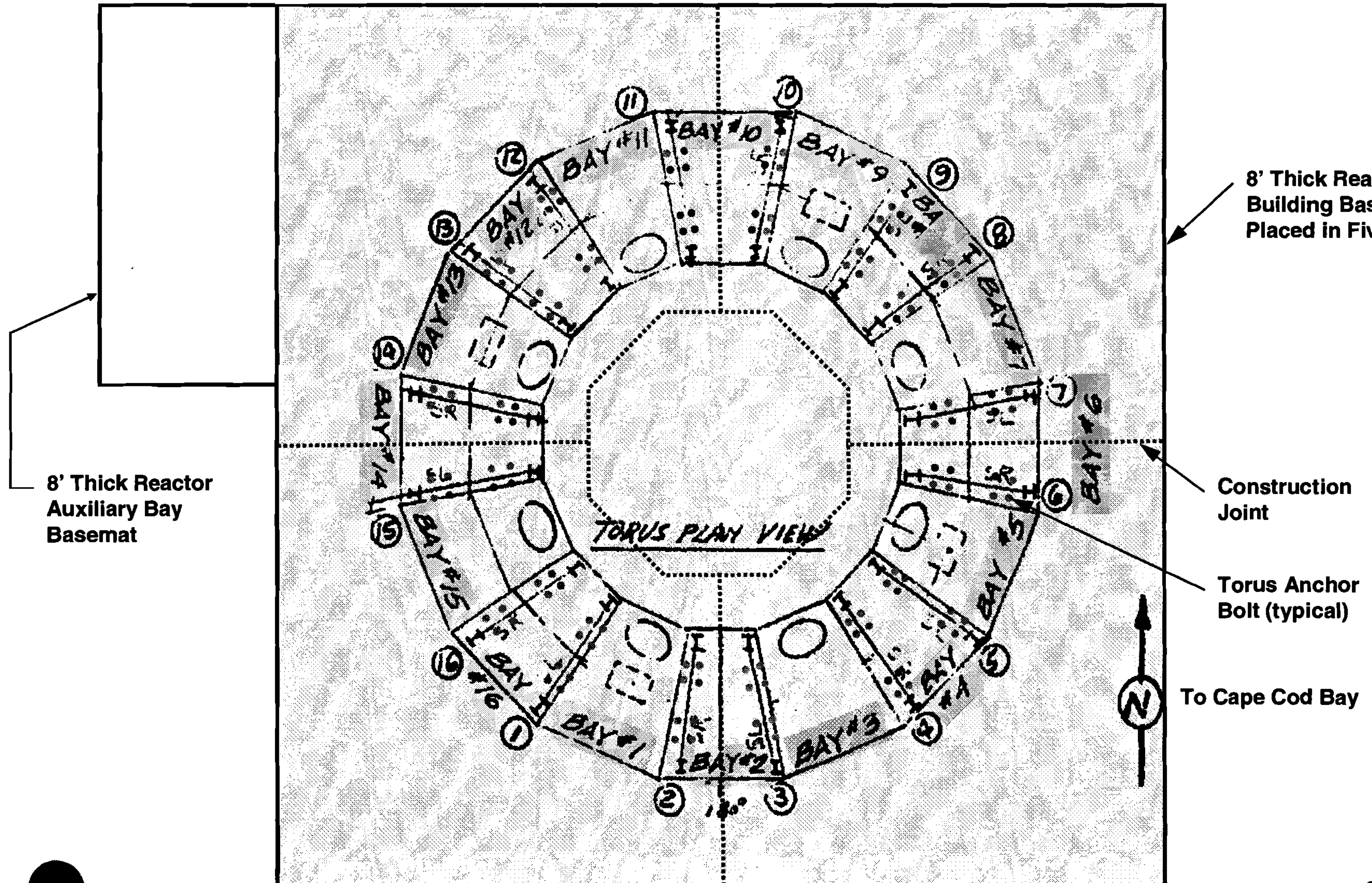
Containment Inservice Inspection

OI 3.0.3.3.2

Water on Torus Room Floor

Containment Inservice Inspection

Torus Room Floor



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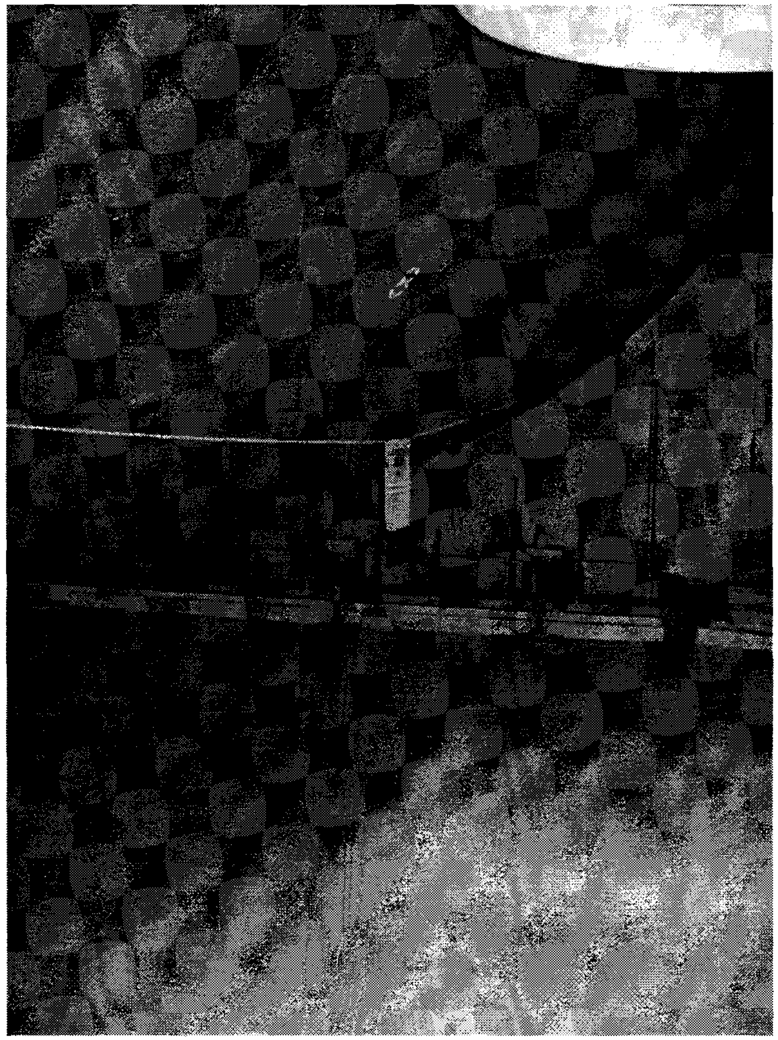
Containment Inservice Inspection

Torus Room Floor

Bay 8



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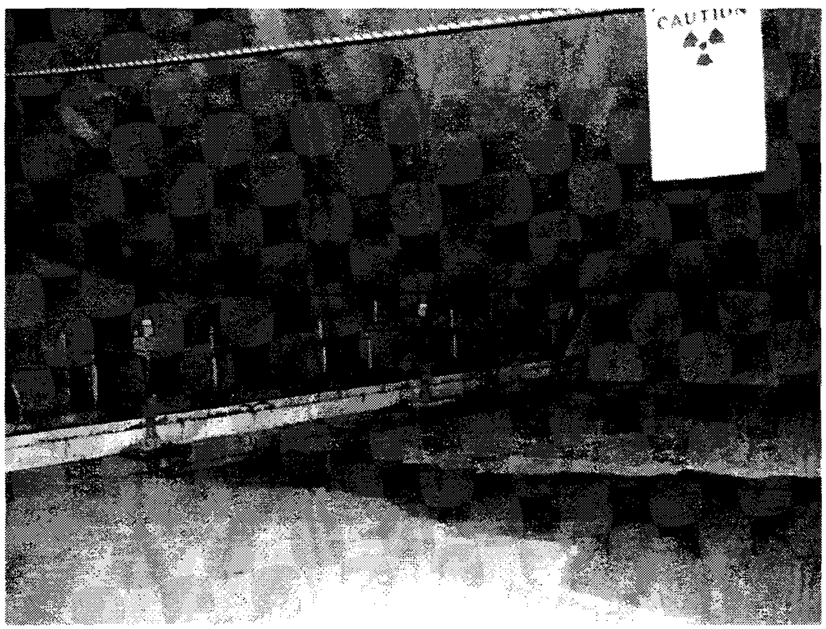
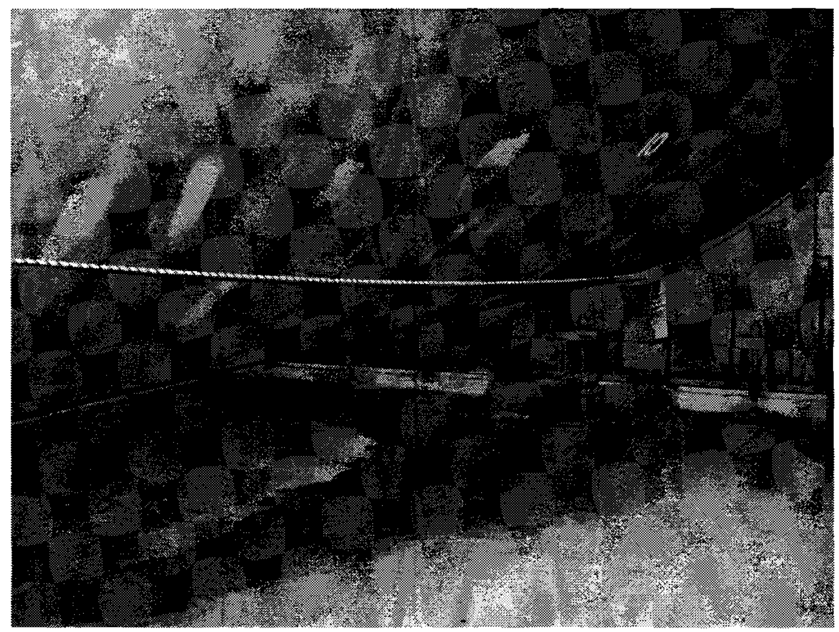
Containment Inservice Inspection
Torus Room Floor

Bay 10





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Containment Inservice Inspection
Water on Torus Room Floor

Aspects Evaluated

- Source of water
- Integrity of anchor bolts
- Structural adequacy of the reactor building
- Inspection and monitoring of water, concrete, and Torus hold down anchor bolts

Independent Assessment by Dr. Franz Ulm - MIT

Containment Inservice Inspection

Water on Torus Room Floor

Source of water

- The source is ground water seepage under hydraulic pressure
- Path is through vertical joints and zones most likely weakened by tensions generated during setting and hydration following the construction (normal occurrence)
- Low seepage rate is counteracted by evaporation
- Non-aggressive, benign water chemistry

Containment Inservice Inspection

Water on Torus Room Floor

Integrity of anchor bolts

- Implemented commitment to inspect grout and bolts for degradation/corrosion

  Two cases evaluated:

    Bay 8: Typically dry (1 bolt inspected)

    Bay 10: Typically wet (4 bolts inspected)

- Inspection included lifting of jacking plate
- Results:

  No degradation of bolt or grout

Containment Inservice Inspection

Water on Torus Room Floor

Structural adequacy of the reactor building

- Past sampling of water on floor demonstrated non-aggressive water chemistry
- No structural distress evident
- Groundwater is non-aggressive to base-mat
- Concrete Water Chemistry
  - Minimum degradation threshold limits for concrete established:
    - Acidic solutions with pH < 5.5
    - Chloride solutions > 500 ppm
    - Sulfate solutions > 1500 ppm
- Water re-analyzed to demonstrate non-aggressiveness

Containment Inservice Inspection

Water on Torus Room Floor

Future Commitments

- Determine additional actions based on inspection of bolts and water analysis, prior to the period of extended operation
- Monitor chemistry of groundwater, every five years
- Monitor chemistry of water on floor
  - Prior to the period of extended operation, and
  - Once every five years during the period of extended operation
- Inspect Structure in accordance with Structures Monitoring Program, every five years

Containment Inservice Inspection

Water on Torus Room Floor

Independent Assessment

- Evaluate functional capability of torus base-mat.
  - Professor Franz Ulm of MIT's Department of Civil Engineering
- Groundwater migration is highly localized
- Does not compromise the overall structural performance of the torus base mat.
- Does not affect the bulk integrity of the concrete slab or the overall compressive and bending load bearing capacity of the reactor foundation.
- Non-aggressiveness of ground water verified
- The localized calcium leaching does not affect the overall structural performance of the slab.

Reactor Vessel Neutron Fluence

OI 4.2

- Lack of benchmarking data to support plant specific fluence calculations for use in TLAAs

Reactor Vessel Neutron Fluence

- Current P-T curves valid through 2011 RFO.
- Commitment to submit RG 1.190 calculations by June 2010
- Current Status:
  - Evaluated TLAAs to determine limiting fluence (RG 1.99)
    - Adjusted Reference Temperature
    - Upper Shelf Energy
    - RPV internals (top guide and shroud tie-down)
    - RPV welds
    - RPV nozzles near beltline
  - Axial Weld Failure Probability is limiting at 5×10^{-6} per Reactor Year
  - Limiting fluence value will not be challenged at 54 EFPY

Entergy

Reactor Vessel Neutron Fluence

License Condition:

On or before June 8, 2010, the applicant will submit to the NRC correctly benchmarked RV neutron fluence calculations, consistent with RG 1.190, that will confirm that the neutron fluence for the lower intermediate shell axial welds, at the inner surface of the RV, will not reach the limiting value of 3.37×10^{18} n/cm^2 ($E > 1.0$ MeV) by the end of the period of extended operation (54EFPY)

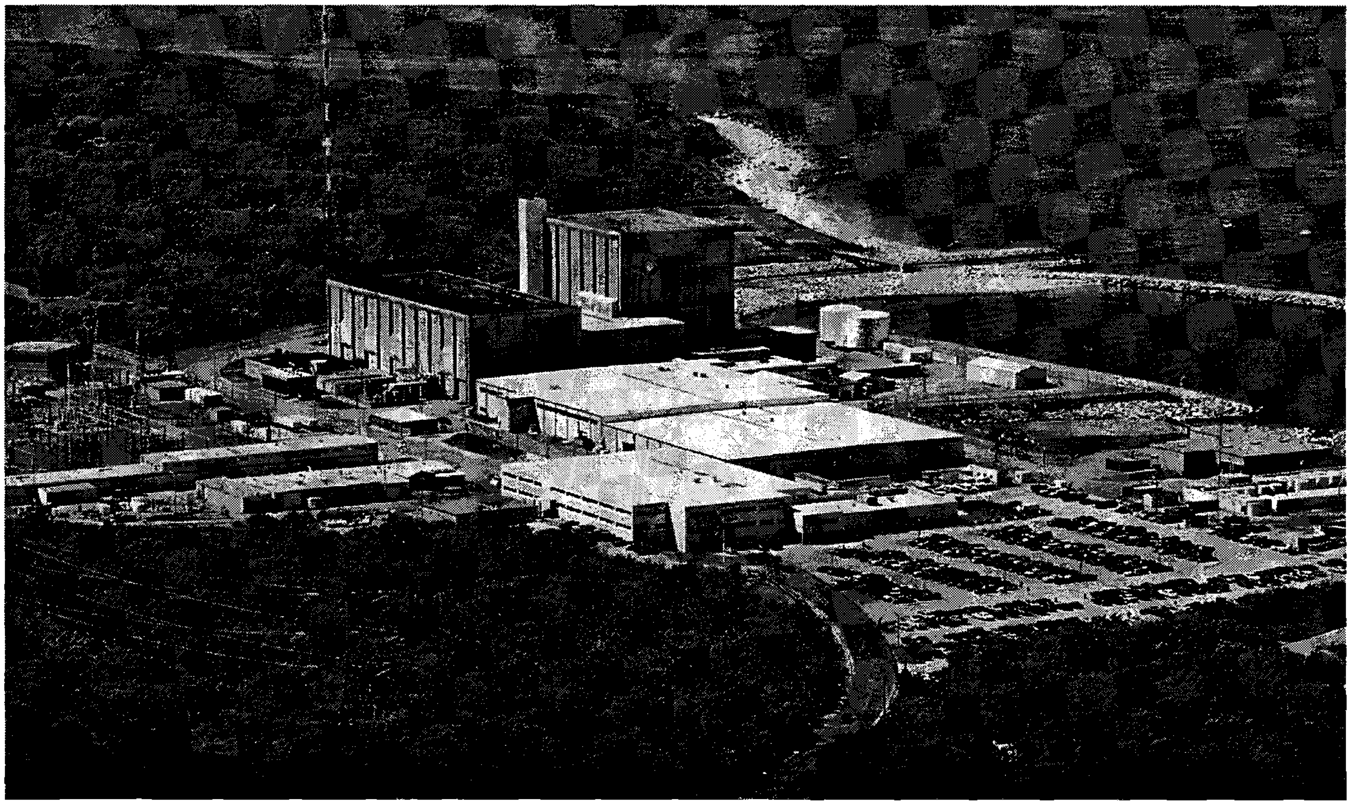
Summary

Pilgrim Station Team

- Understands plant aging issues
- Recognizes the relationship between successful implementation of LR commitments and enhanced reliability of plant SSCs
- Tracking the LR commitments and initiated implementation
- Has integrated the implementation of LR commitments into the organizational culture as an ongoing responsibility through the period of extended operation

Pilgrim Nuclear Power Station

Questions?



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Standard Review Plan (SRP) Sections 19.0 and 19.2

Division of Safety Systems and Risk Assessment
Office of New Reactors

September 2007

Outline

- Background
- Applicable regulations
- Timeline
- RG and SRP renumbering
- Uses of the PRA
- PRA scope, level of detail, and technical adequacy
- PRA documentation
- Revision of SRP Section 19.2
- Clarifications

Background

- September 2006 – DG-1145 issued for comment
- October 2006 – Office of New Reactors established
- October 31, 2006 – Staff issued SECY-06-0220 (deleted the requirement to submit the PRA)
- December 12, 2006 – ACRS letter on DG-1145
- February 2007 – Two PRA branches established in NRO
- April 11, 2007 – Commission issued an SRM on SECY-06-0220 (agreed with the staff)
- June 22, 2007 – RG 1.206, SRP Section 19.0, and SRP Section 19.2 issued
- August 28, 2007 – Revised Part 52 issued (along with conforming changes in other regulations)

Applicable Regulations (1 of 3)

- Design Certifications:
  - 10 CFR 52.47(a)(27) - The FSAR must contain "...a description of the design-specific probabilistic risk assessment (PRA) and its results."
- Combined Licenses:
  - 10 CFR 52.79(a)(46) - The FSAR must contain "...a description of the plant-specific probabilistic risk assessment (PRA) and its results."

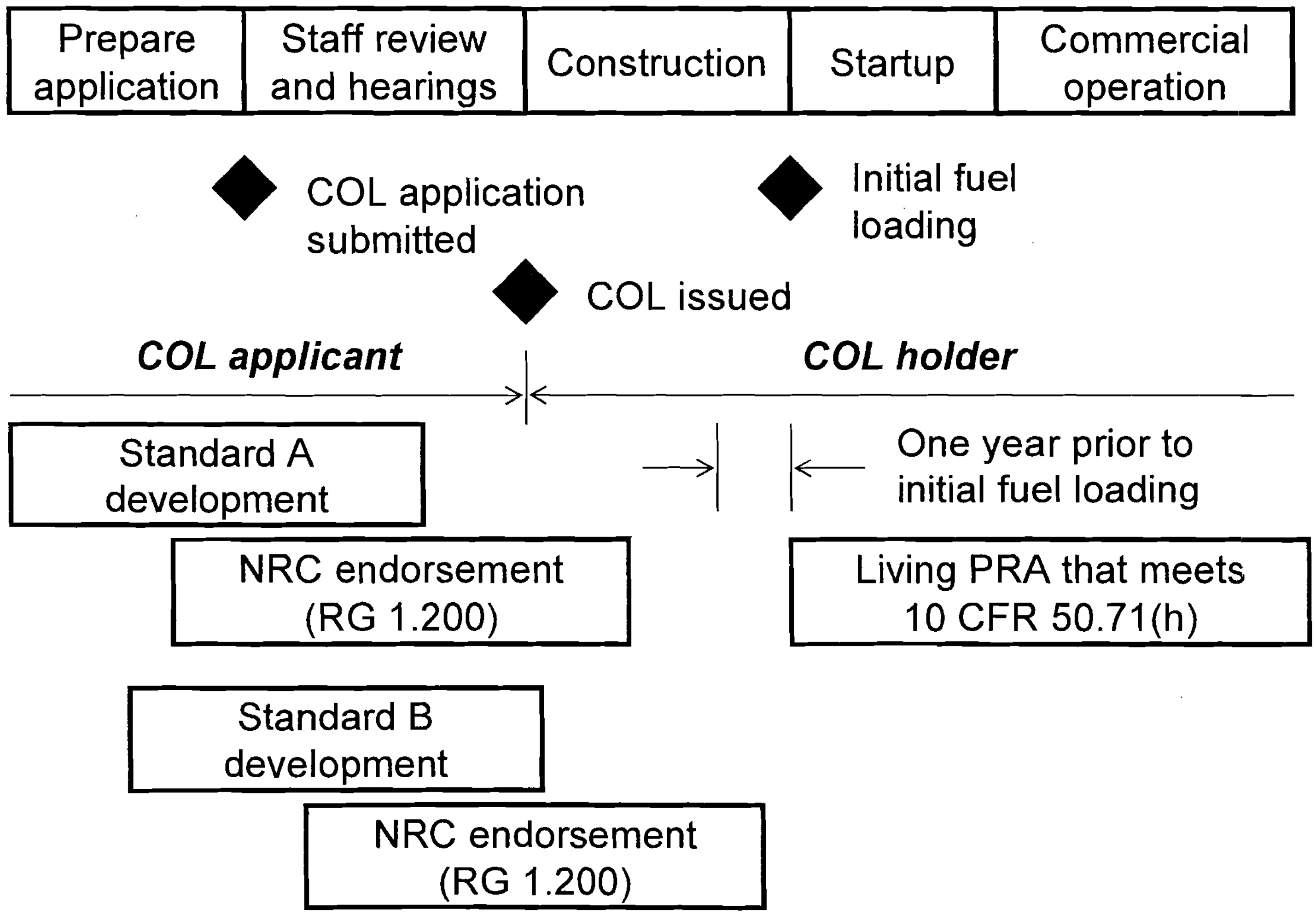
Applicable Regulations (2 of 3)

- If the COL application references a standard design approval, then:
  - 10 CFR 52.79(c)(1) - The plant-specific PRA information must use the PRA information for the design approval and must be updated to account for site-specific design information and any design changes or departures.
- If the COL application references a standard design certification, then:
  - 10 CFR 52.79(d)(1) - The plant-specific PRA information must use the PRA information for the design certification and must be updated to account for site-specific design information and any design changes or departures.
- If the COL application references the use of one or more manufactured nuclear power reactors licensed under subpart F of 10 CFR Part 52, then:
  - 10 CFR 52.79(e)(1) - The plant-specific PRA information must use the PRA information for the manufactured reactor and must be updated to account for site-specific design information and any design changes or departures.

Applicable Regulations (3 of 3)

- For COL holders: PRA maintenance and upgrading
  - 10 CFR 50.71(h)(1) - No later than the scheduled date for initial loading of fuel, each holder of a combined license under subpart C of 10 CFR part 52 shall develop a level 1 and a level 2 probabilistic risk assessment (PRA). The PRA must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist one year prior to the scheduled date for initial loading of fuel.
  - 10 CFR 50.71(h)(2) - Each holder of a combined license shall maintain and upgrade the PRA required by paragraph (h)(1) of this section. The upgraded PRA must cover initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to each required upgrade. The PRA must be upgraded every four years until the permanent cessation of operations under § 52.110(a).
  - 10 CFR 50.71(h)(3) - Each holder of a combined license shall, no later than the date on which the licensee submits an application for a renewed license, upgrade the PRA required by (h)(1) to cover all modes and all initiating events.

Timeline


Prepare application
Staff review and hearings
Construction
Startup
Commercial operation
COL application submitted
Initial fuel loading
COL issued
COL applicant
COL holder
Standard A development
One year prior to initial fuel loading
NRC endorsement (RG 1.200)
Living PRA that meets 10 CFR 50.71(h)
Standard B development
NRC endorsement (RG 1.200)

RG and SRP Renumbering

Old SRP	New SRP	RG	Topic
n/a	Section 19.0	RG 1.206 - C.I.19 - C.III.1	Combined License Applications for Nuclear Power Plants (LWR Edition) COL applications that are not based on a DC COL applications that are based on a DC (see Chapter 19 for PRAs)
Chapter 19.1	Section 19.1	RG 1.200	PRA technical adequacy
Chapter 19	Section 19.2	RG 1.174	Risk information used to support permanent plant-specific changes to the licensing basis

PRA Scope for a COL Application

- Level 1 (core-damage) and Level 2 (containment analysis)
- All initiating events
  - Internal initiators (e.g., transients, LOCAs)
  - External initiators (e.g., seismic, internal fires)
- All operating modes
  - Full-power
  - Low-power and shutdown
- A lack of NRC-endorsed industry consensus standards does not reduce this scope

PRA Level of Detail

- Must reflect the as-to-be-built and as-to-be-operated plant
  - Need to review the DC PRA, and revise as necessary (e.g., site-specific service water system design)
  - Use of bounding analyses is acceptable under certain conditions
    - Identify vulnerabilities, design and operational requirements, ITAACs, COL Action Items
    - Do not mask or distort risk-significant information or risk insights

PRA Technical Adequacy

- RG 1.200 provides one acceptable approach to demonstrating acceptable technical adequacy
- NRC-endorsed industry consensus standards require peer reviews
- The ASME PRA Standard states that users may need to add or revise requirements in the Standard to address advanced LWRs (novel or passive features, digital I&C, etc.)
- Meeting NRC-endorsed industry consensus standards should help expedite the staff's review

PRA Documentation

- Information to be included in the FSAR has been identified in RG 1.206, Section C.I.19, Appendix A
  - COLs based on a DC may include information by reference (see RG 1.206, Section C.III.1, Chapter 19 for guidance)
- Applicants should maintain archival information per RG 1.200
- NRC staff may seek clarifying information through the RAI process or through audits (documented in a publicly available audit report that can be referenced in the staff's SER)

Format and Content

- RG 1.206, Section C.I.19, Appendix A (format and content guidance) provides one acceptable definition of the phrase "description of the PRA and its results."

Description of the PRA

- PRA methodology
- List of initiating events
- Success criteria (what they are, how they were determined including T/H codes used)
- Accident sequences (event tree plots may be helpful)
- List of plant systems and functions, including dependency matrix
- Identify the source of all numerical data used
- PRA software platform
- PRA truncation limit

PRA Results

- Risk metrics (CDF, LRF, CCFP)
- Description of significant sequences and their mean frequencies
- Significant initiating events and their percent contribution to the overall risk metrics
- Significant functions, SSCs, operator actions and their FV importance and RAW values
- PRA assumptions and PRA-based insights
- Results from sensitivity and uncertainty analyses

Revision of SRP Section 19.2

- Updates made in accordance with NRR Office Instruction LIC-200, Rev. 1
- Added references to RG 1.200 and SRP Section 19.1 concerning PRA technical adequacy
- Some rewording as directed by OGC
- Some changes to improve clarity, correct errors, etc.

Clarifications (1 of 4)

- The staff has held three public meetings to discuss PRA information to support DC and COL applications
  - Well-attended by prospective DC and COL applicants
- The meetings help to identify a list of "frequently asked questions"
  - The staff has developed answers to most of the FAQs
  - The staff plans to issue Staff Interim Guidance (ISG) on these clarifications

Clarifications (2 of 4)

- Format is optional, but all content should be provided
- Seismic and fire risk evaluations may use the methods used in the DC PRA; however, once consensus standards are endorsed by the staff, applicants should follow these standards
- 10 CFR 50, Appendix B does not apply to DC or COL PRAs
- Chapter 19 PRA information is not subject to the Tier 2 change process
- Generally, Capability Category 1 is adequate for a DC or COL PRA

Clarifications (3 of 4)

- With respect to 10 CFR 50.71(h) and the use of NRC-endorsed standards that exist one year prior to fuel load, applicants may petition to change the rule or seek an exemption from the rule
- Definition of LRF
  - NRC has not issued a formal definition
  - Applicants may use the definition used to develop the DC PRAs
  - Staff is considering ways to reconcile the use of LRF for Part 52 licensing and the use of LERF for risk-informed LARs per RG 1.174
- PRA maintenance starts at the time of application; PRA upgrade starts at the time of initial fuel load
- COL holders are expected to maintain the entire scope of the PRA performed to support the COL application

Clarifications (4 of 4)

- Summary PRA quantitative results should be provided in Chapter 19 of the FSAR
- COL applications should be complete; RAIs and audits are used to clarify information
- The COL application must be based on a plant-specific PRA; bounding analyses may be used
- The SAMDA evaluation may be included in either the FSAR or the Environmental Report
- The phrase "regulatory oversight processes" refers to items such as MSPI and SDP (not the staff's Reactor Oversight Process – ROP)

Path Forward

- Developing Interim Staff Guidance (ISG) to address clarifications
- Collecting risk insights for technical reviewers from DC PRAs
- Performing QA reviews of EPR and U.S. APWR
- Preparing for acceptance reviews
- Preparing for PRA audits

5

POLICY ISSUE
(Notation Vote)

May 19, 1993

SECY-93-138

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: RECOMMENDATION ON LARGE RELEASE DEFINITION

Purpose:

To provide the Commission the staff's evaluation on the development and usefulness of a large release definition as a plant performance objective and to recommend that work on the development of a large release definition should be terminated.

Background:

On August 4, 1986, the Commission issued its Safety Goal Policy Statement and approved the use of qualitative and quantitative safety goals in the regulatory process (51 FR 28044). The Commission stated that guidance on the use of the safety goals may also include a general performance guideline that was proposed by the Commission for further staff examination.

This guideline was stated as follows:

> "Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

On March 30, 1989, the staff proposed a general framework and Safety Goal Policy Implementation in SECY-89-102 ("Implementation of Safety Goal

NOTE: TO BE MADE PUBLICLY AVAILABLE WHEN THE FINAL SRM IS MADE AVAILABLE

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J. Ridgely
(492-3978)

Policy"). As discussed in SECY-89-102, several different approaches were considered to give more explicit meaning to the term "large release," both quantitatively and qualitatively. At that time, the staff recommended that a large release be defined as follows:

> "A large release is a release that has a potential for causing an offsite early fatality."

In a Staff Requirements Memorandum (SRM) dated June 15, 1990, the staff was requested to re-examine and advise the Commission whether a plant performance objective that focuses on accidental releases from the plant (i.e. large release) and eliminates site characteristics could be developed and be useful.

In response to this direction in SECY-90-405, the staff discussed two possible definitions for a large release, proposed one for further evaluation and provided a plan to evaluate its magnitude utilizing representative site characteristics. The definition proposed was the following:

> "A large release is a release of radioactivity from the containment to the environment of a magnitude equal to or greater than: (An amount, to be determined by the staff, expressed in curies or fraction of the core inventory, which has the potential, based on representative site characteristics, for causing one or more offsite prompt fatalities.)"

The proposed staff evaluation of large release magnitude was to be consistent with the ACRS proposed guidelines linking the hierarchical levels of the safety goal objectives, where the large release guideline was considered the third level objective (the qualitative and quantitative health objectives are the level one and two objectives, respectively). According to these guidelines, each subordinate level of the safety goal objectives should:

1. Be consistent with the level above;
2. Not be so conservative as to create a de facto new policy;
3. Represent a simplification of the previous level;
4. Provide a basis for assuring that the Safety Goal Policy objectives are being met;
5. Be defined to have broad generic applicability;
6. Be stated in terms that are understandable to the public; and
7. Generally comport with current PRA usage and practice.

The Commission, in an SRM dated March 21, 1991, approved the staff proposed definition and evaluation plan, provided guidance for use in selecting the representative site characteristics and requested that the representative site parameters be provided for Commission approval before completing final calculations of a large release. A draft of a paper on the representative site definition was provided to the Commission for information in a memorandum dated October 24, 1991. Several meetings with ACRS have also taken place on this subject over the past two years.

In addition, in a recent memorandum to the staff dated March 2, 1993, Commissioner Remick recognized the difficulty in developing a large release definition that is consistent with the quantitative health objectives and raised the question of whether a large release definition is still needed.

Discussion:

Over the past year and a half the staff has spent considerable time attempting to define a large release magnitude within the framework of the safety goal hierarchy proposed by ACRS and the guidance provided in the Commission's June 15, 1990 and March 21, 1991 SRMs. A discussion of the analyses performed, the methods and assumptions used and the results are contained in the Enclosure 1 to this paper.

The overall conclusion reached by the staff is that development of a large release definition and magnitude, beyond a simple qualitative statement related to the 10^{-6} per year large release frequency (such as is currently contained in the Commission's Safety Goal Policy Statement), is not practical or required for design or regulatory purposes. The factors leading to this conclusion are discussed below. In addition, based upon the work done evaluating the large release, NUREG-1150 and other related activities, the staff notes that the general performance guideline (i.e., large release frequency of 10^{-6} per year) proposed in the Safety Goal Policy Statement and the core damage frequency subsidiary objective (i.e., core damage frequency of 10^{-4} per year) being used by the staff are not consistent with the quantitative health objectives (QHOs) stated in the Safety Goal Policy Statement. This is also discussed below.

Conservatism of the Large Release Guideline

The level two hierarchical safety goal objectives are the quantitative health objectives contained in the Commission's Safety Goal Policy Statement. Stated numerically the 0.1% individual risk values included in the QHOs correspond to:

- prompt fatality goal = 5×10^{-7} per year risk of a prompt fatality to an average individual within one mile of the site boundary
- latent fatality goal = 2×10^{-6} per year risk of a latent fatality to an average individual within ten miles of the site boundary.

Of the two QHOs, the prompt fatality QHO has been found to be the more restrictive objective. It was recognized in SECY-89-102 that, at an overall mean frequency of less than 1 in 1,000,000 per year, any large release definition would represent a QHO inherently smaller than the prompt fatality QHO (the prompt fatality QHO represents a 5×10^{-7} per year risk of a prompt fatality to an average individual within 1 mile of the site boundary). Consideration of wind direction alone (16 possible wind directions in the MACCS calculations) results in about an order of magnitude conservatism. This follows from the definition of an average individual given in the Safety Goal Policy Statement. Specifically, if one uses the mean core damage frequency subsidiary objective of 10^{-4} per reactor year, a conditional containment failure probability of 0.1 (frequency of a release of about 10^{-5} per reactor year), and a probability of a release in a given wind direction of about 1/16, then the maximum risk to an individual can be estimated at $6x10^{-7}$ per reactor year which is approximately the prompt fatality QHO. Similarly if the frequency of a release is taken to be 10^{-6} per reactor year, then the maximum risk to an individual is approximately an order of magnitude less than the prompt fatality QHO. An order of magnitude conservatism was accepted by the Commission in its June 15, 1990 SRM.

However, when the individual risk from a "large release" is evaluated using realistic meteorology, realistic release characteristics and realistic protective actions, several more orders of magnitude conservatism can be and are introduced regardless of how the large release is defined. This can be seen from the results presented in NUREG-1150. For the five plants studied, all plants had a probability of an early containment failure or bypass between 10^{-5} and 10^{-6} per year, yet the prompt fatality risk to the average individual for all plants was over an order of magnitude or more below the prompt fatality QHO. Conversely, the five plants studied in NUREG-1150 could meet the prompt fatality QHO even if the frequency of an early containment failure or bypass was higher by an order of magnitude or more. Also, much higher frequency of core damage can be tolerated without exceeding the quantitative safety goals if one were to base regulatory decisions on the QHOs alone.

Large Release Definition

Given a large release frequency of 10^{-6} per year, any large release definition will result in a degree of conservatism several orders of magnitude more conservative than the QHOs. Furthermore, as discussed in Enclosure 1, the specification of the magnitude of any large release definition is very sensitive to the assumptions used for certain parameters in the calculation. The results will be site, weather, and accident sequence dependent, such as the following parameters:

- o the energy of release (ground level or elevated)
- o the timing and duration of the release
- o the protective action assumptions used
- o the population density of the area immediately surrounding the plant.

Variations in these parameters (given a fixed consequence in terms of risk) can cause the large release magnitude to vary widely and, in some cases, exceed the release estimated to have occurred from the Chernobyl accident. Other parameters can also affect the magnitude to a lesser extent. Expressing the large release magnitude in terms of "equivalent curies of I^{131}" versus fraction of core inventory can eliminate the effect of the timing of release parameter but, the magnitude or quantity released was still subject to wide variation from the assumptions used for the other parameters. Therefore, to implement a large release guideline expressed in terms of a magnitude of radioactive material released to the environment, a prescriptive analytical methodology would be required. Such a prescriptive methodology would tend to be conservative so as to envelop the variations in the above parameters and its implementation would potentially introduce additional conservatism below

NUREG/CR-6094 BNL

the QHO's. This would lead to a prescription that clearly does not represent a useful simplification relative to the implementation of the quantitative health objectives themselves.

The staff also evaluated a qualitative alternative large release definition as discussed in SECY-90-405 to see if this was practical. This definition was:

> "A large release is any release from an event involving severe core damage, reactor coolant system pressure boundary failure, and early failure or significant bypass of the containment."

This evaluation focused on defining reasonable values for "early" and "failure" and whether these were more useful and subject to large variation from the calculational assumptions. Using the same representative site characteristics and based on one or more prompt fatalities, the analyses indicated that "early" would be defined as occurring within approximately the first 24 hours following the onset of core damage.[1] However, the value of early was also subject to large variations depending upon the other assumptions used. The staff also found it difficult to define "failure." The staff's efforts were directed toward defining "failure" in terms of containment leak rate, but the staff also found that the value was subject to large variations depending on the assumptions used and would not represent a performance objective that would be simple to understand and readily useable in the regulatory process.

Overall, it was concluded that implementation of this alternative qualitative definition might represent a degree of simplification compared to a quantitative definition, however, it would still result in a large release guideline several orders of magnitude more conservative than the safety goal prompt fatality quantitative health objective.

Conclusion:

In parallel with work evaluating a large release described above, the staff developed interim guidance regarding implementation of the Commission's Safety Goal Policy (SECY-91-270 and SECY-93-043) and criteria for use in the certification review of advanced LWR designs (SECY-90-016 and SECY-93-087).

The interim guidance proposed in the revised Regulatory Analysis Guidelines (SECY-93-043) provides a framework for regulatory decision making utilizing the core damage frequency (CDF) and conditional containment failure probability (CCFP) as the subsidiary safety goal objectives approved for use in the Commission's June 15, 1990 SRM. The staff has recommended that this guidance be issued for public comment. This framework, if adopted, represents

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[1] In NUREG-1150, early containment failure is defined as: "Those containment failures occurring before or within a few minutes of reactor vessel breach for PWRs and those failures occurring before or within 2 hours of vessel breach for BWRs. Containment bypass failures (e.g., interfacing system loss-of-coolant accidents) are categorized separately from early failures."

a relatively easy to use tool to assess the need for proposed changes in the existing rules and generic requirements considering both core melt prevention and containment performance and is consistent with the 10^{-4} per reactor year CDF contained in Commission guidance and the 10^{-6} per reactor year large release frequency proposed in the Commission's Safety Goal Policy Statement. Further, this approach represents use of the CDF and CCFP values as up-front screening criteria to determine whether to proceed to an in-depth value-impact analysis on proposed rules and generic requirements. We believe this approach would be consistent with the Commission's intent to maintain the defense-in-depth principle. That is, keep the frequency of a core damage event low in the U.S. plants and to assure that the containment provides appropriate mitigation capability to limit the releases.

The criteria developed for the advanced LWR designs provides for containment performance objectives that could be used to assess overall plant performance in regard to the containment of radioactive material from severe accidents, even though the staff has acknowledged that such criterion is likely to result in several orders of magnitude conservatism relative to the constraining QHO. In addition to capturing the containment performance criteria through design certification rulemakings, the criteria associated with containment performance may become codified via rulemaking as discussed in SECY-92-292, "Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future LWRs."

Given the above activities, the staff believes that the need for a precise large release definition has diminished in importance. This, coupled with the difficulty in developing a useful and coherent large release definition and magnitude, has led us to conclude that it is not necessary to further pursue a large release definition or magnitude. Instead, the staff would propose to use the guidance developed for regulatory decisionmaking for assessing reactor designs and safety issues, i.e., Regulatory Analysis Guidelines, for existing plants and Commission approved criteria for reviewing the acceptability of advanced reactor designs.

<u>Coordination</u>:

On April 15, 1993, the staff briefed the Advisory Committee on Reactor Safeguards on the staff's recommendation regarding development of a definition of a large release. The Committee provided its views to the Commission in a letter dated April 22, 1993 (Enclosure 2) and agreed with the staff's recommendation to terminate efforts to develop a definition of a large release.

The Office of General Counsel has reviewed this paper and has no legal objection.

Recommendation:

That the Commission direct the staff to terminate further work on the development of a large release definition and magnitude.

James H. Sniezek / for
James M. Taylor
Executive Director
for Operations

Enclosure: As stated

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Wednesday, June 2, 1993.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Wednesday, May 26, 1993, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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Enclosure 1

Calculations for a Large Release Definition

In examining the definition of a large release, the staff used the Melcor Accident Consequence Code System (MACCS), which was also used in NUREG-1150 to calculate offsite consequences. To calculate consequences, MACCS requires input related to site characteristics (wind speed and direction probabilities; rainfall quantity and duration; population distribution; and exclusion area distance), protective action assumptions, and source term characteristics (thermal energy, timing, duration, and composition of release). These parameters can affect the characteristics of a release that meets the proposed large release definition in SECY-90-405, i.e., having the potential of causing one or more prompt fatalities.

In its March 21, 1991 SRM, the Commission provided guidance in determining the characteristics of a representative site. Specifically, real site characteristics that are representative of sites that would fit within the envelope of the proposed revision of 10 CFR Part 100 were to be used. In addition, the selection of the site characteristics was to reflect a conservative approach such that the resulting site model would encompass the calculated consequences of any actual site. In developing representative site characteristics and determining potential large release magnitudes in accordance with the Commission's guidance, the staff performed sensitivity studies on selected MACCS input parameters. Key parameters evaluated are discussed below.

- Exclusion Area Distance: The staff is currently proposed to codify in Part 100 an exclusion area distance of 0.4 miles. The staff performed sensitivity studies of the effect of varying the exclusion area distance between 0.17 and 1.33 miles in ensuring that the prompt fatality QHO is not exceeded. Results are shown in Table 1 in terms of fraction of I^{131} released and in equivalent curies. Because the effect of varying the exclusion area distance was relatively small and to maintain consistency with the revision of Part 100, as discussed above, a value of 0.4 miles was used in characterizing the representative site.

Table 1 - Effect of Exclusion Area Distance

Release Duration (Hours)	Sheltering Time (Hours)	Radius (Miles)	% I^{131} of Core Inventory (3800 MWT)	Equivalent Curies
1	12	0.40[a]	60	5.8E+7
1	12	0.17[b]	42	4.0E+7
1	12	0.55[c]	73	7.0E+7
1	12	1.33[d]	93	8.9E+7

Notes: [a] Base (Proposed) Case
[b] Smallest distance of existing plants (SECY-90-341)
[c] Average distance of existing plants (SECY-90-341)
[d] Largest distance of existing plants (SECY-90-341)

- Population Density: The staff is currently proposing to codify the population density guidelines of Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," in the proposed revision to Part 100. For evaluation purposes, this includes an initial value of 500 people per square mile within 30 miles of the reactor and a 40 year projected population density below 1000 people per square mile. In an effort to ensure the representative site model would encompass the consequences of any actual site, while staying within the envelope of the proposed Part 100, a uniform population density of 1000 people per square mile was used for the representative site. For comparison, the average population density for current plants within two miles of the plant is approximately 125 people per square mile. If the population density for the representative site is reduced from 1000 to 100 people per square mile (an order of magnitude), a release of approximately twice the equivalent curies of I^{131} ($1.2x10^{8}$) would be required to result in one prompt fatality. A release of this magnitude assumed to occur at 2 hours after the reactor shutdown is equivalent to approximately 100% noble gases, 9% iodine, 9% cesium, and 2% tellurium.

- <u>Meteorology</u>: The meteorological parameters used in the MACCS consequence calculations are the atmospheric stability or dispersion characteristics, the amount and duration of rainfall, and the wind direction frequency (percentage of the time that the wind blows in a given sector/direction). During the staff's efforts to define a large release as a fraction of core inventory, two cases were examined: a mean value case and an 80^{th} percentile case.

  The effect of varying the meteorology on the number of prompt fatalities was found to be small. Therefore, reflecting a conservative approach, the 80^{th} percentile meteorology was used for the majority of the sensitivity calculations.

- <u>Protective Action Assumptions</u>: During the staff's efforts to define a large release as a fraction of core inventory, the staff considered four cases. The first case used the mean of the assumptions made in NUREG-1150 (99.5% of the people evacuate with an evacuation speed of 5.9 mph and after a delay time of 1.9 hours).

  In addition, the staff evaluated several protective action assumptions that were more conservative. The first conservative case assumed that only 95% of the people evacuate at a speed of 2.5 mph after a delay time of six hours.

  A second conservative case, which assumed no protective actions, i.e., the public continues their normal activities, was evaluated in an attempt to decouple the "large release" definition from protective action assumptions. This case, assuming a ground level release, results in a relatively small magnitude of a "large release". This is a result of the long term groundshine dose. (For this case, the MACCS code assumes that people will continue normal activities for seven days after a release.)

  Because the no protective action assumption was considered overly conservative, an additional conservative case was evaluated in which it was assumed that the population is sheltered for 12 hours, after which 100% of the people were relocated. These analyses resulted in a number of different potential "large release" magnitudes ranging from releases of 100% of the noble gases and a few percent of the iodine to release magnitudes of 100% of the noble gases and 20-30% of iodine and cesium along with significant amount of other isotopes. Much of this wide variation resulted from assumptions of the timing of the release in relationship to the warning time for evacuation.

Because a "large release" performance guideline is for use in assessing plant performance, staff believes it should not be dependent on protective action assumptions. Therefore, to reasonably decouple the large release definition from protective action assumptions, while avoiding overly conservative assumptions, the assumption of 12 hour sheltering, followed by 100% relocation, is a reasonable one for use in determining a large release magnitude.

However, during its efforts to define a large release in terms of equivalent curies of I^{131}, staff did evaluate an eight hour sheltering case and a case with no protective actions. The results of these analyses are shown in Table 2.

Table 2 - Effect of Protective Actions

Release Time (Hours)	Release Duration (Hours)	Sheltering Time (Hours)	% I^{131} of Core Inventory	Equivalent Curies	Fraction of Core Inventory			
					Nobles	Iodine	Cesium	Te
2[a]	1	12	60	5.8E+7	100	2	2	.1
2	1	8	67	6.4E+7	100	2.5	2.5	.1
2	1	None[b]	26	2.5E+7	93	0	0	0

Notes: [a] Base (Proposed) Case
[b] Assumes normal activity for seven days

- Energy of Release: In addition to the magnitude or the quantity of radioactivity released, the likelihood of prompt fatality also depends on the thermal energy and the timing of the release. Releases having high thermal energy will ascend to elevated levels in the atmosphere and are more dispersed compared with low energy releases close to the ground. For this reason, high thermal energy releases require a greater magnitude of radioactive material to be released to result in a prompt fatality compared to low energy releases. If a high energy release is assumed, instead of a low energy release, the magnitude of a "large release" would increase from 6×10^7 to 2×10^8 equivalent curies of I^{131}. A release of this magnitude at two hours after reactor shutdown would be equivalent to approximately 100% noble gases, 14% of iodine and cesium and 4% tellurium. Since both high energy and low energy releases are possible, the staff used a low energy release in defining a large release.

- Timing & Duration of Releases: The timing of the release is important in two ways. First, if assuming evacuation, the time of release becomes important in relationship to the time and speed of evacuation. Secondly, because of the decay of short lived isotopes, a late release will require larger fractions of the core inventories to be released to result in one or more prompt fatalities. When expressed in terms of equivalent curies I^{131}, the release magnitude is independent of the time of the release. Table 3 shows the effect of different release times on the radioactive material needed to be released to result in one prompt fatality.

Table 3 - Effect of Release Time

Release Time (Hours)	Release Duration (Hours)	Fraction of Core Inventory			
		Nobles	Iodine	Cesium	Te
2	1	100	2.0	2.0	<1
4	1	100	4	4	<1
12	1	100	8.0	8.0	2
24	1	100	11	11	3
Note: All releases equivalent to a release of 6E7 equivalent curies I^{131}					

During the staff's efforts to define a large release in terms of equivalent curies I^{131}, the staff assumed a one hour release duration to ensure that all of the releases occurred early in the sheltering scenario. However, the staff did evaluate the effect of a longer duration release. The results of these analyses are shown in Table 4.

Table 4 Effect of Release Duration

Release Duration (Hours)	% I^{131} of Core Inventory	Equivalent Curies	Fraction of Core Inventory			
			Nobles	Iodine	Cesium	Te
1	60	5.8E+7	100	2	2	.1
2	91	8.7E+7	100	5	5	1
4	99	9.5E+7	100	6.8	6.8	1.7

- Other Modeling Parameters: In addition to the site parameters discussed above, other modeling parameters also were found to affect the results of the calculations. These parameters include shielding factors assumed during sheltering, binning of the weather data, and grid size used to represent population distribution. Therefore, if a quantitative definition of a large release is used, further consideration would need to be given to the selection of these parameters.

In an effort to develop a large release magnitude that would be readily useable for assessing plant performance, the staff initially focused on establishing a large release magnitude in terms of fractions of core inventory released to the environment. It was believed that this would allow relatively easy comparison with the results of Level II PRAs and could later be converted to equivalent curies for use with reactors of significantly different power levels. Using the range of releases developed for the five plants examined in NUREG-1150 and the LaSalle plant, a set of simplified source terms were developed to reflect a range of release characteristics, including timing, duration, energy, and composition (fractions of core inventory). These simplified source terms, adjusted to an assumed, enveloping, reactor power level of 3800 megawatts (thermal), and the range of site characteristics discussed above were used in the MACCS code to identify those releases that would most nearly lead to one prompt fatality.

Discussion

A. Fraction of Core Inventory

As noted above, the staff performed a number of sensitivity studies to determine the effect of varying selected MACCS code inputs. The important parameters that affect the magnitude of a "large release" are the release characteristics (including the energy of the release and the timing of the release with respect to reactor shutdown) and the protective actions assumed. The effects of varying exclusion area size and meteorology were found to be relatively small. Furthermore, the impact of varying the assumptions of time and duration of a release is dependent on the protective action assumptions and conversely, the impact of varying protective action assumptions depends on the time of a release. When expressed in terms of equivalent curies of I^{131}, the time of release is important, primarily in conjunction with the protective action assumptions.

The sensitivity calculations resulted in a wide range of possible large release magnitudes, when expressed in terms of fraction of core inventory. Assuming a low energy release occurring at approximately 6 hours after reactor shutdown and a bounding assumption of no protective actions, a release of 100% of the noble gases, 3% of the iodine and less than 1% of other isotopes would meet the definition of a "large release," i.e, result in one prompt fatality. Such a release is small enough that many plants likely would not be able to show that its likelihood was less than 10^{-6} per year over the spectrum of potential severe accidents. Assuming an energetic release, instead of a ground level release, the resulting magnitude of a large release would be increased to 100% noble gases, 10% iodine, cesium and tellurium and less than 5% of other isotopes to result in one prompt fatality. The effects of assuming realistic protective action assumptions are even more pronounced. If an energetic release occurring at approximately 12 hours after reactor shutdown is assumed and mean NUREG-1150 protective action assumptions are used, a release of 100% of the noble gases, 25-30% of the iodine and cesium, 10-20% of the tellurium and strontium, and less than 10% of other isotopes would be required to meet the definition of a large release. A release of this magnitude is estimated to be larger than that which occurred at Chernobyl. This wide variation in a potential definition of a "large release" suggests that a quantitative definition may not be practical.

While performing this work, it became apparent that a key factor affecting the magnitude of a large release, when expressed in terms of fraction of core inventory, is the timing of the release. Because of the wide variations in possible "large release" magnitudes and the difficulties envisioned in justifying any single "large release" expressed in terms of fraction of core inventory, the staff subsequently focused on establishing a large release magnitude in terms of equivalent curies of I^{131} released to the environment. This approach eliminates much of the variation in magnitude that results from timing of the release. However, the other key factors that can affect the magnitude (energy of release and protective action assumed) remain.

B. Equivalent Curies of I^{131}

As discussed above, the staff found that the magnitude of a large release, when expressed in terms of fraction of core inventory, varied widely and was very sensitive to assumptions on timing and energy of the release and protective actions taken. In order to reduce the sensitivity of the large release magnitude to the assumptions of timing of release, the staff has also evaluated the magnitude of a large release expressed in terms of curies released. Accordingly, the staff has evaluated a large release magnitude in terms of equivalent curies released, using I^{131} as the representative isotope. The variation of the magnitude, based on variations in energy of release and protective actions assumed, was from approximately $2x10^7$ curies to $4x10^8$ curies (over an order of magnitude difference). For a candidate "large release" magnitude, the staff assumed a low energy release of 1 hour duration with the following "representative site" characteristics:

- o Exclusion area boundary - 0.4 miles
- o Population density - 1000 people per mi^2
- o Protective actions - 12 hr. sheltering followed by 100% relocation
- o Meteorology - 80^{th} percentile

The resulting magnitude of the "large release" is $6x10^7$ curies of I^{131}, which is equivalent to approximately 60% of the core inventory of I^{131}. This is equivalent to a release of approximately 100% of the noble gases and 2% of the iodine and cesium at 2 hours after reactor shutdown (less than 1% of any other isotopes released). At 12 hours after reactor shutdown, this is equivalent to a release of approximately 100% noble gases, 8% iodine, 8% cesium, and 2% tellurium.

Enclosure 2

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

April 22, 1993

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: DEFINITION OF A LARGE RELEASE FOR USE WITH SAFETY GOAL POLICY

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed the staff's recommendations in regard to the definition of a large release related to the implementation of the Commission's Safety Goal Policy. During this meeting, we had the benefit of discussions with members of the NRC staff and of the document referenced.

In the draft Commission paper and in the presentation to the Committee, the staff expressed its belief that the development of the definition of a large release is no longer practical or useful and, therefore, it is requesting Commission approval to terminate efforts in this area. We believe the staff has made a conscientious effort with this activity and we agree with its basic conclusions. Our views are as follows:

1. A large release definition would either represent a replacement for the existing safety goals or, if made consistent with the quantitative health objectives (QHOs), would be redundant and unnecessary.

2. New guidelines being developed for implementing the Safety Goal Policy within regulatory analysis and issue prioritization processes adequately meet the originally perceived need for a large release component of the safety goals. These utilize a core damage frequency (CDF) and a conditional containment failure probability (CCFP).

3. Plant performance objectives, viz CDF $\leq 10^{-4}$ and CCFP ≤ 0.1, provide an easily understandable and adequate surrogate for the QHOs and provide quantitative prioritization for two basic aspects of defense in depth (prevention and mitigation). These could help ensure that a plant does not end up with great core protection but marginal containment performance.

We support the recommendation that the Commission approve the staff's proposal to terminate its effort to develop a definition of a large release.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

Reference:
Memorandum dated March 11, 1993, from Warren Minners, Director, RES/DSIR, for John T. Larkins, Acting Executive Director, ACRS, Subject: ACRS Review of Draft Commission Paper on Large Release Determination, w/Enclosure

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INFORMATION BRIDGE

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Subject	22 GENERAL STUDIES OF NUCLEAR REACTORS; 61 RADIATION PROTECTION AND DOSIMETRY; REACTOR ACCIDENTS; EMERGENCY PLANS; FISSION PRODUCT RELEASE; LETHAL RADIATION DOSE; CLASSIFICATION; PROBABILISTIC ESTIMATION; COMPILED DATA; RADIATION PROTECTION; SOURCE TERMS
Description/Abstract	The Nuclear Regulatory Commission has stated a hierarchy of safety goals with the qualitative safety goals as Level I of the hierarchy, backed up by the quantitative health objectives as Level II and the large release guideline as Level III. The large release guideline has been stated in qualitative terms as a magnitude of release of the core inventory whose frequency should not exceed 10{sup -6} per reactor year. However, the Commission did not provide a quantitative specification of a large release. This report describes various specifications of a large release and focuses, in particular, on an examination of releases which have a potential to lead to one prompt fatality in the mean. The basic information required to set up the calculations was derived from the simplified source terms which were obtained from approximations of the NUREG-1150 source terms. Since the calculation of consequences is affected by a large number of assumptions, a generic site with a (conservatively determined) population density and meteorology was specified. At this site, various emergency responses (including no response) were assumed based on information derived from earlier studies. For each of the emergency response assumptions, a set of calculations were performed with the simplified source terms; these included adjustments to the source terms, such as the timing of the release, the core inventory, and the release fractions of different radionuclides, to arrive at a result of one mean prompt fatality in each case. Each of the source terms, so defined, has the potential to be a candidate for a large release. The calculations show that there are many possible candidate source terms for a large release depending on the characteristics which are felt to be important.
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Subject	22 GENERAL STUDIES OF NUCLEAR REACTORS; 61 RADIATION PROTECTION AND DOSIMETRY; REACTOR ACCIDENTS; EMERGENCY PLANS; FISSION PRODUCT RELEASE; LETHAL RADIATION DOSE; CLASSIFICATION; PROBABILISTIC ESTIMATION; COMPILED DATA; RADIATION PROTECTION; SOURCE TERMS
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NUREG/CR-6595, Rev. 1
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An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events

Final Report

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GI-156.6.1

PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS INSIDE CONTAINMENT

Harold VanderMolen RES/DRASP/OEGIB

Abdul Sheikh RES/DFERR/ERA/MS

Outline

- Issue Description
- Historical Background
- Idaho Screening Analysis
- BWR investigation
- PWR investigation
- Conclusion – Issue can be closed out with no new requirements

Safety Question

- SRP contains specific criteria for postulated pipe break locations, pipe whip restraints, and I&C separation criteria
- Many plants were designed & built <u>before</u> the first SRP was issued in 1975
- Are there possible interactions due to pipe whip and/or jet impingement in these older plants?

Affected plants

- 51 units originally within the scope of this generic issue
- 10 units permanently shut down
- 18 BWRs still operating
- 23 PWRs still operating

History of GI-156.6.1

Begin Systematic Evaluation Program (SEP)	1977
Integrated Safety Assessment Program (ISAP)	1984
SEP program terminated	1990
Remaining open SEP issues transferred to GI program – became GI-156 group	1991
GI-156.6.1 given “Medium” priority	1994
“Enhanced” screening assessment of GI-156.6.1	1999

Idaho screening assessment

- Reviewed FSARs
- Reviewed Integrated Plant Safety Assessment Report
- Reviewed design changes made after SRP issuance
- Performed five site visits
- Developed first-level list of "concerns"
- Narrowed list down to second-level list based on site visits
- Developed initial probabilistic screening to further reduce the list

Idaho analysis results

BWRs

- BWR Mark I all similar
- Design tends to encourage 180° separation
- Water level reference columns & pressure sensors are outside of primary containment
- Dominant sequences involve drywell puncture

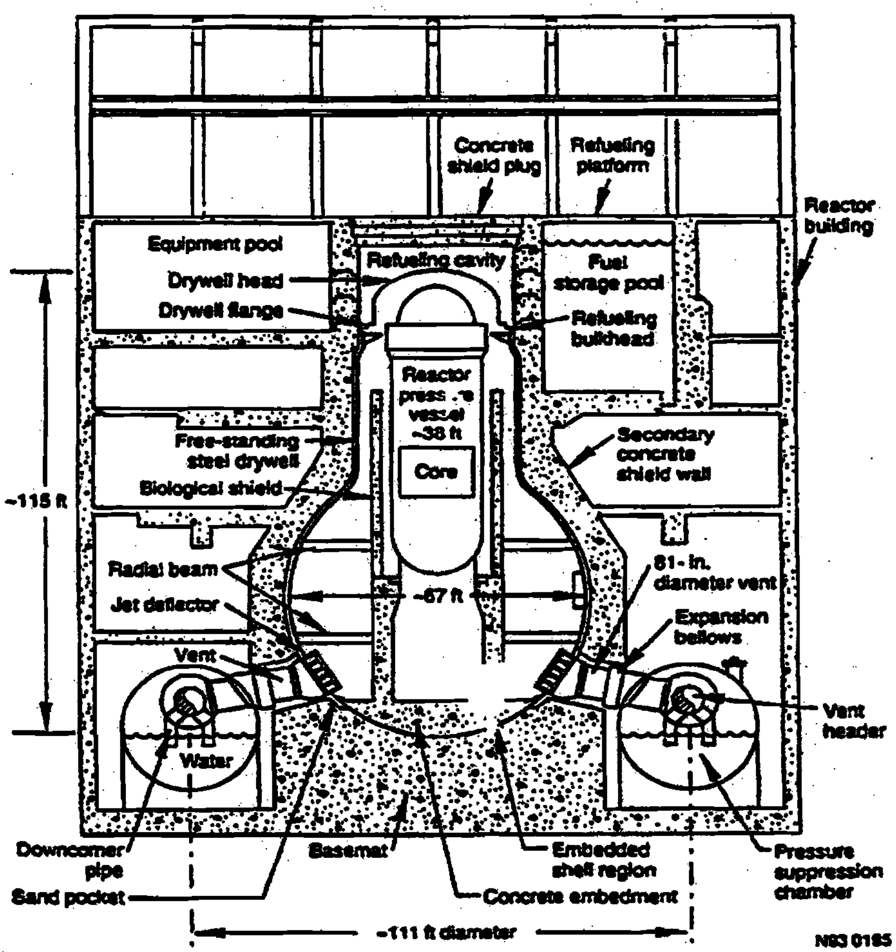
PWRs

- PWR containments vary widely
- Compartmentalization and seismic restraints reduce primary system interactions
- Dominant sequences involve secondary system breaks near electrical penetrations

BWR Scenarios

- Whipping pipe impacts and penetrates steel drywell wall
- Steam discharges into gap between drywell wall and concrete secondary shield wall
- Steam exits gap area, enters area surrounding torus
- Hostile environment disables LPCI, core spray
- Result could be severe core damage with failure of primary containment

BWR layout

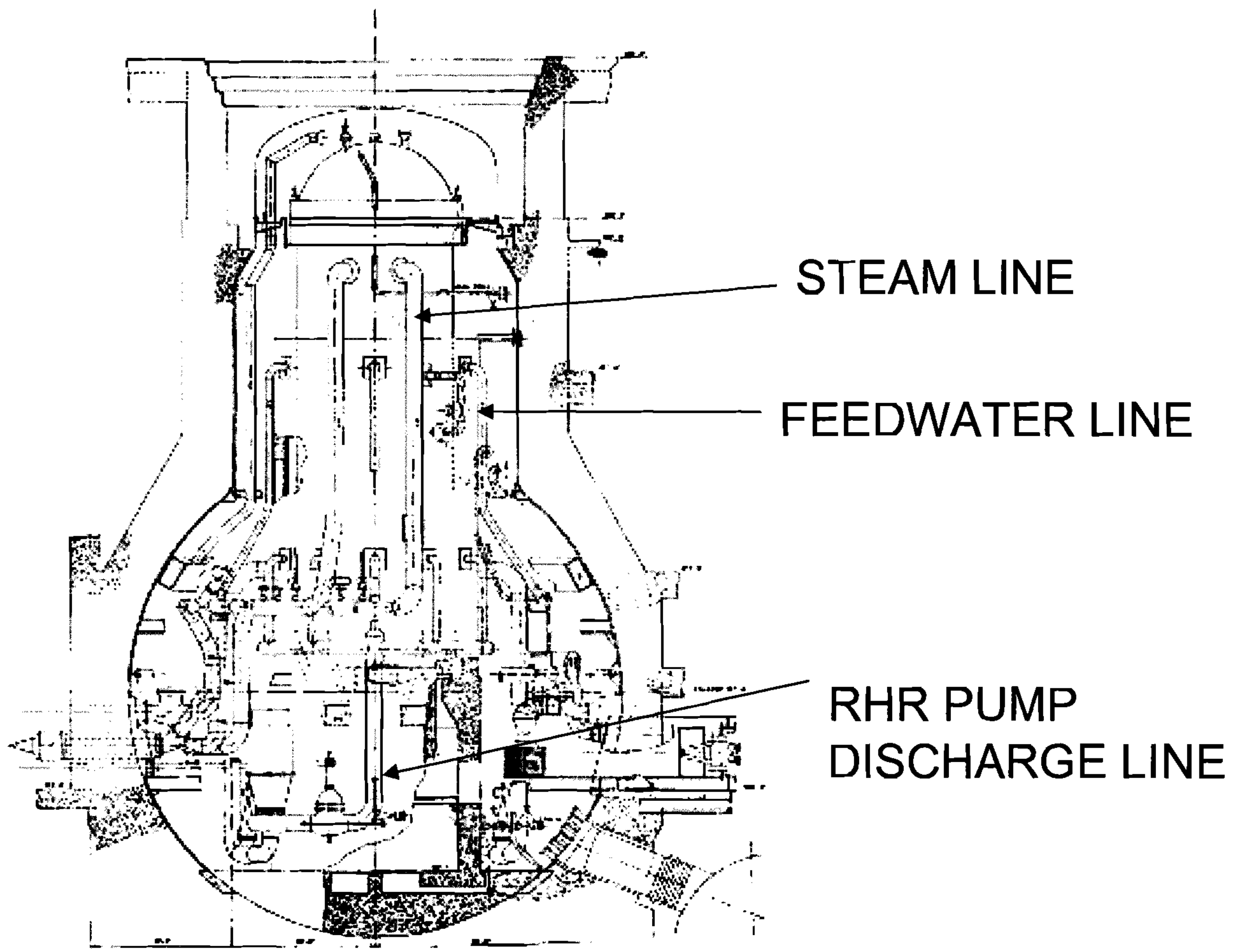


Concrete shield plug
Refueling platform
Reactor building
Equipment pool
Drywell head
Drywell flange
Refueling cavity
Fuel storage pool
Refueling bulkhead
Reactor pressure vessel ~38 ft
Core
Free-standing steel drywell
Biological shield
Secondary concrete shield wall
~115 ft
Radial beam
Jet deflector
~67 ft
81-in. diameter vent
Expansion bellows
Vent
Vent header
Water
Downcomer pipe
Sand pocket
Basemat
Embedded shell region
Concrete embedment
Pressure suppression chamber
~111 ft diameter
N93 0185

Pipe Impact on Steel Drywell

- Postulated Pipe Breaks Inside Drywell
  - Main steam pipe at reactor nozzle
  - Feedwater pipe at reactor nozzle
  - RCS pumps discharge lines at reactor nozzle
- Structural Evaluation
  - ANSYS computer code
  - Lower and upper bound values of blowdown force
  - Minimum thickness of drywell (0.64 inch)
  - Maximum gap between drywell steel and concrete shield (3.125 inch). Normal as-built gap 2.0 inch

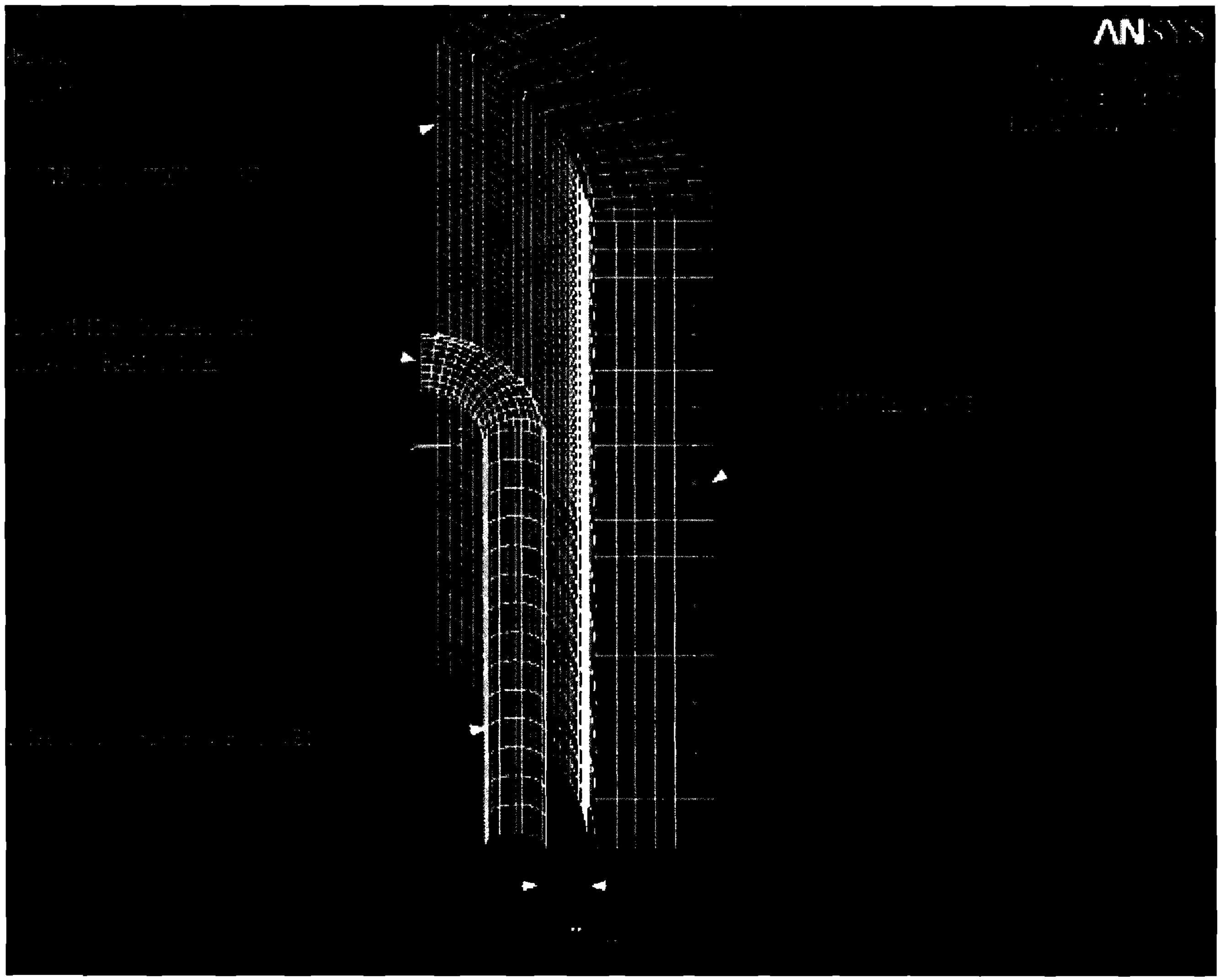
BWR Mark I Piping



Main Steam Line Break

- Pipe
  - Diameter: 24 inch
  - Wall Thickness: 1.30 inch
  - Minimum ultimate strain of pipe material: 22%
  - Gap between drywell and steam line: 16 inch
  - Operating pressure: 1050 psi
- Double ended guillotine break
- Pipe whip force: 0.70 to 1.2 PA
- Maximum drywell strain: 10%
- Drywell will deflect and come into contact with concrete shield
- Drywell will not perforate
- Containment drywell Integrity would not be compromised

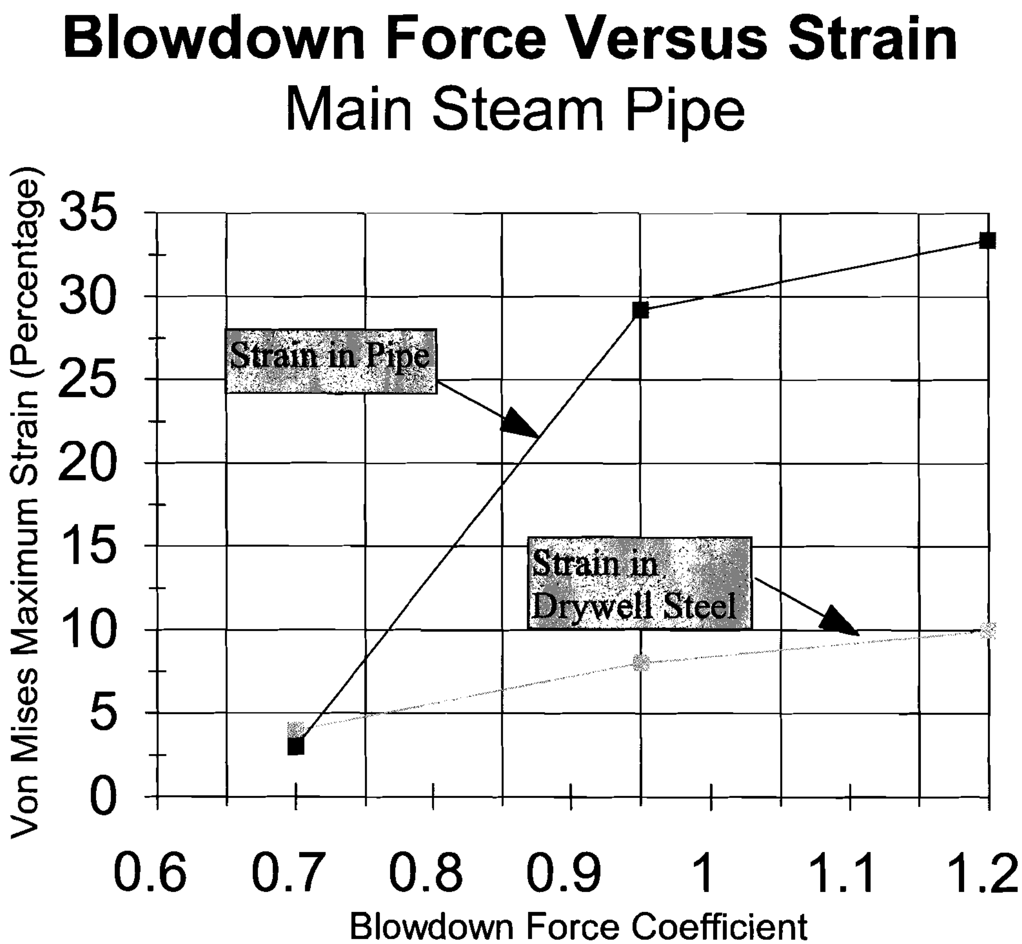
Main Steam Line ANSYS Model



Main Steam Line and Drywell Deflected Shape

Drywell in touch with concrete and pipe

Steam Line and Drywell Strains



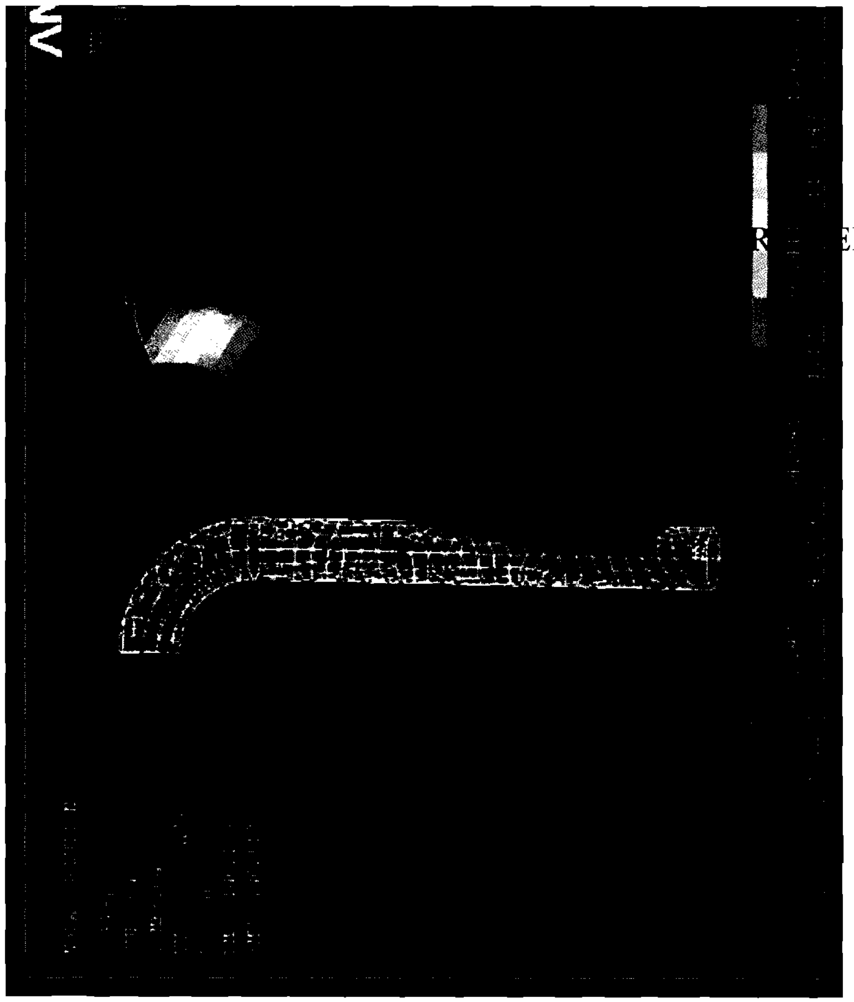
Feedwater Line Break

- Pipe
  - Diameter: 10.75 inch
  - Wall Thickness: 0.625 inch
  - Minimum ultimate strain of pipe material: 22%
  - Gap between drywell and steam line: 24 inch
  - Operating pressure: 1050 psi
- Double ended guillotine break
- Pipe whip force: 1.3 - 2.1PA
- Pipe would deflect 6-18 inches
- Plastic hinge formed
- Pipe would not impact drywell located 24 inch away before failure
- Pipe may impact drywell after failure
- Drywell would not perforate
- Containment drywell Integrity would not be compromised

RCS Pipe Break

- Pipe
  - Diameter: 28.00 inch
  - Wall Thickness: 1.5 inch
  - Minimum ultimate strain of pipe material: 40%
  - Gap between drywell and steam line: 168 inch
  - Operating pressure: 1050 psi
- Double ended guillotine break
- Pipe whip force: 1.3 - 2.2 PA
- Pipe would deflect 62-146 inches
- Pipe impact and damage structural steel beams or PCS piping
- Plastic hinge formed
- Pipe would not impact drywell located 168 inch away before failure
- Pipe may impact drywell after failure
- Drywell would not perforate
- Containment drywell Integrity would not be compromised

RCS Pipe Deflected Shape



Conclusion - BWRs

- Containment penetration scenario does not appear to be credible
- Therefore, there is insufficient basis to justify any further regulatory action

PWR scenarios

- Initiated by pipe break within containment
- Pipe whip or fluid jet disables a system needed to mitigate the break

PWR scenarios (continued)

- Pipes are equipped with seismic restraints, limiting pipe whip
- PWR containments are compartmentalized. A break in one primary loop cannot cause failure of equipment in another loop or in the pressurizer
- Concluded that primary system break very unlikely to initiate this scenario

PWR Scenarios (continued)

- Secondary system piping not necessarily separated by walls
- Secondary system piping will have seismic restraints, but fluid jets could impact I&C cables
- I&C cables will be dispersed except near penetrations
- Biggest vulnerability likely to be a secondary pipe break near cable penetrations

PWR Scenarios (continued)

- Safety systems will still actuate on high containment pressure
- However, loss of I&C cables may interfere with long-term recovery

PWR investigation strategy

Every PWR unique

- Examined FSARs
- Examined plant diagrams
- NRR assisted – resident inspectors & licensee personnel

Looked for:

- >90° separation
- Intervening walls
- Intervening floors
- Large difference in elevation

PWR investigation results

- Nine units have two electrical penetration areas separated by 90° or more
- 10 units have single electrical penetration area, but
  - have concrete floors or walls separating electrical penetrations from piping
  - Have significant distance between electrical penetrations and piping
  - Have some combination of the above

[continued]

PWR Investigation Results (continued)

- Two units had an analysis of piping stresses which concluded that the piping, if overstressed, would break at a location which would not spray water on electrical penetration area
- Two units had the electrical penetration area partly shielded by a concrete floor. A steam or feedwater pipe could disable one channel of temperature instrumentation and one bank of pressurizer heaters, but not both channels.

Conclusion – PWRs

- No plant found to have a significant vulnerability
- Therefore, there is insufficient basis to justify any further regulatory action

Final Recommendation

- Generic Issue 156.6.1 be closed out
- ACRS concur in letter to EDO

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Fire Protection Program Briefing for ACRS

Office of Nuclear Reactor Regulation
Division of Risk Assessment
Fire Protection Branch

September 6, 2007

1

Briefing Objective –

Alex Klein – AFPB Acting Branch Chief

- For the Office of Nuclear Reactor Regulation (NRR) Fire Protection Branch (AFPB) to provide ACRS a status update on key fire protection program activities
- Additional near-term ACRS interactions are also anticipated

2

Topics

- 10 CFR 50.48(c): NFPA 805 Transition
  - Paul Lain – Senior Fire Protection Engineer
- Multiple Spurious Actuations (MSAs)
  - Daniel Frumkin – Acting AFPB Team Leader
- Post-Fire Operator Manual Actions
  - Peter Barbadoro – Fire Protection Engineer
- Hemyc and MT Generic Letter
  - Daniel Frumkin – Acting AFPB Team Leader

3

10 CFR 50.48(c): NFPA 805 Transition – Paul Lain

- Status
- Lessons Learned
- Transition
- Guidance

4

NFPA 805 – Status

- Letters of Intent for 42 Units at 27 Sites
- 37 Units at 23 Sites are Actively Transitioning
- 36 Month Discretion Period to Transition
- Nine Pilot Observation Visits
- Frequently Asked Question (FAQ) Process
- 14 Public Meetings w/ NEI 805 Task Force
- Non-Pilot Update at the NEI FP Info Forum

5

NFPA 805 – Lessons Learned

- PRA Compartmentation
- Ignition Frequency Database
  - Counting Electronic Cabinets
  - Counting HEAF Sources
- Configuration Control
- NEI 04-02, Appendix B Table Details
- LP/SD Qualitative Review
- Carrying Forward Existing Licensing Bases

6

NFPA 805 – Transition

- Pilots
  - Two more Observation Visits
  - Staff Review their Fire PRAs of Pilots
  - LAR Submittal expected next May/June '08
- Non-Pilots
  - Complete their Fire PRAs
  - Conduct Fire PRA Peer Reviews
  - LAR Submittal Start in Nov/Dec '08

7

NFPA 805 - Guidance

- NUREG/CR-6850
- NUREG-1824
- FAQ RIS
- NEI Fire PRA Peer Review Guidance
- ANS Fire PRA Standard
- NEI 04-02 Revision Scheduled for Dec/Jan
- RG 1.205 Revision
- Standard Review Plan
- Post-Transition Inspection Procedures

8

Multiple Spurious Actuations (MSAs) – Dan Frumkin

- Background
- Highlights of NEI's Multiple Spurious Actuation Resolution Methodology
- NRC Staff's Views of the NEI Methodology
- Next Steps

9

MSAs - Background

- NRC Staff proposed Generic Letter (GL) 2006-XX requesting licensees to confirm compliance in light of the relatively high probability of multiple spurious actuations
- Commission disapproved issuing proposed GL in SECY/SRM-06-0196, "Issuance of Generic Letter 2006-xx, 'Post-Fire Safe-Shutdown Circuits Analysis Spurious Actuations'" December 1, 2006
- NRC staff continues to use the SECY/SRM-06-0196 for direction
- NRC staff met with Industry and received Industry's methodology of a method in 02/2007
- Industry presented their detailed methodology to address multiple spurious actuation on September 6, 2007

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MSAs - Highlights of NEI's Multiple Spurious Actuation Resolution Methodology

- Uses insights regarding MSA's of concern based on systems interactions developed by owners groups
- The NEI resolution methodology uses risk information when available but an expert panel is used for completeness
- NEI proposes that the methodology applies to III.G.1 and III.G.2
- The technical aspects of the framework would be applicable to all non-805 plants

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MSAs - NRC Staff's Views of the NEI Methodology

- Proposed methodology includes consideration of risk in determining compliance outside of 10 CFR 50.48(c)
- Cumulative and synergistic effects should be considered, which may not be effectively considered by an expert panel
- If PRA methods or tools are used, these methods or tools should be of adequate detail and quality
- Need to consider MSAs in III.G.3 (III.L) areas

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MSAs - Next Steps

- NRC staff will continue to engage NEI to address MSA's
- Commission directed in SECY/SRM-06-0196, that the NRC staff should continue to encourage licensees to transition to 10 CFR 50.48(c), NFPA 805, the agency's risk-informed, performance-based fire protection rule.

13

Post-Fire Operator Manual Actions (OMAs) – Peter Barbadoro

- SECY/SRM-06-0010
- Status of Issuance of NUREG-1852
- Final Remarks

14

OMAs - SECY/SRM-06-0010 –

"Withdraw Proposed Rulemaking - Fire Protection Program Post-Fire Operator Manual Actions"

- Proposed rule has been withdrawn
- Standard Review Plan (SRP) Section 9.5.1, "Fire Protection Program", and Inspection Procedure (IP) 71111.05T, "Fire Protection [Triennial]," have been updated
- Regulatory Issue Summary 2006-10, "Regulatory Expectations with Appendix R Paragraph III.G.2 Operator Manual Actions," issued June 30, 2006
- Reactor Oversight Process continues to verify compliance with regulations and commitments

15

OMAs - Status of Issuance of NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire"

- NUREG-1852 addresses the performance of post-fire operator manual actions
- Public comments have been dispositioned
- July 13, 2007 ACRS letter stated that NUREG-1852 should be published as final
- CRGR Meeting August 8, 2007, awaiting final CRGR position.
- NUREG-1852 will be issued following acceptable review by CRGR

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OMAs - Final Remarks

- Licensees are expected to bring operator manual actions back into compliance as described in RIS 2006-10
- NRC Staff intends to use NUREG-1852 for future licensing actions or exemptions relating to the use of post-fire operator manual actions

17

Hemyc and MT Generic Letter - Daniel Frumkin

- GL 2006-03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations" Issued April 10, 2006
- All licensees responded in accordance with the information request
- 16 licensees reported Hemyc or MT
  - 10 licensees resolving Hemyc or MT issues through NFPA 805
  - 1 licensee removed Hemyc
  - 3 licensees requested exemptions
  - 2 licensees use as radiant energy shields

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10

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
545th	9	9.1

Title: PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES

Authors:
Sam Duraiswamy

List of Documents Attached	
	9

Instructions to Preparer	From Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	Sam Duraiswamy

CERTIFIED: **ISSUED:**

09/06/2007 3:56 PM
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INTERNAL USE ONLY

SUMMARY/MINUTES OF THE ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING September 5, 2007

The ACRS Subcommittee on Planning and Procedures held a meeting on September 5, 2007, in Room T-2B1, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 8:35 am and adjourned at 9:45 am.

ATTENDEES
W. Shack
M. Bonaca
S. Abdel-Khalik

ACRS STAFF
F. Gillespie
S. Duraiswamy
H. Nourbakhsh
G. Hammer
D. Fischer
J. Gallo
C. Santos
M. Afshar-Tous
Z. Abdullahi
M. Banerjee
G. Shukla
D. Bessette

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

Member assignments and priorities for ACRS reports and letters for the September ACRS meeting are attached (pp. 6). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the September ACRS meeting be as shown in the attachment.

2) <u>Anticipated Workload for ACRS Members</u>

The anticipated workload for ACRS Members through November 2007 is attached (pp. 7-8). The objectives are to:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action (pp. 9-10).

<u>RECOMMENDATION</u>

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

3) <u>*Federal Register Notice* and Press Release to Solicit Qualified Candidates for Membership on the ACRS</u>

Federal Register Notice and press release have been issued on August 1, 2007 (pp. 11-12) soliciting candidates with expertise in the areas of Digital I&C, Plant Operations, or Materials Engineering. Interested persons can submit resumes until November 30, 2007.

<u>RECOMMENDATION</u>

The Subcommittee recommends that the ACRS Executive Director keep the Committee informed periodically of the applications received.

4) <u>RES Regulatory Guide Project</u>

During the June 2007 meeting, the Committee was informed of the RES staff's plan to update, as necessary, all NRC Regulatory Guides by December 2009. These updates will be performed in various phases. Phase 1 was completed in March 2007. Phase 2 Regulatory Guide updates will be completed by December 2008 and Phase 3 by December 2009. During that meeting, the Committee asked the ACRS staff to obtain a detailed schedule for ACRS review of the updated Regulatory Guides in Phases 2 and 3. Also, the Committee decided to consider all updated Reg. Guides in Division 1, "Power Reactors," and those Guides in other Divisions that have safety implications. A tentative schedule for submitting Phases 2 and 3 Reg. Guides for ACRS review is attached (pp. 13-17). The staff plans to submit updated Reg. Guides in Phase 2 starting November 2007. Member assignments for reviewing the updated Guides in Phase 2 will be provided during the October ACRS meeting.

RECOMMENDATION

The Subcommittee recommends the following:

- Mr. Maynard should take the lead and coordinate the Committee's review of the updated Reg. Guides.
- The ACRS staff should provide proposed member assignments for reviewing the updated Reg. Guides in Phase 2 for consideration by the Planning and Procedures Subcommittee and the full Committee during their October meetings.

5) ACRS Meeting With the Commission

The ACRS is tentatively scheduled to meet with the Commission on Friday, December 7, 2007. Since there are no major topics, we requested, through the Office of SECY, that the Commission consider postponing this meeting to April/May 2008. The Commission has agreed to postpone this meeting to May 9, 2008.

6) Quality Assessment of Selected NRR Products

On July 13, 2007, Mr. Wiggins, Deputy Director, NRR, met with the members of the Planning and Procedures Subcommittee and requested that the ACRS perform a review of the quality of selected NRR products. This review would be similar to that being performed by the Committee on the quality of selected RES research projects. If such a review is not feasible, Mr. Wiggins would like to know what mechanisms could be used to obtain Committee's feedback on the quality of selected NRR products.

RECOMMENDATION

The Subcommittee recommends that since the ACRS has already been providing feedback on the NRR safety evaluation reports associated with license renewal and power uprate applications and the Committee's workload is expected to be heavy, the Committee not undertake this task. The members should provide their views on this matter.

7) Cooperative Severe Accident Research Program (CSARP) and MELCOR Code Assessment Program (MCAP) Technical Review Meetings

The annual CSARP/MCAP technical review meetings, being organized by RES, is scheduled to be held on September 18-20, 2007, at the Hyatt Regency Hotel, Albuquerque, New Mexico. A preliminary agenda for this meeting is attached (pp. 18-22). This meeting serves as a forum for exchanging technical information and research findings in the area of severe accidents.

RECOMMENDATION

The Subcommittee recommends that those members who are interested in attending this meeting inform Mr. Tanny Santos. If no members are able to attend this meeting, the Committee should consider sending Dr. Kress, ACRS Consultant, to this meeting.

8) Operating Plan, Self-Assessment, and Letter Matrix

The ACRS staff is in the process of preparing the ACRS/ ACNW&M Operating Plan for 2008. This is in three parts, 2008 operations, resources, and annual self-assessment. Contained within the annual self-assessment is the traditional letter matrix. The current due date to the Commission is November 1, 2007. An early draft will be provided to the Planning and Procedures Subcommittee on September 5, 2007 for its information and comment as appropriate. A draft will be sent to all ACRS members prior to the October meeting. The information is similar to last year's plan reformatted to eliminate material wherever possible.

Preparation of the ACRS letter matrix involves summarizing the ACRS reports and letters issued in 2007. In order to preclude violation of the ACRS Bylaws, the Committee needs to authorize the ACRS Executive Director and/or his designee to summarize the Committee letters and reports.

RECOMMENDATION

The Subcommittee recommends the following:

- The ACRS staff should send the draft Operating Plan, Self-Assessment, and Letter Matrix to the members following the September full Committee meeting for review and comment.
- The Committee should authorize the ACRS Executive Director and/or his designee to summarize the ACRS letters and reports issued in 2007.

10) Quadripartite Working Group Meeting

Germany's Reaktor-Sicherheitskommission (RSK) will host the first Quadripartite Working Group (WG) meeting on the topic of "Sump Screen Blockage" on October 17-18, 2007, in Erlangen, Germany. During the April meeting, the Committee authorized Dr. Banerjee and Dr. Wallis to attend this WG meeting.

Dr. Banerjee will present a paper on "Overview of US Investigations/Analyses on Sump Screen Blockage" and Dr. Wallis will present a paper on "Impact of Downstream and Chemical Effects on Sump Screen Blockage." RSK has asked for the presenters to provide a copy of their paper and presentation by the end of September 2007.

RECOMMENDATION

The Subcommittee recommends that Dr. Banerjee and Dr. Wallis provide their papers and presentations to Mugeh Afshar-Tous by September 28, 2007. Mugeh will keep the Committee informed of the progress in planning these WG meetings.

11) <u>Proposed ACRS Meeting Dates for CY 2008</u>

Proposed ACRS meeting dates from CY 2008 are included in the attached calendar (pp. 23-34) and also summarized below.

Meeting No.	Dates
—	January 2008 (No Meeting)
549	February 7 – 9, 2008
550	March 6 - 8, 2008
551	April 3 - 5, 2008
552	May 8 – 10, 2008
553	June 4 – 6, 2008 (Wed – Fri)
554	July 9 – 11, 2008 (Wed – Fri)
—	August, (No Meeting)
555	September 4 – 6, 2008
556	October 2 – 4, 2008
557	November 6 – 8, 2008
558	December 4 – 6, 2008

RECOMMENDATION

The Subcommittee recommends that the Committee approve the meeting dates for CY 2008 either during the September meeting or during the October meeting.

ANTICIPATED WORKLOAD
SEPTEMBER 6-8, 2007

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	SUB. MTG DATES
Bonaca	–	Hammer	**Subcommittee Report** – Interim Review of the License Renewal Application for the FitzPatrick Nuclear Plant	–	–	**9/5/07**
Maynard	–	Banerjee	License Renewal Application and the Final SER for the Pilgrim Nuclear Power Station	A	To support staff schedule	**4/4/07**
		Hammer	Status of NRR Activities in the Fire Protection Area **[INFORMATION BRIEFING]**	–	–	–
Powers	Shack/ Banerjee/ Maynard	Nourbakhsh	Draft Report on the Quality Assessment of the Research Projects: Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping; Cable Response to Live Fire (CAROLFIRE) Testing; and Technical Review of On-line Monitoring Techniques for Performance Assessment	B	Report to be completed in October	–
	–	Nourbakhsh	Draft Report on the NRC Safety Research Program	B	Report to be completed in December	–
Shack	–	Bessette	Proposed RES Recommendation for Resolving GSI-156.6.1, Pipe Break Effects on Systems and Components Inside Containment	A	To support staff schedule	–
	–	Fischer	Technology-Neutral Framework for Future Plant Licensing	A	To provide Committee's views	–

ANTICIPATED WORKLOAD
OCTOBER 4-6, 2007

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	SUB. MTG DATES
Abdel-Khalik	—	Abdullahi/ Bessette	Draft Generic Letter 2007-XX, "Managing Gas Intrusion in ECCS, Decay Heat Removal, and Containment Spray Systems"	Report as needed	To support staff schedule	—
Apostolakis	—	Shukla	Interim Staff Guidance on Digital I&C	A	To provide Committee's views	**9/13/07**
Maynard	—	Banerjee	Meeting with NEI, EPRI, and INPO to Discuss Industry Activities **[INFORMATION BRIEFING]**	—	—	—
Powers	Shack/ Banerjee/ Maynard	Nourbakhsh	Draft Final Report on Quality Assessment of the NRC Research Projects on: Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping; Cable Response to Live Fire (CAROLFIRE) Testing; and Technical Review of the Online Monitoring Techniques for Performance Assessment	A	To support pre-established schedule	—
	Cognizant Members	Nourbakhsh	Draft Report on the NRC Safety Research Program	B	Report to be completed in December	—
Shack	—	Hammer	Dissimilar Metal Weld Issue	Report as needed	To provide Committee's views	—

ANTICIPATED WORKLOAD
NOVEMBER 1-3, 2007

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	SUB. MTG DATES
Banerjee	—	Abdullahi	Extended Power Uprate Application for the Susquehanna Nuclear Plant	A	To support staff schedule	**10/9-10/07**
Bonaca	—	Banerjee	License Renewal Issues Related to Exceptions to GALL Report and Use of the Audit Report	Report as needed	—	—
Corradini	—	Hammer	Selected Chapters of the SER Associated with the ESBWR Design Certification	Report as needed	—	**10/2-3/07** **10/25/07**
Powers	—	Fischer	Vogtle Early Site Permit Application and the Associated SER	A	To support staff schedule	**10/24/07**
		Fischer	Response to November 8, 2006 SRM that as Licensing Under Part 52 Continues, the Committee Should Advise the Commission on the Effectiveness and Efficiency of Staff's Implementation of Lessons Leaned in Areas it has Reviewed, for Example, the Development of Guidance Documents for Early Site Permits	A	To respond to commission SRM. Due Date 11/30/07	—
		Nourbakhsh	Draft Report on the NRC Safety Research Program	B	Report to be completed in December	—
Shack	—	Shukla	Meeting with Commissioner Lyons to Discuss Items of Mutual Interest	—	—	—
	—	Nourbakhsh/ Bessette	State-of-the Art Reactor Consequence Analysis	A	To support staff schedule	**10/26/07**

1 **Draft Final Rulemaking and Draft Final Regulatory Guides Regarding ASME Code Cases** (Open)

Member: William Shack **Engineer:** Gary Hammer

Estimated Time:

Purpose: Determine a Course of Action

Priority:

Requested by: NRR/RES L. Mark Padovan (NRR), W. Norris (RES)

The staff has revised the following Regulatory Guides (RGs) on ASME Code Cases to support a final rulemaking associated with 10 CFR 50.55a, "Codes and Standards:"

(1) RG 1.84 (DG-1133), Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Division 1
(2) RG 1.147 (DG-1134), Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1
(3) RG 1.193 (DG-1135), ASME Code Cases Not Approved for Use

A memorandum dated February 14, 2006, from John Larkins, Executive Director, ACRS to Luis Reyes, EDO, stated that the Committee had no objection to the staff's proposal to issue proposed revisions to these regulatory guides for public comment and that the Committee would prefer to review the RGs and associated rulemaking as a package. The RGs were issued for public comment in October 2006.

In an August 7, 2007 memorandum from Jennifer Uhle, Director, DFERR, RES, to Frank Gillespie, Executive Director, ACRS/ ACNW&M, the staff provided draft final versions of RG 1.84 and RG 1.145 as well as staff response to public comments. The staff requested that the ACRS determine whether they wish to review these Regulatory Guides.

In an August 6, 2007 memorandum from James Wiggins, Deputy Director, NRR, to Frank Gillespie, Executive Director, ACRS/ ACNW&M, the staff provided the final rulemaking package for 10 CFR 50.55a that incorporates by reference the latest revisions to RG 1.84 and RG 1.147. The staff requested feedback regarding ACRS review of this rulemaking.

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

ACRS Items Requiring Committee Action

2. **Generic Letter 2007-XX Managing Gas Intrusion In ECCS, Decay Heat Removal And Containment Spray Systems**

Member: Said Abdel-Khalik **Engineer:** David Bassette

Estimate Time:

Purpose: Determine a Course of Action

Priority:

Requested by: NRR David Beaulieu

NRC is proposing to issue a generic letter on the impact of gas "intrusion" in the piping systems and components relied upon to perform the safety functions. Operating history data show that gas intrusion has affected the operability of safety systems at nuclear plants.

The generic letter was already issued for public comments. The CRGR meeting is scheduled for early September. The GL is intended to be issued on October 30, 2007.

The P&P requests that Dr. Abdel-Khalik propose a course of action on this matter.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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

No. 07-097 August 1, 2007

NRC INVITES PUBLIC TO SUBMIT NOMINATIONS FOR THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Printable Version

The U.S. Nuclear Regulatory Commission (NRC) is seeking qualified candidates for appointment to its Advisory Committee on Reactor Safeguards (ACRS).

The ACRS is a part-time advisory group which is statutorily mandated by the Atomic Energy Act of 1954, as amended. ACRS provides independent technical review of, and advice on, matters related to the safety of existing and proposed nuclear facilities and on the adequacy of proposed reactor safety standards. Of primary importance are the safety issues associated with the operation of 104 commercial nuclear power plants in the United States, and regulatory initiatives including risk-informed and performance-based regulations, license renewal, power uprates, and the use of mixed oxide and high burnup fuels. An increased emphasis is being given to safety issues associated with new reactor designs and technologies including passive system reliability and thermal hydraulic phenomena, use of digital instrumentation and control, international codes and standards for use in multinational design certifications, material and structural engineering, and nuclear analysis and reactor core performance.

Currently, the Commission is seeking individuals with technical expertise in one or more of the areas of materials engineering, digital instrumentation and control, or plant operations.

The ACRS membership includes individuals who possess specific technical expertise along with a broad perspective in addressing safety concerns. Committee members are selected from a variety of engineering and scientific disciplines, such as risk assessment, chemistry, mechanical engineering, civil engineering, materials sciences, and earth sciences. At this time, candidates are being sought who have 10 years of experience in one or more of the areas of materials engineering, digital instrumentation and control, or plant operations. Candidates with pertinent graduate level education will be given additional consideration. Committee members serve a four-year term with the possibility of two reappointments for a total service of 12 years. The Commission hopes to fill three vacancies as a result of this request.

Criteria used to evaluate candidates includes education and experience, demonstrated skills in nuclear reactor safety matters, the ability to solve complex technical problems, and the ability to work collegially on a board, panel, or committee. The Commission, in selecting its committee members, considers the need for a specific expertise to accomplish the work expected to be before the ACRS. For these positions, the expertise must be preferably related to one or more of the areas of materials engineering, digital instrumentation and control, or plant operations. Consistent with the requirements of the Federal Advisory Committee Act, the Commission seeks candidates with varying views and of diverse backgrounds so that the membership on the committee will be fairly balanced in terms of the points of view represented and functions to be performed by the Committee. Candidates will undergo a thorough security background check to obtain the security clearance that is mandatory for all ACRS members. The security background check will involve the completion and submission of paperwork to NRC. Candidates for ACRS appointments may be involved in or have financial interests related to NRC-regulated aspects of the nuclear industry. Because conflict-of-interest considerations may restrict the participation of a candidate in ACRS activities, the degree and nature of such restriction on an individual's activities as a member will be considered in the selection process. Each qualified candidate's financial interests must be reconciled with applicable Federal and NRC rules and regulations prior to final appointment. This might require divestiture of securities or discontinuance of certain contracts or grants. Information regarding these restrictions will be provided upon request.

A résumé describing the educational and professional background of the candidate including any special accomplishments, publications, and professional references, should be provided. Candidates should also provide their current address and telephone number, and email address. All candidates will receive careful consideration. Appointment will be made without regard to factors such as race, color, religion, national origin, sex, age or disabilities. Candidates must be citizens of the United States and be able to devote approximately 100-130 days per year to committee business.

Résumés will be accepted until November 30, 2007. Résumés should be sent to Angelina Chapeton, ACRS/ACNW, Mail Stop T2E-26, US Nuclear Regulatory Commission, Washington, DC 20555-0001 or emailed to ahc@nrc.gov.

From: Charles Hammer
To: Sam Duraiswamy
Date: 08/22/2007 7:55:36 AM
Subject: Fwd: Regulatory Guides

Sam, Tanny asked me to forward this to you for a P&P item. Thanks, Gary

>>> John Ridgely 08/21/2007 3:50 PM >>>
Good Afternoon!

Attached for your information and use in planning are two files. One file contains the current list of Phase 2 Regulatory Guides and the expected date of having a draft guide available. The earliest that it could be provided to you for Committee review is a couple of weeks after this date.

The second file shows how many are expected to become available by month. This shows that there is expected to be a significant number becoming available as this year closes.

It is our intention to provide these guides to you at the earliest possible time for Committee review to determine if the Committee desires to waive review of the guide. We intent to follow the same procedure as for the Phase 1 guides, namely to provide them by e-mail to be followed up by memorandum.

This information is transitory and represents the best information at this time. However, the Committees should be aware that the specific guides in Phase 2 and related schedules are expected to continually change. We will endeavor to keep you apprized of all of the latest information as it becomes available.

Have a great day!

Thank you,

John N. Ridgely

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      _/_/    _/ _/    _/ _/      _/
     _/  _/  _/ _/    _/ _/          John N. Ridgely
    _/   _/ _/ _/   _/  _/         E-mail: JNR@NRC.gov
   _/    _/_/ _/_/_/   _/       Telephone: (301)415-6555
  _/     _/ _/   _/   _/           FAX: (301)415-5062
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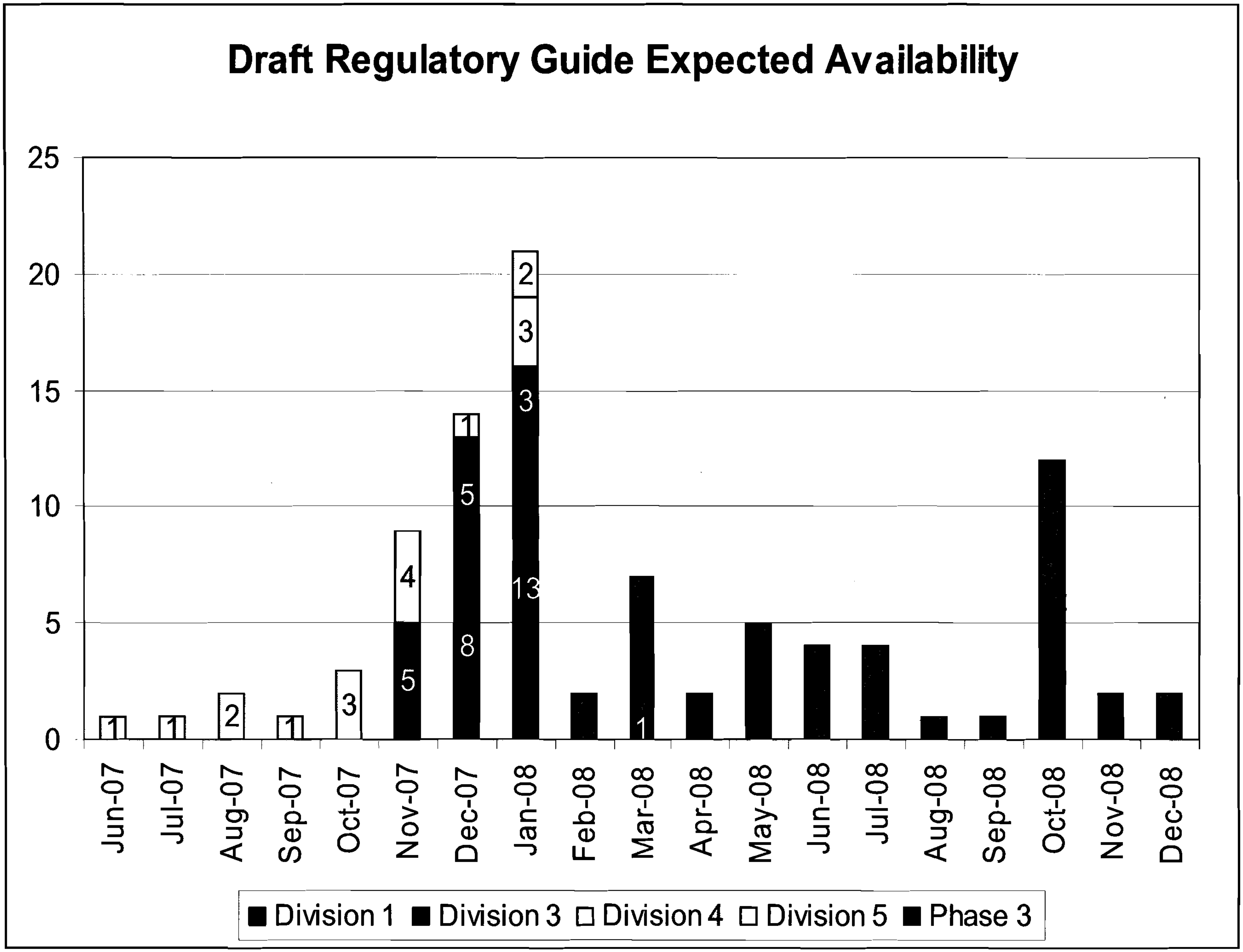
RG No	DG No	RG Title	Estimated Draft Date
1.6		Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	12/31/2007
1.8		Qualification and Training of Personnel for Nuclear Power Plants	01/31/2008
1.27		Ultimate Heat Sink for Nuclear Power Plants	01/31/2008
1.40		Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants	12/31/2007
1.45	1173	Reactor Coolant Pressure Boundary Leakage Detection Systems	
1.47		Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	12/31/2007
1.62		Manual Initiation of Protective Actions	12/31/2007
1.84	1133	Design and Fabrication and Materials Code Case Acceptability, ASME Section III	
1.89		Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	03/31/2008
	1136	Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire	01/31/2008
	1138	(Proposed Appendix C to Regulatory Guide 1.200) NRC Staff Regulatory Position on ANS External Hazards PRA Standard	
	1148	Qualification of Safety-Related Battery Chargers and Inverters	
	1149	Qualification of Safety-Related Motor Control Centers	11/30/2007
1.93		Availability of Electric Power Sources	12/31/2007
1.100		Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants	11/30/2007
1.101	1174	Emergency Planning and Preparedness for Nuclear Power Plants	01/31/2008
1.105	1141	Setpoints for Safety-Related Instrumentation	11/31/07
1.114		Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit	01/31/2008
1.127		Inspection of Water-Control Structures Associated with Nuclear Power Plants	01/31/2008
1.131	1132	Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants	
1.135		Normal Water Level and Discharge at Nuclear Power Plants	01/31/2008
1.137		Fuel-Oil Systems for Standby Diesel Generators	01/31/2008
1.147	1134	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1	

1.149		Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations	01/31/2008
1.151		Instrument Sensing Lines	11/30/2007
1.153		Criteria for Safety Systems	12/31/2007
1.155		Station Blackout	
1.183		Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	01/31/2008
1.193	1135	ASME Code Cases Not Approved for Use	
	New-003	Performing Seismic Margin Analysis for Nuclear Power Plants	01/31/2008
	New-004	Containment Performance and Fragility Assessments for Nuclear Power Plants	01/31/2008
	New-009	Proposed RG on performance of non-Appendix VIII UT examination"	11/30/2007
	New-016	Sizing Large Lead Storage Batteries	12/31/2007
	New-017	Installation of Valve Regulated Batteries	12/31/2007
	New-018	Maintenance and Testing of Valve Regulated Batteries	01/31/2008
3.5	3024	Standard Format and Content of License Applications for Uranium Mills	12/31/2007
3.8	3025	Preparation of Environmental Reports for Uranium Mills	12/31/2007
3.11		Design, Construction, and Inspection of Embankment Retention Systems for Uranium Mills	12/31/2007
3.11.1		Operational Inspection and Surveillance of Embankment Retention Systems for Uranium Mill Tailings	12/31/2007
3.12		General Design Guide for Ventilation Systems of Plutonium Processing and Fuel Fabrication Plants	01/31/2008
3.13		Guide for Acceptable Waste Storage Methods at UF6 Production Plants	01/31/2008
3.42		Emergency Planning for Fuel Cycle Facilities and Plants Licensed Under 10 CFR Parts 50 and 70	01/31/2008
3.46	3026	Standard Format and Content of License Applications, Including Environmental Reports, for In Situ Uranium Solution Mining	12/31/2007
3.67		Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities	
4.11		Terrestrial Environmental Studies for Nuclear Power Stations	12/31/2007
4.13		Performance, Testing, and Procedural Specifications for Thermoluminescence Dosimetry: Environmental Applications	01/31/2008
4.15	4010	Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment	

4.16		Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants	01/31/2008
4.20		Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors	01/31/2008
	4012	Minimization of Contamination and Radioactive Waste Generation in Support of Decommissioning	
5.12		General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials	11/30/2007
5.27		Special Nuclear Material Doorway Monitors	01/31/2008
5.54	5016	Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants	09/30/2007
5.62	5019	Reporting of Safeguards Events	06/28/2007
5.65		Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls	11/30/2007
5.66		Access Authorization Program for Nuclear Power Plants	11/30/2007
5.68		Protection Against Malevolent Use of Vehicles at Nuclear Power Plants	08/31/2007
5.69		Guidance for the Application of the Radiological Sabotage Design-Basis Threat in the Desing, Development, and Implementation of a Physical Security Program that Meets 10 CFR 73.55 Requirements	
5.7		Guidance for the Application of the Theft & Diversion of Category I Special Nuclear Material in the Design, Development, and Implementation of a Physical Security Program that Meets 10 CFR 73.45 and 73.46 Requirements	
	5014	Physical Security	10/31/2007
	5015	Training and Qualification	10/30/2007
	5020	Applying for Enhanced Authority and Accomplishing Firearms Background Check	
	5021	Managing Safety / Security Interface	07/05/2007
	5022	Cyber Security Programs for Nuclear Power Reactors	11/30/2007
	New-020	Fatigue Management for Nuclear Power Plants	08/15/2007
	New-022	Construction Fitness for Duty	01/31/2008
10.3		Guide for the Preparation of Applications for Special Nuclear Material Licenses of Less than Critical Mass Quantities	
	New-021	Assessment of Beyond-Design-Basis Aircraft Impacts	10/01/2007



Draft Regulatory Guide Expected Availability
25
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Nov-08
Dec-08
Division 1
Division 3
Division 4
Division 5
Phase 3

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

August 16, 2007

MEMORANDUM TO: Jennifer Uhle, Director, RES/DFERR
James Lyons, Director, NRO/DSER
Charles Ader, Director, NRO/DSRA
William Ruland, Director, NRR/ADES/DSS
Mark Cunningham, Director, NRR/ADES/DRA
Frank Gillespie, Executive Director, ACRS/ACNW
William Shack, Chairman, ACRS

RECEIVED

AUG 20 2007

FROM: Farouk Eltawila, Director
Division of Risk Assessment and Special Project
Office of Nuclear Regulatory Research

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

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 18 through 20, 2007, at the Hyatt Regency Hotel, Albuquerque, New Mexico. A preliminary agenda for the meeting is enclosed. As you may know, the Office of Nuclear Regulatory Research organizes the CSARP/MCAP 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 countries who are CSARP members and by delegates from the national laboratories, academia, and other organizations who are engaged in severe accident research.

The preliminary agenda has been distributed to cognizant staff involved in severe accident and source term research in your office. The agenda for CSARP/MCAP can be found at the Sandia National Laboratory website: http://melcor.sandia.gov/. Dr. Hossein Esmaili of my staff is the coordinator of the CSARP/MCAP meeting. Please contact Dr. Esmaili at 301-415-6084 or e-mail him at hxe1@nrc.gov if you have any questions about the meeting or would like to notify us of your interest in attending the meeting. As always, we welcome you and your staff participation in CSARP/MCAP meeting.

Enclosure:
As stated

cc w/encl.:
B. Sheron, RES
M. Johnson, RES
A. Drozd, NRO
J. Lee, NRO
M. Hart, NRR
M. Blumberg, NRR
H. Nourbakhsh, ACRS

Tuesday, September 18, 2007

8:00 am	Registration	
8:30 am	Opening Remarks/Overview of the NRC Severe Accident Research Program	NRC

Technical Session 1 – Severe Accident Research 1
Co-Chairs: S. Basu, NRC, and W. Tromm, FZK

9:00 am	SERENA-2 Preliminary Work Program	D. Magallon, CEA
9:30 am	KAERI Activities for Resolution of Ex-Vessel Steam Explosion Risk	S.W. Hong et al., KAERI
10:00 am	BREAK	
10:30 am	Research at the Sweden's KTH on Debris Coolability, Steam Explosion and Plant Applications	Truc-Nam Dinh, KTH
11:00 am	QUENCH Program: Recent Results and Future Activities	M. Steinbrueck, FZK
11:30 am	In-Vessel Melt Behavior Experiments at FZK	A. Miassoedov, FZK
12:00 pm	LUNCH	

Technical Session 2 – Severe Accident Research 2
Co-Chairs: R. Lee, NRC, and J.P. Van Dorsselaere, IRSN

1:00 pm	International Source Term Program: Status of Oxidation Experimentation on Boron Carbide Rods and Zircaloy Cladding	Ch. Dominguez, IRSN
1:30 pm	Material Behavior in the Containment During the Phebus-FPT2 Test with a Focus on Iodine Behavior	Ph. March, IRSN
2:00 pm	The European Decision Support System RODOS for Off-site Emergency Management	W. Tromm, FZK
2:30 pm	Validation of the Belgian Severe Accident Guidelines	J. Verpoorten, Tractebel

3:00 pm	BREAK	
3:30 pm	ARTIST International Consortium Project: Facilities and Preliminary Results	Detlef Suckow, PSI
4:00 pm	Progress with Understanding of Zircaloy/air Oxidation: Status	Y. Liao, PSI
4:30 pm	DBA Source Term Analysis	J. Lee, NRC
5:00 pm	Status of SOARCA Project	S. Burns, SNL
5:30 pm	Natural Circulation of BWR/PWR Fuel Assemblies	K.C. Wagner, SNL
6:00 pm	ADJOURN	

Wednesday, September 19, 2007

Technical Session 3 –Severe Accident Codes: Development and Assessment
Co-Chairs: A. Drozd, NRC, and M. Sonnenkalb, GRS

8:00 am	Status of MELCOR 2.1	I. Khalil, SNL
8:30 am	Towards a Wider Scope of ASTEC Application Beyond Severe Accidents in PWR and VVER	J.P. Van Dorsselaere, IRSN
9:00 am	Experience from MELCOR Application for German LWR and Future Code Development Suggestions	Martin Sonnenkalb, GRS
9:30 am	CFD Severe Accident Analysis and MELCOR Applications at JNES	M. Ogino, JNES
10:00 am	BREAK	
10:30 am	Simulation of the FP Vapor Plate-out in OGL-1 Experiment using MELCOR	Jong-hwa Park, KAERI
11:00 am	Partners' Meeting (open to partners only)	
12:00 pm	LUNCH	

Technical Session 4 – High Temperature Gas Reactor Safety Analysis

Co-Chairs: H. Esmaili, NRC, and W. Hering, FZK

1:00 pm	Overview of NGNP PIRT Activities	S. Basu, NRC
1:30 pm	Status of NGNP Fission Product PIRT	R. Lee, NRC
2:00 pm	HTR Test Loop and Modeling Activities	W. Herring, FZK
2:30 pm	BREAK	
3:00 pm	Strategy to Simulate the H3 Distribution, Plate-out and Graphite Dust in HTGR System using MELCOR	Jong-hwa Park, KAERI
3:30 pm	MELCOR Modeling of a Pebble Bed Modular Reactor	K. Vierow et al., TAMU
4:00 pm	MELCOR Development for HTGR Application	R. Gauntt, SNL
4:30 pm	ADJOURN	

MELCOR Cooperative Assessment Program (MCAP) Meeting

Thursday, September 20, 2007

8:00 am	Registration	
8:30 am	Introduction and Opening Remarks	NRC / SNL
	Technical Session 1 – Recent Applications of MELCOR Co-Chairs: Chang-Wook, KINS, and R. Gauntt, SNL	
9:00 am	Evaluation of Kori-1 Severe Accident Management Guideline using MELCOR FCL Model	Chang-Wook et al., KINS
9:30 am	MELCOR Assessment against PUMA Bottom Drain Line Break and GDCS Line Break Integral Tests	K. Vierow et al., TAMU
10:00 am	BREAK	
10:30 am	Demonstration on the MELMACCS Utility	N. Bixler, SNL
11:00 am	Regulatory Application of MELCOR	R. Gauntt, SNL
12:00 pm	LUNCH	
1:00 pm	MELCOR application in Yucca Mountain Cask assessment	D. Kalinich, SNL
1:30 pm	Status of Decision Making Tool for NRC Operations Center	H. Esmaili, NRC
	Technical Session 2 – MELCOR Code Assessment Co-Chairs: J. Duspiva, NRI, and L. Humphries, SNL	
2:00 pm	MELCOR Analysis of the Paks Event	KC Wagner, SNL
2:30 pm	Simulation of Quench-11 Test with MELCOR 1.8.6 Code	J. Duspiva, NRI
3:00 pm	BREAK	
3:30 pm	MELCOR TMI-2 Assessment	L. Humphries, SNL
4:00 pm	MELCOR 2.1 Assessment Studies	V. Strizhov, RAS
4:30 pm	Assessment of MELCOR point kinetics model	R. Gauntt, SNL
5:00 pm	ADJOURN	

January 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
		1 **New Year's Day**	2	3	4	5
6	7	8	9	10	11	12
13	14	15	16	17	18	19
20	21 **Martin Luther King Day**	22	23	24	25	26
27	28	29	30	31		

February 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
					1	2
3	4	5	6 **Ash Wednesday**	7 **ACRS 549th**	8 **ACRS 549th**	9 **ACRS 549th**
10	11	12 **ACNW&M 186th**	13 **ACNW&M 186th**	14 **ACNW&M 186th**	15	16
17	18	19	20	21	22	23
24	25	26	27	28	29	

March 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
						1
2	3	4	5	6 **ACRS 550th**	7 **ACRS 550th**	8 **ACRS 550th**
9	10	11	12	13	14	15
16 **Palm Sunday**	17 **St. Patrick's Day**	18 **ACNW&M 187th**	19 **ACNW&M 187th**	20 **ACNW&M 187th**	21 **Good Friday**	22
23 **Easter Sunday** 30	24 31	25	26	27	28	29

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April 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
		1	2	3 **ACRS 551st**	4 **ACRS 551st**	5 **ACRS 551st**
6	7	8	9	10	11	12
13	14	15 **ACNW&M 188th**	16 **ACNW&M 188th**	17 **ACNW&M 188th**	18	19
20 **First Day of Passover**	21	22	23	24	25	26
27 **Last Day of Passover**	28	29	30			

May 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
				1	2	3
4	5	6	7	8 **ACRS 552nd**	9 **ACRS 552nd**	10 **ACRS 552nd**
11	12	13	14	15	16	17
18	19	20 **ACNW&M 189th**	21 **ACNW&M 189th**	22 **ACNW&M 189th**	23	24
25	26 **Memorial Day**	27	28	29	30	31

June 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
1	2	3	4 **ACRS 553rd**	5 **ACRS 553rd**	6 **ACRS 553rd**	7
8 **ANS Meeting - Anaheim, CA**	9 **ANS Meeting - Anaheim, CA**	10 **ANS Meeting - Anaheim, CA**	11 **ANS Meeting - Anaheim, CA**	12 **ANS Meeting - Anaheim, CA**	13	14
15	16	17 **ACNW&M 190th**	18 **ACNW&M 190th**	19 **ACNW&M 190th**	20	21
22	23	24	25	26	27	28
29	30					

July 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
		1	2	3	4 **Independence Day**	5
6	7	8	9 **ACRS 554th**	10 **ACRS 554th**	11 **ACRS 554th**	12
13	14	15	16	17	18	19
20	21	22 **ACNW&M 191st**	23 **ACNW&M 191st**	24 **ACNW&M 191st**	25	26
27	28	29	30	31		

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August 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
					1	2
3	4	5	6	7	8	9
10	11	12	13	14	15	16
17	18	19	20	21	22	23
24 31	25	26	27	28	29	30

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September 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
	1 **Labor Day**	2 **Ramadan Begins**	3	4 **ACRS 555th**	5 **ACRS 555th**	6 **ACRS 555th**
7	8	9	10	11	12	13
14	15	16 **ACNW&M 192nd**	17 **ACNW&M 192nd**	18 **ACNW&M 192nd**	19	20
21	22	23	24	25	26	27
28	29	30 **Rosh Hashana**				

October 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
			1	2 **ACRS 556th**	3 **ACRS 556th**	4 **ACRS 556th**
5	6	7	8	9 **Yom Kippur**	10	11
12	13 **Columbus Day**	14	15	16	17	18
19	20	21 **ACNW&M 193rd**	22 **ACNW&M 193rd**	23 **ACNW&M 193rd**	24	25
26	27	28	29	30	31	

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November 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
						1
2	3	4	5	6 **ACRS 557th**	7 **ACRS 557th**	8 **ACRS 557th**
9 **ANS Meeting – Reno, NV**	10 **ANS Meeting – Reno, NV**	11 **ANS Meeting – Reno, NV** **Veterans Day**	12 **ANS Meeting – Reno, NV**	13 **ANS Meeting – Reno, NV**	14	15
16	17	18 **ACNW&M 194th**	19 **ACNW&M 194th**	20 **ACNW&M 194th**	21	22
23 30	24	25	26	27 **Thanksgiving Day**	28	29

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December 2008

Sunday	Monday	Tuesday	Wednesday	Thursday	Friday	Saturday
	1	2	3	4 **ACRS 558th**	5 **ACRS 558th**	6 **ACRS 558th**
7	8	9 **ACNW&M 195th**	10 **ACNW&M 195th**	11 **ACNW&M 195th**	12	13
14	15	16	17	18	19	20
21	22 **First Day of Chanukah**	23	24	25 **Christmas Day**	26 **Kwanzaa**	27
28	29	30	31			