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

June 30, 2006

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

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 533rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, May 31—June 1, 2006, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 533rd meeting, May 31–June 1, 2006, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following letters and memoranda:

LETTERS:

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

- Draft Final Generic Letter 2006-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems," dated June 15, 2006
- Draft Final Generic Letter 2006-XX: Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations, dated June 16, 2006

MEMORANDA:

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

- Results of the Staff's Initial Screening of Generic Issue-197, "Iodine Spiking Phenomena," dated June 21, 2006
- Draft Final Revision 1 to Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," dated June 6, 2006

HIGHLIGHTS OF KEY ISSUES

1. Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations"

The Committee met with representatives of the NRC staff, Nuclear Energy Institute (NEI), Duke Energy, and Progress Energy to discuss the draft final Generic Letter (GL) 2006-XX, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations." The staff provided presentations on the potential of spurious actuations due to fires, as well as a summary and objective of the GL. The GL requests that each licensee submit within 90 days, a description of the plant's licensing basis with respect to the regulatory requirement for protecting redundant safe shutdown trains from multiple simultaneous spurious actuations and maintaining one train free of fire damage including a conclusion regarding the compliance of the plant. If not in compliance, the licensees should submit a functionality assessment of systems, structures, and components (SSCs) that affect the ability to achieve and maintain safe shutdown and a description of compensatory measures put in place. Additionally within 6 months, they should submit a plan to return all affected SSCs to compliance with regulatory requirements. By complying with the GL, all risk-significant circuit situations will be identified and addressed. A representative of NEI stated that issuing a GL was not the best approach to address post-fire safe-shutdown circuit analysis spurious actuations. Representatives from Duke Power and Progress Energy stated that it was not reasonable to perform the requested analysis of multiple spurious actuations within the 90 days specified in the GL. The Committee agreed with the staff's objective to bring licensees into compliance with regulatory requirements expeditiously; however, the Committee recognized the magnitude of the effort required to provide the information requested within the GL. The staff agreed to clarify the scope of information to be provided at each milestone in the schedule and to provide additional time for the functionality assessment of affected SSCs.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated June 16, 2006, recommending that the GL be issued after the scope of the requested information is clarified and the submittal dates are made more realistic.

2. Draft Final Generic Letter 2006-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems"

The Committee met with representatives of the NRC staff and NEI to discuss the draft final Generic Letter 2006-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems." The staff described a concern that the failure of inaccessible power cables for safety-related equipment due to moisture exposure could disable multiple mitigation systems. The GL requests that licensees provide the following information: a description of failures of inaccessible or underground power cables that are within the scope of the maintenance rule; a description of all inspection, testing, and monitoring programs for these power cables; and if such a program is not in place, an explanation why it is not necessary. Comments from the public and the Committee to Review Generic Requirements resulted in minor revisions to the GL. After reviewing the responses, the staff will determine what, if any, regulatory action should be taken. A representative from NEI commented that it is not clear what concern is being addressed by this GL and there is no cable monitoring technique that is available to industry that would be effective for all cases.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated June 15, 2006, recommending that Generic Letter 2006-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems," be issued.

3. Interim Staff Guidance on Aging Management Program for Inaccessible Areas of Boiling Water Reactor (BWR) Mark I Containment Drywell Shell

The Committee met with representatives of the NRC staff and NEI to discuss the proposed license renewal Interim Staff Guidance (ISG) on a plant-specific aging management program for inaccessible areas of BWR Mark I containment drywell shells. The staff described a concern that water seepage could cause corrosion of the drywell shell in inaccessible areas. The ISG recommends a plant-specific aging management program be implemented to address this aging effect. As part of this program, applicants should do the following: develop a corrosion rate for the drywell shell; demonstrate that moisture accumulation does not exist in the exterior portion of the drywell; and identify actions that will be taken if moisture is detected in inaccessible areas. A representative from NEI commented that the staff's concern should be addressed on a plant-specific basis. The licensees had addressed this issue in response to Generic Letter 87-05, "Request for Additional Information Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells." NEI plans to submit written comments to this proposed ISG.

Committee Action

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

4. Overview of New Reactor Licensing Activities

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the staff's activities associated with new reactor design certification (DC); early site permit (ESP); and combined license (COL) applications. The staff provided the Committee with a forecast of new reactor licensing activities and projected review schedules from 2005 through 2012. To the extent practical, they identified which COL applicants would be using which new reactor design (e.g., AP1000, ESBWR, EPR). The staff highlighted the COL safety review process which includes the development of draft safety evaluation reports (SERs) with open items; supplemental SERs; ACRS interactions and review; COL issuance with inspections, tests, analyses, and acceptance criteria (ITAAC); and NRC verification of ITAAC completion. The staff discussed its design-centered approach for reviewing the anticipated DCs and COL applications. These reviews may need to be done in parallel. The design-centered approach uses, to the maximum extent practical, a "one issue, one review, one position" strategy in order to optimize the review effort, the resources needed to perform these reviews, and the review schedules. To clarify, the staff will conduct one technical review for each reactor design issue and use this one decision to support the decision on a DC and on multiple COL applications. In order for the design-centered review approach to be fully effective, it is important that the DC and COL applicants achieve a consistent level of standardization among related COLs. This approach clearly can not be used for site-specific design, construction, or operational issues. The staff described and provided the Committee with the status of the development of the draft COL regulatory guide (DG-1145). DG-1145 will provide application content and process guidance for COL applications submitted under 10 CFR 52. The staff also described its ongoing and coordinated efforts to update various regulatory guides and standard review plan sections in anticipation of COL applications. Finally, the staff briefed the Committee on its construction inspection program development efforts and highlighted some of the anticipated inspection resource needs.

Committee Action

This was an information briefing. No Committee action was necessary. The Committee will use the information obtained during the briefing to formulate its plans for the efficient and effective review of DC, ESP, and COL applications.

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 May 30, 2006 meeting with the NRC staff and the Nuclear Management Company (NMC) to review the license renewal application for the Monticello Nuclear Generating Plant and the associated draft SER. The current operating license expires on September 8, 2010. During the meeting, NMC described the plant design, operating history, the license renewal review methodology, and its commitment tracking system. The staff's draft SER was issued on April 26, 2006 and contains no open or confirmatory items.

6. Quality Assessment of Selected NRC Research Projects

The Committee discussed the status of the quality assessment of the research projects on Containment Capacity Studies and on Molten Core Coolant Interaction, that were selected for FY 2006. The Committee agreed that the panel review of the research project on Containment Capacity Studies should be focused on draft NUREG/CR report entitled, "Containment Integrity Research at Sandia National Laboratory." This report summarizes the work that has been performed over the past thirty years to improve the understanding of the response of containment structures and their capacity to withstand accidents beyond design basis loads, and identifies common themes that have emerged.

Committee Action

The Committee plans to discuss the draft report on quality assessment of these two projects during its September 7- 9, 2006 meeting.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of May 18, 2006, to comments and recommendations included in the April 20, 2006 ACRS letter concerning the Draft Final Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of April 27, 2006, to comments and recommendations included in the March 23, 2006 ACRS report on the Safety Aspects of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of May 18, 2006, to comments and recommendations included in the April 14, 2006 ACRS letter on the Review of the 1994 Addenda to the ASME Code for Class 1, 2, and 3 Piping Systems and the Resolution of the Differences between the NRC staff and ASME. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of May 2, 2006, to comments and recommendations included in the March 24, 2006 ACRS report on the Final Review of the Exelon Generating Company, LLC, Application for Early Site Permit and the Associated NRC Staff's Final Safety Evaluation Report. The Committee decided that it was satisfied with the EDO's response.

  The staff plans to discuss proposed revisions to Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," with the ACRS during a future meeting.

- The Committee considered the EDO's response of May 18, 2006, to comments and recommendations included in the April 14, 2006 ACRS letter on the Grand Gulf Early Site Permit Application: Evaluation of Transportation Accidents on the Mississippi River. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of May 2, 2006, to comments and recommendations included in the April 10, 2006 ACRS report on Generic Safety Issue-191 - Assessment of Debris Accumulation on PWR Sump Performance. The Committee decided that it was not satisfied with the EDO's response, and it will consider a further response following a staff's presentation to the Thermal-Hydraulic Phenomena Subcommittee on June 13-14, 2006.

  The staff plans to develop integrated plans to acquire sufficient technical bases to evaluate the proposed PWR sump modifications. This will include continuing to participate in industry efforts to address sump performance issues as well as incorporating the information obtained into the staff's issue resolution strategy. The staff plans to review the approaches used by each of the five vendors

selected by licensees to support them in addressing GSI-191. The staff plans to develop and update guidance needed in some areas such as chemical effects and water management strategies. The ACRS Subcommittee on Thermal-Hydraulic Phenomena and/or the full Committee plans to discuss the above activities as progress has been made by the staff.

- The Committee considered the EDO's response of March 30, 2006, to the February 15, 2006 memorandum which forwarded an anonymous letter concerning the TRACE code that was received by Dr. Wallis and Dr. Ransom. The Committee decided that it was satisfied with the EDO's response.

  The staff has committed to discuss these comments in the context of a meeting with the Thermal-Hydraulic Phenomena Subcommittee later in the year regarding the status of the TRACE code.

- The Committee considered the EDO's response of May 15, 2006, to comments and recommendations included in the April 19, 2006 ACRS letter on Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs." The Committee decided that it was not satisfied with the EDO's response, but it will arrange for a future meeting to discuss the issue again with the staff.

  The staff committed to meet with the Committee to further discuss the respective points of view and reach a common understanding of this issue.

- The Committee considered the EDO's response of May 22, 2006, to the April 21, 2006 ACRS letter on the Application of the TRACG Computer Code to Evaluate the Stability of the Economic Simplified Boiling Water Reactor (ESBWR). The Committee decided that it was satisfied with the EDO's response.

  The staff plans to discuss with the ACRS the results of the application of this code during ACRS review of the ESBWR design certification.

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

During the period from May 4, 2006 through May 30, 2006, the following Subcommittee meetings were held:

- <u>Planning and Procedures</u> — May 30, 2006

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

- <u>Plant License Renewal</u> — May 30, 2006

The Subcommittee reviewed the license renewal application for the Monticello Nuclear Generating Plant and the associated draft Safety Evaluation Report prepared by the NRC staff.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The staff agreed to clarify the scope of information to be provided in response to GL 2006-XX, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations," and to provide additional time for the functionality assessment of affected SSCs.

- The Committee plans to review the final Safety Evaluation Report related to the license renewal of the Monticello Nuclear Generating Plant during a future meeting.

- The Committee plans to discuss the draft report on quality assessment of the research projects on Containment Capacity Studies and Molten Core Coolant Interaction during its September 7-9, 2006 meeting.

PROPOSED SCHEDULE FOR THE 534th ACRS MEETING

The Committee agreed to consider the following topics during the 534th ACRS meeting, to be held on July 12-14, 2006:

- Final Review of the License Renewal Application for the Nine Mile Point Nuclear Station
- Results of the Study to Determine the Need for Establishing Limits for Phosphate Ion Concentration
- Integrating Risk and Safety Margins
- Safeguards and Security Matters

Sincerely

Graham B. Wallis

Graham B. Wallis
Chairman

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

July 14, 2006

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

FROM: Graham B. Wallis
ACRS Chairman

Graham B. Wallis

SUBJECT: CERTIFIED MINUTES OF THE 533rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), MAY 31 - JUNE 1, 2006

I certify that based on my review of the minutes from the 533rd 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.

CERTIFIED

Date Issued: 07/06/2006
Date Certified: 07/14/2006

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MAY 31 — JUNE 1, 2006

LETTERS:

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

- Draft Final Generic Letter 2006-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems," dated June 15, 2006

- Draft Final Generic Letter 2006-XX: Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations, dated June 16, 2006

MEMORANDA:

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

- Results of the Staff's Initial Screening of Generic Issue-197, "Iodine Spiking Phenomena," dated June 21, 2006

- Draft Final Revision 1 to Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," dated June 6, 2006

APPENDICES

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

CERTIFIED

MINUTES OF THE 533rd MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
May 31 - June 1, 2006
ROCKVILLE, MARYLAND

The 533rd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on May 31-June 1, 2006. Notice of this meeting was published in the *Federal Register* on May 15, 2006 (65 FR 28055) (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. Graham B. Wallis (Chairman), Dr. William J. Shack (Vice Chairman), Mr. John D. Sieber, (Member-at-Large), Dr. George E. Apostolakis, Dr. J. Sam Armijo, Dr. Mario V. Bonaca, Dr. Richard S. Denning, Dr. Thomas S. Kress, Mr. Otto L. Maynard, and Dr. Dana A. Powers. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

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

Dr. Graham B. Wallis, Committee Chairman, convened the meeting at 8:30 a.m. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee.

II. Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations" (Open)

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

The Committee met with representatives of the NRC staff, Nuclear Energy Institute (NEI), Duke Energy and Progress Energy to discuss the final draft Generic Letter (GL) 2006-XX, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations." The staff provided presentations on the potential of spurious actuations due to fires, as well as a summary and objective of the draft GL. The data presented identified that concurrent hot shorts within a cable are probable and should be considered during circuit analysis. The draft GL requests that each licensee submit within 90 days, a description of the plant's licensing basis with respect to the regulatory requirement for protecting redundant safe shutdown trains from multiple simultaneous spurious actuations and maintaining one train free of fire damage including a conclusion regarding the compliance of the plant. If not in compliance, submit a functionality assessment of SSC's that affect ability to maintain and achieve safe shutdown, and submit a description of compensatory measures put in place. Additionally within 6 months, submit a plan to return all affected SSC's to compliance with regulatory requirements. By complying with the draft GL all risk-significant circuit situations will be identified and addressed.

The representative from NEI stated that issuing a GL was not the best approach to address post-fire safe-shutdown circuit analysis spurious actuations. Representatives from Duke Power and Progress Energy stated that it was not reasonable to perform the requested analysis of multiple spurious actuations within 90 days specified in the draft GL.

The Committee agreed with the staff's objective to bring the licensees into compliance with regulatory requirements expeditiously, however, the Committee recognized the magnitude of the effort required to provide the information requested within the draft GL. The staff agreed to clarify the scope of information to be provided at each milestone in the schedule and to provide additional time for the functionality assessment of affected SSC's. The Committee recommended the GL be issued following these changes.

Committee Action

The Committee issued a letter dated June 16, 2006, recommending that the draft GL issued after the scope of the requested information is clarified and the submittal dates are made more realistic.

III. Draft Final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems" (Open)

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

The Committee met with representatives of the NRC staff and NEI to discuss the draft final Generic Letter (GL) 2006-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems."

Mr. Mayfield, Office of Nuclear Reactor Regulation (NRR), stated that the staff is seeking ACRS endorsement of this GL.

Mr. Koshy, NRR, described the staff's safety concerns, the scope of the GL, the information requested in the GL, modifications to the GL as a result of comments from the public and the Committee to Review Generic Requirements (CRGR), and staff responses to the NEI 06-05 (Medium Voltage Underground Cable).

The purpose of this meeting was for the staff to obtain ACRS endorsement on the issuance of this GL. The staff is concerned that the failure of inaccessible power cables for safety-related equipment due to moisture exposure could disable multiple mitigation systems. The staff examined licensee event reports to review operating experience associated with these cable failures. Since 1989, 17 sites have experienced medium voltage cable failures and over 100 medium voltage cables have been replaced. Most of the faulty cables were not discovered until a failure occurred. The staff found that most of the failed cables were a minimum of 12 years old and the cable was submerged or exposed to moisture for sometime.

A monitoring program for these cables will increase confidence in the capability of the cable to respond to design basis events of significant duration and prevent unanticipated failures that cause plant transients. The staff provided examples of the benefits of this type of monitoring program. At Oconee, tests showed that two out of six cables were degraded and needed replacement. A plan was developed to track the degradation and replace the cables during a scheduled refueling outage. At Peach Bottom, a global replacement of 60 cables was performed within a three month period. This approach was extremely conservative. Experience at Oyster Creek shows that just replacing cables does not prevent repeat failures.

This GL is focused on inaccessible power cables that are within that scope of the maintenance rule. It requests that licensees provide the following information: a description of failures of inaccessible or underground power cables that are within the scope of the maintenance rule; a description of all inspection, testing, and monitoring programs for these power cables; and, if such a program is not in place an explanation why it is not necessary.

Comments on the proposed GL were received from four nuclear utility organizations, NEI, and a cable testing company. The staff described how some of these comments were addressed. One comment was that cable failures are random and no NRC action is required. The staff responded that based on available data, the cable failure rate is increasing. Another comment was that low voltage cables and cables included within the maintenance rule should not be the scope of the GL. The staff responded that the GL is focused on power cables that have the most significant impact to plant safety. Another comment was that surveillance tests are adequate testing for cables. The staff responded that brief cycles of operation during these surveillance tests cannot detect insulation degradation. A program to detect this type of degradation could prevent unanticipated failures while responding to design basis events. Another comment asked for the basis for considering multiple cable failures. The staff responded that an event at Davis Besse occurred in which insulation degradation caused multiple failures.

The staff stated that most of the changes to the proposed GL were editorial. These changes included revising the scope of the GL to include above ground and below ground duct banks and to remove the broadband impedance spectroscopy technique as an available testing method. CRGR review resulted in two changes to the GL. These changes were to specify that the focus of the GL is on power cables and to include an example of safety-related cable failure in the GL.

NEI 06-05 describes a graded approach for monitoring and replacing cables. The staff responded that the cables within the scope of the GL are significant because they are needed to prevent plant transients and mitigate accidents. This NEI document also recommends that plants provide dry environments for cables, be prepared for cable failures, and share resolutions of cable failures. The staff responded that providing a periodic pumping to provide a dry environment would increase cable life but not prevent failures.

Mr. Marion, NEI, stated that it is not clear what concern is being addressed by this GL and there is no cable monitoring technique that is available to industry that would be effective for all cases.

Dr. Wallis, ACRS Member, noted that it is not essential for moisture to be present to cause cable failures. Other chemicals can also cause cables to fail.

Dr. Bonaca, ACRS Member, asked about the 23 licensee event reports and two morning reports which described a small fraction of failures since not all cable failures are reported. The staff stated that the actual number of failures is difficult to project because of variations in plant locations and design features.

In response to a question from Dr. Wallis, the staff stated that after receiving the information requested in the GL, it will determine if generic or plant-specific regulatory action is needed.

A member of the public asked the staff if the information requested in this GL has already been submitted to the NRC as part of license renewal applications. The staff stated that it would verify that the information requested in the GL has not already been submitted by licensees.

Committee Action

The Committee issued a letter to the EDO dated June 15, 2006 recommending that Generic Letter 2006-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems" be issued.

IV. Interim Staff Guidance on Aging Management Program for Inaccessible Areas of Boiling Water Reactor (BWR) Mark I Containment Drywell Shell (Open)

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

The Committee met with representatives of the NRC staff and NEI to discuss proposed license renewal Interim Staff Guidance (ISG) on a plant-specific aging management program for inaccessible areas of BWR Mark I containment drywell shells.

Mr. Gillespie, NRR, provided some introductory remarks and noted that the staff is reviewing several license renewal applications for BWR plants with Mark I containments.

Ms. Tran, NRR, and Mr. Ashar, NRR, described the staff's concern addressed by this ISG, the purpose of the ISG, and the ISG recommendations.

The staff is concerned that water seepage could cause corrosion of the drywell shell in inaccessible areas. As a result, the staff has issued numerous requests for additional information during its review of license renewal applications. This ISG will provide guidance to applicants regarding the information that should be included in license renewal applications. The ISG does not impose any new technical requirements. The ISG recommends that a plant-specific aging management program be implemented to address corrosion of inaccessible areas of the drywell shell. As part of this aging management program, applicants should do the following: develop a corrosion rate for the drywell shell; demonstrate that responses to Generic Letter 87-05 (Request for Additional Information Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells) are consistent with this corrosion rate; and demonstrate that moisture accumulation does not exist in the exterior portion of the drywell. If moisture is detected in inaccessible areas, the component identified as the source of moisture should be included within the scope of license renewal and the corrosion rate should be demonstrated to be occurring at a manageable rate. An augmented inspection plan in accordance with ASME Section XI, Subsection IWE should also be described.

Mr. Marion, NEI, commented that the staff's concern is not a generic issue and should be addressed on a plant-specific basis. He added that licensees addressed this issue in response to Generic Letter 87-05. NEI plans to submit written comments to this proposed ISG. Mr. Marion also commented that the ISG process is not needed because the NRC could use its existing generic communication process.

The Committee discussed the variation in Mark I containments. Most plants have a drywell shell which is a free standing steel structure. One plant has a drywell liner that is attached to a concrete drywell. There are also variations in the location and design of the attached drain lines. Some of these lines are located near the top of the sand pocket region while others are located at the bottom of this region.

The staff's presentation describes actions to be performed if moisture is suspected in inaccessible areas. Dr. Wallis asked for clarification regarding suspected moisture. The staff stated that these actions are to be performed if moisture is detected in inaccessible areas. Dr. Wallis also asked how the corrosion rates for the drywell shells are established. The staff responded that some applicants have used a corrosion rate from a more severe location to demonstrate that the minimum wall thickness of the drywell shell will be maintained for the period of extended operation.

Drs. Shack and Kress, ACRS Members, asked why the staff did not just issue guidance to include refueling seals within the scope of license renewal. The staff stated that the refueling seals are just one possible source of water or moisture that could lead to corrosion of the drywell liner.

In response to a question from Dr. Kress, the staff stated that there are no plans for NRR to issue a User Need Memorandum to RES requesting the development of a technique to monitor the condition of the drywell shell in these areas.

Committee Action

This was for information only. No Committee action was necessary.

V. Overview of New Reactor Licensing Activities (Open)

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

The Committee received a briefing by representatives of the NRC staff regarding the staff's activities associated with new reactor design certification (DC); early site permit (ESP); and combined license (COL) applications.

Dr. Kress opened the session by telling the Committee that the staff had developed a design-centered approach to efficiently review DC, ESP, and COL applications. He said that the staff would discuss schedules and resources necessary to complete these new reactor licensing activities. He told the Committee that this briefing was for their information only and that they were not expected to write a letter on this topic at this time. He said that the briefing should help the Committee decide where it could be most useful in the process and how to best accommodate the staff's needs and schedules.

Dr. Bill Beckner, Deputy Director of the Division of New Reactor Licensing, NRR, said that the staff is not asking for the Committee to review or approve anything at this point. He said that the purpose of the briefing to establish a dialogue with the Committee and get it's feedback on the new reactor licensing review process. He explained that the anticipated workload of new reactor licensing activities would likely impact the Committee as well as the rest of the Agency. He said that the staff is not quite sure what to anticipate but said that it will be significant, and that they must be prepared for it. Dr. Beckner then introduced the presenter and other staff in the audience who were available to answer questions.

Mr. John Tappert, Chief of the Planning and Scheduling Branch in the Division of New Reactor Licensing, outlined the staff's overall presentation and then provided the Committee with an overview of the Part 52 licensing process. He said that in 2008 the staff expects to be doing multiple reviews of ESP, DC, and COL applications simultaneously. He said that they expect that all of the COL applications will reference a certified design. However, not all COL applications will reference an ESP. He provided a forecast of new reactor licensing activities (based on letters of intent which have been submitted to the Commission) and projected review schedules from 2005 through 2012. Mr. Tappert said that since the passage of the Energy Policy Act last summer, utilities have expressed significant interest in submitting COL applications. Dr. Wallis noted that many of the COL reviews would be done in parallel and asked if the subsequent reviews could be done more quickly. Mr. Tappert indicated that while the staff's expectation is that the subsequent reviews would require less resources, the duration of the subsequent COL reviews will be somewhat constrained by the time it will take to complete the reference plant COL review. Dr. Wallis suggested staggering the staff's reviews of the COL applications based on submission date. Mr. Tappert said that while the staff would prefer staggering the COL application reviews, that does not appear to be an option at this time. He also said that the staff's review schedule model does not appreciate any schedule efficiencies for having an early site permit (although he did say there would be some issues resolved already for these sites and some resource savings). To the extent practical Mr. Tappert identified which COL applicants would be using which new reactor design (e.g., AP1000, ESBWR, EPR, ABWR). When asked by Dr. Apostolakis, Mr. Tappert clarified that the COL reviews would be using the existing licensing framework, as opposed to risk-informed licensing framework. He said that these new reactor licensing activities will have a significant impact on the workload of many groups at the NRC, including those associated with recruiting, training, space allocation, Office of the General Counsel, licensing board, and the ACRS. Mr. Tappert highlighted the COL safety review process which includes the development of draft safety evaluation reports (SERs) with open items; supplemental SERs; ACRS interactions and review; COL issuance with inspection, test, and analyses acceptance criteria (ITAAC); and NRC verification of ITAAC completion. He told the Committee that the staff was going to work with a contractor to develop a schedule model that goes down to the standard review plan (SRP) section level of detail. The COL review schedule nominally has three passes through the ACRS (e.g., at the SER with open item phase, at the Supplemental SER phase, and at the final SER phase). Mr. Tappert mentioned that the staff has been working with the ACRS staff to develop an efficient ACRS review approach that will satisfy the ACRS's oversight responsibility.

Mr. Phil Ray, Acting Chief of the New Reactor Infrastructure and Guidance Development Branch in the Division of New Reactor Licensing, discussed the staff's design-centered approach for reviewing the anticipated DC and COL applications. These reviews may need to be done in parallel. The design-centered approach uses, to the maximum extent practical, a "one issue, one review, one position" strategy in order to optimize the review effort, the resources needed to perform these reviews, and the review schedules. To clarify, the staff will conduct one technical review for each reactor design issue and use this one decision to support the decision on a DC and on multiple COL applications. In order for the design-centered review approach to be fully effective, it is paramount that the DC and COL applicants maximize standardization among related COLs (approximately 70% of the issues can be standardized). This approach clearly cannot be used for site-specific design, construction, or operational issues, approximately 30% of the issues (e.g., operational programs, local meteorology, seismology, cooling water designs, ultimate heat sinks, off-site power). Mr. Ray said that the staff will optimize the review process by making certain infrastructure changes (i.e., making revisions to certain regulatory guides and standard review plan sections), by detailed planning (e.g., reviewers and applicants), by pre-application reviews, and by reviewing topical reports for issues that are generic and that can be reviewed in advance. He also said that the Agency is increasing the size and qualification of its staff, including contractor support, in response to new reactor licensing activities. Mr. Ray told the Committee that they had discussed the staff's design-centered review approach with industry and indicated that a regulatory information summary was about to be issued. He said that industry agreed to this approach and were organizing themselves into groups by reactor vendor technology type (e.g., AP1000, ESBWR, EPR, ABWR) and the associated COL applicants. Mr. Wilson said that the staff is considering amending the design certification rule for the Westinghouse AP1000 to provide for an even greater degree of standardization and prior staff approval. Mr Ray showed an illustration of how resources may be optimized for COL and design-centered reviews for FY 2007, FY 2008, and FY 2009.

Mr. Ray described and provided the Committee with the status of the development of the draft COL regulatory guide (DG-1145). DG-1145 will provide application content and process guidance for COL applications submitted under 10 CFR 52. The guidance will be provided in four major areas: (1) standard format and content, similar to that specified in Regulatory Guide 1.70; (2) supplemental information in areas such as PRA, ITAAC, and environmental report; (3) guidance for applicants that reference a certified design or early site permit; and (4) guidance in various miscellaneous topics (e.g., submittal specifications, general, and financial information). Dr. Apostolakis asked about the PRA scope and level-of-detail that would be available at the COL stage. Mr. Wilson said that the staff is in the process of developing PRA guidance and said that the Committee would have the opportunity to comment on the guidance when DG-1145 is sent to the Committee for review. The staff did indicate that a complete PRA may not be available at the COL stage because detailed operational procedures would not be available at that time. Mr. Wilson said that at the COL stage the applicant will at least have a design certification PRA plus the increase in scope needed to deal with site specific design features. Mr. Ray discussed the staff's schedule for developing DG-1145. A draft version of DG-1145 will be on the NRC's external webpage in June 2006. It will subsequently be sent out for a formal public comment period. The final guide will accompany the final Part 52 rule. Dr. Wallis questioned whether the staff should do more to get public comments from other than the usual industry stakeholders.

The staff also described its ongoing and coordinated efforts to update various regulatory guides and standard review plan sections in anticipation of COL applications. These updated guides and standards need to be completed six months in advance of receiving the first COL application (i.e., by March 2007) in order to make the same guidance applicable to all of the COL applicants and thus support the staff's design-centered approach. Mr. Ray said that new staff positions in the revised guides and standards would be sent to the ACRS for review. Guidance that is being consolidated or is just being re-formatted would not be sent to the ACRS for review. Mr. Maynard ask if the staff had considered having one submittal that would be applicable to several plants, as was done for the SNUPPS plants (Callaway and Wolf Creek) back in the 1980's. He said that they ended up with a combined joint FSAR, and each plant had an addendum for the site-specific aspects.

Mr. Tappert highlighted some of the efforts to recruit new staff to do the impending new reactor licensing (college graduates, experienced individuals, and re-employed annuitants). He said that NRR needs to hire over 300 new employees over the next couple of years which is above the anticipated attrition level. He indicated that NRR had hired over 170 people this year. He said that the principle role of the re-employed annuitants is to work with the younger staff for knowledge transfer. He described ongoing training initiatives and qualification programs to bring these new employees "up to speed" as regulators. He also mentioned NRR's Strategic

Workforce Planning initiative that is used to identify critical knowledge, skills, and abilities and compare them to the existing knowledge, skills, and abilities of employees for the purpose of projecting staffing and training needs. Finally, Mr. Tappert said that the staff is soliciting for contractor support to perform the new reactor licensing reviews.

Mr. Stuart Richards, Deputy Director, Division of Inspection and Regional Support, briefed the Committee on the staff's construction inspection program (CIP) development efforts and highlighted some of the anticipated inspection resource needs. He started by identifying some of the challenges associated with CIP development such as: the inspection of construction activities worldwide, performing timely inspections during an aggressive construction schedule, performing effective selection of inspections to verify ITAAC completion, and the regulatory framework is quite a bit different.

Mr. Richards mentioned a contractor report that had recently been sent to the ACRS that documents the staff's sampling plan for inspections, tests, analyses and acceptance criteria (ITAAC) during construction inspection. The ITAACs are first classified based on the activity required to satisfy the ITAAC and then grouped by this "same activity." The overall idea is that observing licensee performance of the activity with one component (or ITAAC) gives an idea of what is done for the other components in the group. Once grouped, the ITAACs are then rank-ordered based on defined attributes (e.g., safety significance, propensity of making errors, construction and testing/training experience, licensee oversight attention, opportunity to verify by other means). The ITAACs are then weighted according to their impact on the overall objective using an Analytic Hierarchy Process (AHP). The outcome of rating the ITAAC provides an idea of the value of inspecting that ITAAC.

Mr. Richards said that updating the construction inspection procedures is more administrative than technical because a lot of the construction techniques and work activities which have been done in the past are similar, i.e., concrete and welding inspections. He stated that there are four inspection manual chapters related to the CIP. IMC-2501 deals with early site permits and contains five inspection procedures each of which has been issued and has been used. IMC-2502 contains nine inspection procedures that support issuing a COL. These nine procedures have also been issued and are ready for use. IMC-2503 contains 25 inspection procedures which address specific attributes of the different kinds of ITAAC. Mr. Richards stated that these procedures will be issued over the next 12 months. Finally, IMC-2504 contains 150 procedures related to non-ITAAC work such as pre-operational and startup testing, operational programs, et al. He said these procedures will be issued over the next 18 months. Mr. Richards said that the staff has had the benefit of personnel with prior construction inspection experience to help develop the CIP. Dr. Bonaca asked how the staff would inspect components manufactured to foreign codes and standards. Mr. Wilson and Mr. Beckner explained that plants licensed in the United States would need to meet U.S. codes and standards. Mr. Beckner also mentioned the ongoing Multi-National Design Approval Program for the EPR design.

Mr. Richards said that the CIP requires off-site fabrication inspectors, construction resident inspectors, and construction specialist inspectors. He said that the staff envisioned having four staff onsite plus administrative support. One of the four onsite inspectors would be a dedicated scheduler type person working with the Construction Inspection Program Information Management System (CIPMS). There would also be up to three specialist inspectors from the regional office for each plant. The Committee discussed with the staff the pros and cons of having resident versus regional inspection support at new plant construction sites.

Dr. Beckner concluded by saying that the staff is preparing for an unprecedented level of new reactor licensing activities, but said that the actual level of activity is still uncertain. He said that the staff's ability to handle the anticipated heavy workload depends a lot on standardization and their design-centered review approach.

Committee Action

The Committee will use the information obtained during the briefing to formulate it's plans for the efficient and effective review of DCs, ESPs, and COL applications.

VI. Plant License Renewal Subcommittee Report (Open)

[Note: Mr. Cayetano Santos 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 May 30, 2006 meeting with the NRC staff and the Nuclear Management Company (NMC) to review the license renewal application for the Monticello Nuclear Generating Plant and the associated draft safety evaluation report (SER). The current operating license expires on September 8, 2010. During the meeting, NMC described the plant, its operating history, the license renewal review methodology, and its commitment tracking system. The staff's draft SER was issued on April 26, 2006 and contains no open or confirmatory items.

VII. Status Report on the Quality Assessment of Selected NRC Research Projects (Open)

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

The Committee discussed the status of the quality assessment of the research projects selected for FY 2006. The Committee agreed that the panel review of research project on Containment Capacity Studies should only be focused on draft NUREG/CR report entitled, "Containment Integrity Research at Sandia National Laboratory". This report summarizes the work that has been performed over the past thirty years to improve the understanding of the response of containment structures and their capacity to withstand accidents beyond design basis loads, and identifies common theme that have emerged.

Committee Action

The Committee plans to discuss the draft report on quality assessment of the selected projects during September 7- 9, 2006 ACRS meeting.

X. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations

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

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

- The Committee considered the EDO's response of May 18, 2006, to comments and recommendations included in the April 20, 2006 ACRS letter concerning the Draft Final Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of April 27, 2006, to comments and recommendations included in the March 23, 2006 ACRS report on the Safety Aspects of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of May 18, 2006, to comments and recommendations included in the April 14, 2006 ACRS letter on the Review of the 1994 Addenda to the ASME Code for Class 1, 2, and 3 Piping Systems and the Resolution of the Differences between the NRC staff and ASME. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of May 2, 2006, to comments and recommendations included in the March 24, 2006 ACRS report on the Final Review of the Exelon Generating Company, LLC, Application for Early Site Permit and the Associated NRC Staff's Final Safety Evaluation Report. The Committee decided that it was satisfied with the EDO's response.

  The staff plans to discuss proposed revisions to Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," with the ACRS during a future meeting.

- The Committee considered the EDO's response of May 18, 2006, to comments and recommendations included in the April 14, 2006 ACRS letter on the Grand Gulf Early Site Permit Application: Evaluation of Transportation Accidents on the Mississippi River.

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

- The Committee considered the EDO's response of May 2, 2006, to comments and recommendations included in the April 10, 2006 ACRS report on Generic Safety Issue-191 - Assessment of Debris Accumulation on PWR Sump Performance. The Committee decided that it was not satisfied with the EDO's response, and it will consider a further response following a staff's presentation to the Thermal-Hydraulic Phenomena Subcommittee on June 13-14, 2006.

  The staff plans to develop integrated plans to acquire sufficient technical bases to evaluate the proposed PWR sump modifications. This will include continuing to participate in industry efforts to address sump performance issues as well as incorporating the information obtained into the staff's issue resolution strategy. The staff plans to review the approaches used by each of the five vendors selected by licensees to support them in addressing GSI-191. The staff plans to develop and update guidance needed in some areas such as chemical effects and water management strategies. The ACRS Subcommittee on Thermal-Hydraulic Phenomena and/or the full Committee plans to discuss the above activities as progress has been made by the staff.

- The Committee considered the EDO's response of March 30, 2006, to the February 15, 2006 memorandum which forwarded an anonymous letter concerning the TRACE code that was received by Dr. Wallis and Dr. Ransom. The Committee decided that it was satisfied with the EDO's response.

  The staff has committed to discuss these comments in the context of a meeting with the Thermal-Hydraulic Phenomena Subcommittee later in the year regarding the status of the TRACE code.

- The Committee considered the EDO's response of May 15, 2006, to comments and recommendations included in the April 19, 2006 ACRS letter on Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs." The Committee decided that it was not satisfied with the EDO's response, but it will arrange for a future meeting to discuss the issue again with the staff.

  The staff committed to meet with the Committee to further discuss the respective points of view and reach a common understanding of this issue.

- The Committee considered the EDO's response of May 22, 2006, to the April 21, 2006 ACRS letter on the Application of the TRACG Computer Code to Evaluate the Stability of the Economic Simplified Boiling Water Reactor (ESBWR). The Committee decided that it was satisfied with the EDO's response.

The staff plans to discuss with the ACRS the results of the application of this code during ACRS review of the ESBWR design certification.

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

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on May 30, 2006. The following items were discussed:

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

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

Anticipated Workload for ACRS Members

The anticipated workload for the ACRS members through September 2006 were addressed. The objectives were:

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

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

Appointment of New Members to the ACRS

On May 9, 2006, the ACRS Member Candidate Screening Panel sent a list of candidates to the Commission, recommending appointment of three new members to the ACRS. The Commission has unanimously approved the Panel's and Committee's recommendation on appointing three new members to the ACRS, subject to security and conflict of interest reviews. These individuals will be invited to the July 2006 ACRS meeting as invited experts. In his vote sheet, Commissioner Lyons endorsed the ACRS/ACNW work on Knowledge Management. The Subcommittee commends the ACRS Executive Director on his recruitment of new members and facilitating the appointment process. Additionally, the Subcommittee commends the ACRS staff's work on Knowledge Management.

Quadripartite Meeting Status

On March 31, 2006, all ACRS abstracts for the 2006 Quadripartite meeting were uploaded to the web site. During the April ACRS meeting, these abstracts were provided to the members for review. During the June meeting, the members were provided with copies of the abstracts from Germany and Japan. The members are reminded that final papers and power point presentation slides are due by Friday, July 28, 2006.

In addition, several meeting attendees from some of the Member Countries are scheduled to visit the Calvert Cliffs Nuclear Plant on October 17, 2006. So far, Armijo, Maynard, Sieber, and Wallis plan to attend.

On July 5, 2006, letters will be sent to the Commissioners, EDO, and NRC Program Office Directors inviting them to participate or attend the Quadripartite Meeting.

Streamlining the NRR Rulemaking Process

In a memorandum (COMEXM-06-0006) dated April 7, 2006, Chairman Diaz and Commissioner McGaffigan sent a proposal to Commissioners Merrifield, Jaczko, and Lyons for streamlining the NRR Rulemaking Process. In that memo, it is stated that ". . . not withstanding 10 CFR 2.809 and the Memorandum of Understanding between the ACRS and the EDO, the staff may waive review by the ACRS at the proposed rule stage." Also, it is stated "comments from the ACRS may be submitted to the Commission either during the comment period for the proposed rule or following the close of the public comment period, but prior to issuance of the final rule."

If implemented, this proposal will limit the number of opportunities that the ACRS has now to review a proposed rule. Also, this will contradict Commission direction in previous SRMs. For example, in the April 5, 2000 SRM, the Commission stated that the ACRS should work with the NRC staff to enhance efforts to risk-inform 10 CFR Part 50, including Appendices A and B.

Also, in the April 13, 2006 SRM, the Commission stated that the ACRS and the staff should continue to work together to ensure that staff and ACRS reviews of important technical issues are coordinated in a manner to ensure timely resolution of these issues.

Without involvement by the ACRS in the early stages of the development of a proposed rule, the Committee may not be able to contribute effectively to the development of a rule. During the survey of the NRC staff related to 2005 self-assessment of ACRS, some NRC staff members stated that "early interaction by the ACRS with the EDO and the NRC staff on the regulatory significance of complex technical issues was very useful."

The ACRS Chairman, the Executive Director, and Deputy Executive Director contacted Commissioner McGaffigan on May 3, 2006 and provided comments on a draft version of the SRM. On May 31, 2006, a final SRM related to this matter was issued. The final SRM, which has been significantly changed compared to the previous version, incorporates the comments provided by the ACRS Chairman.

Visit to the Limerick Nuclear Plant and Meeting with the Region I Administrator

During the May ACRS meeting, the members decided to meet with the Region I Administrator on July 26 and visit the Limerick plant on July 27, 2006. A list of discussion topics proposed by Mr. Sieber was discussed. The following members have agreed to participate:

Wallis	Armijo
Sieber	Maynard
Shack	
Powers	

ACRS Meeting with the NRC Commissioners

The ACRS is tentatively scheduled to meet with the NRC Commissioners on Thursday, September 7, 2006, between 9:30 and 11:30 a.m. The Committee should approve a list of topics during the June meeting. Topics proposed by the Planning and Procedures Subcommittee are as follows:

I. Overview

- License Renewal
- Power Uprate
- Risk-Informing 10 CFR 50.46
- Future Activities

II. PWR Sump Performance

III. Safety Research Program Report

IV. Safety Conscious Work Environment/Safety Culture

V. Future Plant Design Activities

- ESP
- 10 CFR Part 52
- ESBWR

ACRS Letter Writing Process and Related Matters

Because of recent experience, the Committee should discuss whether changes, if any, are needed to make the letter writing process more efficient. The Committee appears to spend time on unrelated issues during the preparation of these reports than is necessary. This can be particularly distracting when the Committee is working on a plant specific application (e.g., license renewal, power uprate) and generic type issues unrelated to the application are introduced. The Committee should make sure that the contents of the letters, including additional comments, are relevant to the plant specific issues discussed. Comments not related to the subject matter of a particular letter diminishes the value of the recommendation on the main issue.

Staff Requirements Memorandum on Regulatory and Resource Implications of a DOE Spent Nuclear Fuel Recycling Program

In a Staff Requirements Memorandum dated May 16, 2006, the Commission directed the staff to focus on the development of a conceptual licensing process for the Administrations's Global Nuclear Energy Partnership (GNEP) related facilities. The conceptual process should consider the most effective and efficient elements of the NRC's licensing processes for major facilities, including review of the one-step licensing provisions for enrichment facilities, and features of the nuclear power plant combined licensing under 10 CFR Part 52 (i.e., construction authorization and operating license hearing process, design certification process, and early site permit process). The development of a conceptual licensing process is an inter-office undertaking, likely with NMSS in the lead, but NRR, NSIR, RES, and OGC all having significant roles. The Commission stated that the ACRS and ACNW could also help in defining the issues most important to licensing, inspecting, and ultimate decommissioning of reprocessing facilities (and related fuel-cycle facilities).

SRM related to an expanded work scope for the Center for Nuclear Waste Regulatory Analyses

In a SRM dated February 9, 2006 (pp 19-20), the Commission tasked the ACNW with providing recommendations to the Commission, with input from the ACRS, on how the Center for Nuclear Waste Regulatory Analyses (CNWRA) might broaden its assistance to the NRC. The CNWRA is part of the Southwest Research Institute and the ACNW has been providing advice to the Commission on the CNWRA's programs for a number of years. The work at CNWRA is being managed by NMSS as a Technical Assistance Program, and at present it is primarily focused on providing support for NRC activities associated with the proposed Yucca Mountain repository. NMSS, NRR, and RES are aware of the CNWRA's capabilities and are considering possible additional use of the CNWRA's expertise. Dana Powers, as the ACRS lead member on the review of the NRC's research programs, is being kept informed of the ACNW activities in developing a response to the Commission request. CNWRA representatives will brief the ACNW on the status of its current programs in July 2006. NMSS, NRR, and RES will be invited to discuss these offices' views on broadening the CNWRA's assistance to the NRC.

Seminar on 9/11 Event

The staff plans to hold a seminar on September 11, 2006 during which representatives from NIST will present the results of their analysis of the 9/11 event, specifically the impact damage, the fire effects on structural steel, and the collapse of the twin towers. The staff invites interested ACRS members to attend this seminar. Also, the staff would like to know whether any ACRS members are interested in visiting the NIST Fire Research Facilities in Gaithersburg.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 534th ACRS Meeting, July 12-14, 2006.

The 533rd ACRS meeting was adjourned at 7:00 p.m. on June 1, 2006.

alternatives to the proposed action, including the no-action alternative. NRC staff assesses the impacts of the proposed action and its alternatives on public and occupational health, air quality, water resources, waste management, geology and soils, noise, ecology resources, land use, transportation, historical and cultural resources, visual and scenic resources, socioeconomics, accidents and environmental justice. Additionally, the FEIS analyzes and compares the costs and benefits of the proposed action.

Based on the evaluation in the FEIS, NRC environmental review staff has concluded that the proposed action would have small effects on the physical environment and human communities with the exception of: (1) Short-term moderate impacts associated with increases in particulate matter released to the air during the construction phase; (2) short-term moderate impacts related to increased traffic congestion during the construction phase; (3) potential moderate impacts due to transportation accidents; (4) potential moderate impacts from facility operation accidents; (5) moderate impacts associated with a potential operating extension of the DOE depleted uranium tails conversion facility; and (6) moderate employment impacts on the local communities associated with the construction and operation phases.

After weighing the impacts, costs, and benefits of the proposed action and comparing alternatives, NRC staff, in accordance with 10 CFR part 51.91(d), set forth their final recommendation regarding the proposed action. NRC staff recommend that, unless safety issues mandate otherwise, the action called for is the approval of the proposed action (i.e., issue a license).

NRC staff in the Division of Fuel Cycle Safety and Safeguards are currently completing the safety review for USEC's license application and is currently scheduled for completion in June 2006.

Dated at Rockville, Maryland, this 9th day of May 2006.

For the Nuclear Regulatory Commission.

Scott C. Flanders,

Deputy Director, Environmental and Performance Assessment Directorate, Division of Waste Management and Environmental Protection, Office of Nuclear Material Safety and Safeguards.

[FR Doc. E6–7364 Filed 5–12–06; 8:45 am]

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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 May 31—June 1, 2006, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Tuesday, November 22, 2005 (70 FR 70638).

Wednesday, May 31, 2006, 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.–11:30 a.m.: Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Nuclear Energy Institute regarding the draft final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations."

1:30 p.m.–3 p.m.: Draft Final Generic Letter 2006–xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final Generic Letter 2006–xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems."

3:15 p.m.—4:15 p.m.: Interim Staff Guidance on Aging Management Program for Inaccessible Areas of Boiling Water Reactor (BWR) Mark I Containment Drywell Shell (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed Interim Staff Guidance on Aging Management Program for Inaccessible Areas of BWR Mark I Containment Drywell Shell.

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

Thursday, June 1, 2006, 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.–11 a.m.: Overview of New Reactor Licensing Activities (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding staff's activities associated with the licensing of new reactors; early site permits; and combined license applications, as well as the related schedule and milestones.

11:15 a.m.–11:45 a.m.: Subcommittee Report (Open)—The Committee will hear a report by and hold discussions with the cognizant Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim review of the license renewal application for the Monticello Nuclear Power Plant.

12:45 p.m.–1:15 p.m.: Status Report on the Quality Assessment of Selected NRC Research Projects (Open)—The Committee will hear a report by and hold discussions with the cognizant Panel Chairman regarding the status of the quality assessment of selected NRC research projects.

1:15 p.m.–2 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

2 p.m.–2:15 p.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.

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

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 29, 2005 (70 FR 56936). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Cognizant ACRS staff (301–415–7364), between 7:30 a.m. and 4:15 p.m., ET.

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

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

Dated: May 9, 2006.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E6–7348 Filed 5–12–06; 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 May 30 2006, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

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

The agenda for the subject meeting shall be as follows:

Tuesday, May 30, 2006, 11 a.m.–12:30 p.m.

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

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Sam Duraiswamy (telephone: 301–415–7364) between 7:30 a.m. and 4:15 p.m. (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 4:15 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: May 8, 2006.

Michael R. Snodderly,

Acting Branch Chief, ACRS/ACNW.

[FR Doc. E6–7349 Filed 5–12–06; 8:45 am]

BILLING CODE 7590–01–P

PENSION BENEFIT GUARANTY CORPORATION

Required Interest Rate Assumption for Determining Variable-Rate Premium for Single-Employer Plans; Interest Assumptions for Multiemployer Plan Valuations Following Mass Withdrawal

AGENCY: Pension Benefit Guaranty Corporation.

ACTION: Notice of interest rates and assumptions.

SUMMARY: This notice informs the public of the interest rates and assumptions to be used under certain Pension Benefit Guaranty Corporation regulations. These rates and assumptions are published elsewhere (or can be derived from rates published elsewhere), but are collected and published in this notice for the convenience of the public. Interest rates are also published on the PBGC's Web site *http://www.pbgc.gov*.

DATES: The required interest rate for determining the variable-rate premium under part 4006 applies to premium payment years beginning in May 2006. The interest assumptions for performing multiemployer plan valuations following mass withdrawal under part 4281 apply to valuation dates occurring in June 2006.

FOR FURTHER INFORMATION CONTACT: Catherine B. Klion, Attorney, Legislative and Regulatory Department, Pension Benefit Guaranty Corporation, 1200 K Street, NW., Washington, DC 20005, 202–326–4024. (TTY/TDD users may call the Federal relay service toll-free at 1–800–877–8339 and ask to be connected to 202–326–4024.)

SUPPLEMENTARY INFORMATION:

Variable-Rate Premiums

Section 4006(a)(3)(E)(iii)(II) of the Employee Retirement Income Security Act of 1974 (ERISA) and § 4006.4(b)(1) of the PBGC's regulation on Premium Rates (29 CFR part 4006) prescribe use of an assumed interest rate (the "required interest rate") in determining a single-employer plan's variable-rate premium. The required interest rate is the "applicable percentage" (currently 85 percent) of the annual yield on 30-year Treasury securities for the month preceding the beginning of the plan year for which premiums are being paid (the "premium payment year"). The required interest rate to be used in determining variable-rate premiums for premium payment years beginning in May 2006 is 4.30 percent (*i.e.*, 85 percent of the 5.06 percent Treasury Securities Rate for April 2006).

The Pension Funding Equity Act of 2004 ("PFEA")—under which the

APPENDIX II

May 9, 2006

SCHEDULE AND OUTLINE FOR DISCUSSION
533rd ACRS MEETING
MAY 31 - JUNE 1, 2006

WEDNESDAY, MAY 31, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1)	8:30 - 8:35 A.M.	Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD) 1.1) Opening statement 1.2) Items of current interest	
2)	8:35 - 11:30 A.M. (~~10:00-10:15~~ **BREAK)** 10:10-10:20 A.M.	Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations" (Open) (RSD/MAJ/HPN) 2.1) Remarks by the Subcommittee Chairman 2.2) Briefing by and discussions with representatives of the NRC staff and Nuclear Energy Institute regarding the draft final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations." Members of the public may provide their views, as appropriate.	
	11:30 - 1:30 P.M.	*****LUNCH*****	
3)	1:30 - ~~3:00~~ P.M. 2:23	Draft Final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems" (Open) (MVB/CS) 3.1) Remarks by the Subcommittee Chairman 3.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems." Representatives of the nuclear industry and members of the public may provide their views, as appropriate.	
	3:00 - 3:15 P.M.	*****BREAK*****	
4)	3:15 - 4:15 P.M.	Interim Staff Guidance on Aging Management Program for Inaccessible Areas of Boiling Water Reactor (BWR) Mark I Containment Drywell Shell (Open) (MVB/CS) 4.1) Remarks by the Subcommittee Chairman 4.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed Interim Staff Guidance on Aging Management Program for Inaccessible Areas of BWR Mark I Containment Drywell Shell.	

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

	4:15 - 4:30 P.M.	*****BREAK*****
5)	4:30 - 6:30 P.M.	Preparation of ACRS Reports (Open) Discussion of proposed ACRS reports on: 5.1) Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations" (RSD/MAJ/HPN) 5.2) Draft Final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failure that Disable Accident Mitigation Systems" (MVB/CS)

THURSDAY, JUNE 1, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

6)	8:30 - 8:35 A.M.	Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD) 6.1) Opening statement 6.2) Items of current interest
7)	8:35 - ~~11:00~~ A.M. 10:00 **(10:00-10:15 BREAK)**	Overview of New Reactor Licensing Activities (Open) (TSK/DCF) 7.1) Remarks by the Subcommittee Chairman 7.2) Briefing by and discussions with representatives of the NRC staff regarding staff's activities associated with the licensing of new reactors; early site permits; and combined license applications, as well as the related schedule and milestones.
	11:00 - 11:15 A.M.	*****BREAK*****
8)	~~11:15~~ - 11:45 A.M. 1:15	Subcommittee Report (Open) (MVB/CS) Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding interim review of the license renewal application for the Monticello Nuclear Power Plant.
	~~11:45 - 12:45~~ P.M. 12:00-1:00	*****LUNCH*****
9)	~~12:45 - 1:15~~ P.M. 1:00-1:15	Status Report on the Quality Assessment of Selected NRC Research Projects (Open) (GBW/HPN) Report by and discussions with the cognizant Panel Chairman regarding the status of the quality assessment of selected NRC research projects.
10)	~~1:15 -2:00~~ P.M. 10:15-11:50 A.M.	Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD) 10.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

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

11) ~~2:00 - 2:15~~ P.M. 11:50-12:00 — Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

2:15 - 2:30 P.M. — *****BREAK*****

12) 2:30 - 6:30 P.M. — Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
12.1) Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations" (RSD/MAJ/HPN)
12.2) Draft Final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failure that Disable Accident Mitigation Systems" (MVB/CS)

13) 6:30 - 7:00 P.M. — Miscellaneous (Open) (GBW/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.
- Thirty-Five (35) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.

APPENDIX III

MEETING ATTENDEES

533rd ACRS MEETING
MAY 31-JUNE 1, 2006

NRC STAFF (5/31/2006)

M. Mayfield, NRR
C. Jackson, NRR
K. Tanabe, NRR
R. DeLaGarza, NRR
D. Merzke, NRR
D. Wrona, NRR
D. Haung, NRR
N. Dudley, NRR
S. Hoffman, NRR
L. Tran, NRR
T. Le, NRR
H. Ashar, NRR
J. Davis, NRR
R. Karas, NRR
L. Lund, NRR
T. Terry, NRR
D. Frumkin, NRR
R. Wofgang, NRR
A. Klein, NRR
T. Quay, NRR
R. Woods, RES
G. Morris, NRR
B. Richter, NRR
T. Dinh, NRR
D. Andrukat, NRR
M. H. Salley, RES
J. Presbach, RES
A. Kouchinsky, RES
T. Koshy, NRR
K. Corp, NRR
K. Tanabe, NRR
J. Vora, RES
A. Wilson, RES
G. Wilson, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

A. Marion, NEI
D. Miskiewics, Progress Energy
H. Barrett, Duke Power
B. Jamar, NEI
N. Chapman, SERCH/Bechtel
S. Waino, Dominion Nuclear
S. Dolley, Platts/Inside NRC
D. Raleigh, LIS, Scientech
L. Seamans, NMC-Palisades
M. Fallin, Constellation Energy
J. Ross, NEI

NRC STAFF (6/1/2006)

J. Wilson, NRR
S. Bloom, NRR
K. Cozens, NRR
B. Beckner, NRR
C. Nolan, NRR
V. Klco, NRR
S. Lingam, NRR
A. Ziedonis, NRR
T. Wengert, NRR
P. Cochran, RES
J. Lamb, OEDO
J. Ridgely, RES
J. Williams, NRR
S. Richards, NRR
J. Ortega, NRR
S. Coffin, NRR
P. Ray, NRR
R. Jervey, NRR
M. Kowal, NRR
D. Szwarc, NRR
A. Nielson, RII
D. Merzke, NRR
J. Mitchell, RES
C. Ader, RES
I. Ata, RES
R. Assa, RES
S. Ali, RES
A. Sheikh, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

J. Coron, SRI International
A. Trepod, SRI International
A. Levin, Areva NP

June 20, 2006

SCHEDULE AND OUTLINE FOR DISCUSSION
534th ACRS MEETING
JULY 12-14, 2006

WEDNESDAY, JULY 12, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:00 A.M. Final Review of the License Renewal Application for the Nine Mile Point Nuclear Station (Open) (JDS/MAJ)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Constellation Energy Company, LLC regarding the license renewal application for the Nine Mile Point Nuclear Station, Units 1 and 2 and the associated NRC staff's final Safety Evaluation Report.

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

3) 10:15 - 11:45 A.M. Results of the Study to Determine the Need for Establishing Limits for Phosphate Ion Concentration (Open) (DAP/CS)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff and their contractor regarding the results of the study for use by the staff in deciding on the need for establishing limits for phosphate ion concentration in groundwater at the sites of plants applying for license renewal.

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

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

4) 12:45 - 4:00 P.M. **(2:15-2:30 P.M. BREAK)** Integrating Risk and Safety Margins (Open) (WJS/HPN/EAT)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff regarding a proposed framework for integrating risk and safety margins.

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

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

5) 4:15 - 4:45 P.M. Subcommittee Report (Open) (GBW/RC)
Report by and discussions with the chairman of the ACRS Subcommittee on Thermal-Hydraulic Phenomena regarding the status of activities associated with the resolution of Generic Safety Issue-191 - Assessment of Debris Accumulation on PWR Sump Performance that were discussed during the June 13-14, 2006 Subcommittee meeting.

6) 4:45 - 6:45 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
- 6.1) Final Review of the License Renewal Application for the Nine Mile Point Nuclear Station, Units 1 and 2 (JDS/MAJ)
- 6.2) Results of the Study to Determine the Need to Establish Limits on Phosphate ion concentration (DAP/CS)
- 6.3) Integrating Risk and Safety margins (WJS/HPN/EAT)
- 6.4) Response to the May 2, 2006 Letter from the NRC Executive Director for Operations Responding to the March 24, 2006 (Revised April 10, 2006) ACRS Report on GSI-191- Assessment of Debris Accumulation on PWR Sump Performance (GBW/RC)

THURSDAY, JULY 13, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - 10:30 A.M. Safeguards and Security Matters (Closed) (MVB/EAT)
- 8.1) Remarks by the Subcommittee Chairman
- 8.2) Briefing by and discussions with representatives of the NRC staff regarding safeguards and security matters.

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

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

9) 10:45 - 11:00 A.M. Subcommittee Report (Open) (MVB/EAT)
Report by and discussions with the Acting Chairman of the ACRS Subcommittee on Digital Instrumentation and Control Systems regarding matters discussed during the Subcommittee meeting on June 27, 2006.

10) 11:00 - 12:00 Noon Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)
- 10.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
- 10.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

11) 12:00 - 12:15 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

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

12) 1:15 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
12.1) Final Review of the License Renewal Application for the Nine Mile Point Nuclear Station, Units 1 and 2 (JDS/MAJ)
12.2) Results of the Study to Determine the Need to Establish Limits on Phosphate Ion Concentration (DAP/CS)
12.3) Integrating Risk and Safety margins (WJS/HPN/EAT)
12.4) Response to the May 2, 2006 Letter from the NRC Executive Director for Operations Responding to the March 24, 2006 (Revised April 10, 2006) ACRS Report on GSI-191- Assessment of Debris Accumulation on PWR Sump Performance (GBW/RC)

FRIDAY, JULY 14, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

13) 8:30 - 12:00 Noon **(10:15-10:30 A.M. BREAK)** Preparation of ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 12.

14) 12:00 - 12:30 P.M. Miscellaneous (Open) (GBW/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.

- Thirty-Five (35) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.

APPENDIX V

LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
533RD ACRS MEETING
MAY 31-JUNE 1, 2006

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

MEETING HANDOUTS

AGENDA ITEM NO. — DOCUMENTS

1 Opening Remarks by the ACRS Chairman
1. Items of Interest dated May 31-June 1, 2006

2 Draft Generic Letter, "Post-Fire Safe-Shutdown Circuit analysis Spurious Actuations"
2. Final Draft Generic Letter "Post-Fire Safe-shutdown Circuits Analysis Spurious Actuations" presentation by NRR [Viewgraphs]
3. Bounding the Fire Risk from Circuit Spurious Actuations at Nuclear Power Plants written by Raymond H.V. Gallucci, Ph.D., P.E. [Handout]
4. Bounding Risk Analysis for Multiple Spurious Actuations presentation on Generic Letter 2006-xx, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations" [Viewgraphs]

3 Draft Final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems"
5. Presentation to ACRS 1:30 pm, May 31, 2006, Generic Letter on Inaccessible or Underground Cable Failures [Viewgraphs]

4 Interim Staff Guidance on Aging Management Program for Inaccessible Areas of Boiling Water Reactor (BWR) Mark I Containment Drywell Shell
6. Proposed License Renewal Interim Staff Guidance LR-ISG-2006-01: Plant-Specific Aging Management Program for Inaccessible Areas of BWR Mark I Steel Containment Drywell Shell presentation by NRR [Viewgraphs]

7 Overview of New Reactor Licensing Activities
7. Challenges and Strategies for Licensing New Reactors presentation by NRR [Viewgraphs]

10 Future ACRS Activities/Report of the Planning and Procedures Subcommittee
8. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - May 30, 2006 [Handout #10.1]

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

9 Status Report on the Quality Assessment of Selected NRC Research Projects
10. Proposed Schedule for Quality Assessment of Selected NRC Research Projects [Handout]

8 Subcommittee Report

MEETING NOTEBOOK CONTENTS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
533rd FULL COMMITTEE MEETING

May 31- June 1, 2006

TODAY'S DATE: May 31, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Michael Mayfield	NRR/DE
2	Chris Jackson	NRR / DPR
3	Kiyoto Tanabe	NRC/NRR/DLR/RLRC. F.A.
4	Rodrigo De La Garza	NRR/DLR
5	DANIEL MERZKE	NRR /DLR
6	DAVE WRONA	NRR /DLR
7	DAN HOANG	NRR/DLR
	NOEL DUDLEY	NRC/DLR
9	Steve Hoffman	NRC/NRR/DLR
10	Linh Tran	NRC/NRR/DLR/ RLRB
11	Tommy Le	NRC/DLR
12	HANS ASHAR	NRC/DE
13	JIM DAVIS	NRC/NRR/DLR / RLRC
14	Ken Chang	NRC/NRR/RLRC
15	Rebecca Karas	NRC/ NRR/DE/EGCB
16	Louise Lund	NRC / NRR/ DLR/ RLRA
17	Tomeka Terry	NRC /NRR/DE/EGCB
18		
20		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
533rd FULL COMMITTEE MEETING

May 31- June 1, 2006

TODAY'S DATE: May 31, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Daniel Frumkin	NRR/DRA/AFPB
2	Robert Wolfgang	NRR/DRA/AFPB
3	Alex Klein	NRR/DRA/AFPB
4	TED QUAY	NRR/DRA/ DD
7	~~Mike Junge~~	~~ACRS~~
5	Roy Woods	RES/DRASP/OERA
6	GEORGE MORRIS	NRR/DE/EEEB
	BRIAN RICHTER	NRR/DPR/PRAB
9	Chris Jackson	NRR/DPR/PGCB
10	Thinh Dinh	NRR/DRA/AFPB
11	DENNIS ANDRUKAT	NRR/DRA/AFPB
12	Mark Henry Salley	NRC/RES
13	JASON DREISBACH	NRC/RES
14	Alan Kouchinsky	NRC/RES
15	T. KOSHY	EEEB/DE/NRR
16	K. CORP	EEEB/DE/NRR
17	Kiyoto Tanabe	NRC/NRR/DLR/RLRC,EA.
18	Jit Vora	NRC/RES.
	Adam Wilson	RES
20	George Wilson	NRC/EEEB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
533rd FULL COMMITTEE MEETING
May 31 - June 1, 2006

May 31, 2006
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

	NAME	AFFILIATION
1	ALEX MARION	NEI
2	DAVID MISKIEWICZ	Progress Energy
3	Harry Barrett	Duke Power
4	BRANDON JAMAR	NEI
5	Nancy Chapman	SERCH / Bechtel
6	STEVE WAINIO	DOMINION NUCLEAR CT MPS
7	Steven Dolley	Platts / [illegible] NRC
8	Diana Raleigh	LIS, Scientech
9	LARRY SEAMANS	NMC - Palisades
10	MICHAEL FALLIN	CONSTELLATION ENERGY
11	James Ross	NEI
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
533rd FULL COMMITTEE MEETING

May 31- June 1, 2006

TODAY'S DATE: June 1, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	JERRY N. Wilson	NRR/DNRL
2	Steven D Bloom	NRR/DNRL
3	KURT COZENS	NRR/DNRL
4	Bill Beckner	NRR/DNRL
5	CHRIS NOLAN	NRR/DNRL
6	VINCE KLCO	NRR/DNRL
7	SIVA P. LINGAM	NRR/DORL
	Adam Ziedonis	NRR/DSS/SBWB
9	Tom Wengert	NRR/DORL
10	Pete Cochran	RES/NARB
11	JOHN G. LAMB	NRC/EDO
12	John W. Ridgely	NRC/RES
13	JOE WILLIAMS	NRC/DNRL
14	STU RICHARDS	NRC/DIRS
15	Jonathan Ortega	NRR/DIRS
16	STEPHANIE COFFIN	NRR/DNRL
17	Phillip Ray	NRR/DNRL
18	R A JERVEY	NRR/DORL
	Mark Kowal	NRR/DNRL
20	Dariusz Szwarc	NRR/DRA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
533rd FULL COMMITTEE MEETING

May 31- June 1, 2006

TODAY'S DATE: June 1, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Adam Nielsen	Region 2 / DRS
2	DANIEL MERZKE	NRR / DLR
3	Jocelyn Mitchell	RES
4	Charles Ader	RES
5	ISTAR, ATA	RES
6	Ramin Assa	RES
7	Syed Ali	RES
	ABDUL SHEIKH	RES
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
533rd FULL COMMITTEE MEETING
May 31 - June 1, 2006

June 1, 2006
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

	NAME	AFFILIATION
1	JEFF CORON	SRI INTERNATIONAL
2	Allison Trepod	SRI INTERNATIONAL
3	ALAN LEVIN	AREVA NP
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ITEMS OF INTEREST

533RD ACRS MEETING

MAY 31- June 1, 2006

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
533rd MEETING
May 31 - June 1, 2006

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-06-012

"A Changing Paradigm"

Prepared Remarks by

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

at the

Conference of Radiation Control Program Directors

Detroit, Michigan
May 7, 2006

I am pleased to be here today. Since 1968 your organization has played an important role in the regulation of radioactive materials – providing a forum for radiation protection officials to regularly communicate with each other. The regulatory work that the U.S. Nuclear Regulatory Commission (NRC) and state radiation control agencies do is vital because we are responsible for the safe use of the nuclear materials with which the public is most likely to come into contact.

Nuclear Power Plants get a lot of the publicity in this business, but it is the myriad of productive uses of radioactive materials in medical, research and industrial applications that provide the most opportunities for interaction between people and radioactive materials. This fact highlights the need for strong, effective, consistent and transparent regulation, control and security.

We understand the benefits of radioactive materials when properly handled. In the last year and a half I have visited medical facilities including Johns Hopkins University in Maryland and the University of Pennsylvania to meet with materials licensees and hear about the benefits and challenges they face in using radioactive materials for medical diagnosis, therapy and research. During my visit to the University of Pennsylvania I had an opportunity to stop by the Children's Hospital. I met Dr. John Maris who oversees neuroblastoma treatments in children, which is an iodine-131 base therapy. Meeting Dr. Maris to discuss his work, in which parents serving as primary care givers are exposed to radiation during the treatment, proved to be an important discussion as I later approved a measure before the Commission allowing licensees to justify doses to care givers on a case-by-case basis.

We are all also very familiar with the improper use of these materials. I think I only need to invoke a single example to underscore the importance of setting clear rules and inspecting licensees to ensure they are meeting them – and it is an incident that occurred shortly after I became a Commissioner last year. An infant, not yet six months old, was being treated at a mid-western hospital. Through a mis-administered dose, this infant received more than twenty times the prescribed amount of technetium-99 metastable. The whole body dose to the patient was calculated between 5 and 10 REM.

Of course, we will never eliminate human error, but we have adhered to a philosophy of defense-in-depth to limit the negative aspects of incorporating radioactive materials into our everyday lives. That is where the NRC and state radiation control agencies provide the most value: ensuring that proper precautions are followed, due diligence is maintained, and enforcement actions are taken in the event of violations.

If I were giving this talk before 2001, I could probably wrap up a little early by concluding with a discussion of how we do that – how we regulate to control the use of radioactive materials to avoid unnecessarily exposing members of the public. Today, however, I must go further and focus not only on controlling these materials, but also on securing them.

There is an important distinction between controlling and securing. It is relatively new for the community that uses nuclear material to have to think about the possibility that someone would seek to use those materials for malevolent purposes. The events of September 11, 2001, have forced us into a new paradigm – one that not only requires the NRC and State agencies to continue to strengthen efforts to control sources, but to also better secure them.

This shift has broad implications for the relationship between the NRC and the Agreement States. The Congress recognized way back in 1959 that it could be beneficial to provide the federal agency responsible for regulating the use of radioactive materials with the authority to enter into agreements delegating to state agencies the responsibility for controlling their use. While the NRC has direct responsibility over approximately 4,500 licensees, those agreement states regulate more than 20,000 specific and 150,000 general materials licensees. With Minnesota recently becoming an agreement state – raising the number to 34 – those numbers continue to shift. This approach has proven to be efficient because proximity and familiarity fosters closer relationships between regulators and users of radioactive materials.

The recovery effort to ensure these materials were accounted for following hurricane Katrina demonstrates the effectiveness of the agreement state program. State officials were the first on the ground - and they coordinated effectively with the NRC and other federal agencies in accounting for the various radioactive materials and devices. While this arrangement makes perfect sense for the public health and safety issues related to controlling the use of these materials, it presents some challenges to the common defense and security responsibilities of the federal government.

One must look no further than last year's Energy Policy Act to see how the paradigm is changing – to see how the security of radioactive materials is an evolving and increasing area of focus. The Act added federal requirements for nuclear facilities and materials security. It added requirements for export controls of radiation sources, radiation source tracking, the creation of a Task Force on Radiation Source Protection and Security; requirements for fingerprinting and criminal history record checks on individuals having unescorted access to radioactive materials, and the secure transfer of nuclear materials that requires people accompanying or receiving these materials to be subject to a security background check by an appropriate federal entity.

While the NRC has the primary responsibility for executing these new requirements, to succeed we must cooperate with other federal agencies – the Department of Homeland Security, the Department of Energy, the Department of Transportation – involved in ensuring the security of the U.S. As recently as March of this year, Customs and Border Protection committed to working with us to implement a program to determine the legitimacy of a license for entities bringing nuclear materials across the border. The NRC will continue to see an increase in requirements for coordination with a wide array of Federal, State, and local agencies related to protecting the homeland.

It is natural and appropriate for the NRC to play the leadership role in these efforts. After all, Agreement State programs, while a crucial part of the effort to control radioactive materials, are not in the position to lead federal government coordination activities relating to common defense and security.

The national source tracking system is a good example of how I believe federal and state responsibilities will complement each other in this new paradigm focused both on control and security. Agreement State programs will, of course, continue to have regulatory authority over the materials licensees in their states for public health and safety issues. The NRC will have a legitimate need for information from those licensees to incorporate into a new national framework designed to make all 50 states more secure. I believe a national source tracking system built upon the foundation of the NRC's common defense and security authority is critical.

I would like to comment on an important aspect of the national source tracking system -- I also strongly believe that we should seize the opportunity as the Commission finalizes the national source tracking system rule, to include category 3 sources which, when aggregated, could pose the same level of risk as category 1 and 2 material.

I would like to mention one other example of how the regulatory paradigm is changing. The Energy Policy Act also included broad new authority for the NRC to regulate naturally occurring and accelerator produced radioactive material (NARM). It will be beneficial to continue discussions with you about how you have handled these issues as we develop regulations to implement this program. Just as with the issue of security, the NRC is in a position with this new NARM authority to develop a consistent national framework for dealing with these issues that should benefit every state.

Finally, I would be remiss if I did not take the opportunity to mention how I think you can contribute even more to this new paradigm. In 1999 the NRC and the states looked into options for a National Materials Program from an alliance approach to the current blended option of the NRC delegating regulatory authority to some states and retaining it in others. In light of the changing paradigm that I have discussed here today, it may be an appropriate time to reevaluate the options for this program. As new national responsibilities are implemented, and the responsibilities for control versus securing sources are more clearly defined, it may make sense to move towards more states becoming Agreeement States. This situation could allow for a more consistent nationwide framework of state and federal responsibilities and I look forward to hearing your views regarding this idea.

Again, I am pleased to have the opportunity to be part of your conference, and look forward to continuing to work in partnership with you on the important issues of controlling and securing radioactive materials used daily in our society. Thank you for inviting me, I look forward to the presentations and discussions to follow, and I would welcome any questions you may have for me now or throughout the day.

May 3, 2006

The Honorable Jerry Weller
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Weller:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter of February 15, 2006, concerning the tritium contamination that has been identified at the Braidwood and Dresden facilities. The Commission understands your concerns regarding this issue and we have been actively addressing it. Although NRC assessments do not indicate a hazard to public health or safety, we understand the public concern about the release of radioactive material, even in amounts that do not pose a hazard, in a manner that is not intended. The NRC's objective is to determine the circumstances that led to the contamination and to ensure that the plant operators take appropriate corrective actions.

A number of developments associated with the Braidwood tritium contamination have occurred since your letter was received. Because the corrective actions being taken by plant operators and the NRC's oversight of these actions are evolving daily, the detailed summary and time line of events you requested in your letter will be forwarded to you under separate cover as soon as practicable. In this letter, I describe the NRC's strategic approach toward resolving this problem.

The NRC responded appropriately to the recent developments in accordance with the established inspection program, policies, and the relative safety significance. The NRC Region III office promptly initiated an inspection of tritium-related issues at all operating nuclear power plants in the State of Illinois, including the Zion facility, which has been permanently shut down. This inspection is in addition to the routine examination of effluent and environmental monitoring programs at all nuclear power reactors performed pursuant to the NRC's established Reactor Oversight Program (ROP). All commercial nuclear reactors in the United States release liquid effluents containing tritium. These controlled releases range from hundreds of curies to about 1,500 curies per year. All of these controlled releases are within the allowances of technical specifications established during licensing that ensure that potential doses remain within regulatory limits. Typically, the contribution of controlled releases of tritiated water to the dose that could be received by a member of the public is a small fraction of a millirem per year. For comparison, a millirem is the dose a person would receive from spending about 25 hours in the U.S. Capitol. The NRC staff has also commenced an augmented sampling and analysis program to provide a level of independent measurement of environmental ground water sampling and to verify the adequacy of the licensees' groundwater analysis. The NRC's primary focus has been, and will continue to be, to ensure that the extent and level of contamination are accurately determined and that appropriate corrective actions are taken to prevent future recurrence. The staff will evaluate the outcomes of the inspection activities and develop enforcement actions, as appropriate, consistent with the NRC's Enforcement Policy and the ROP. Copies of the NRC inspection reports documenting the licensee's and the NRC's actions, as well as any enforcement actions which may be taken, will be sent to your office and will also be public.

In addition, the NRC has established a task force to conduct a comprehensive evaluation of the inadvertent, unmonitored releases of radioactive liquids containing tritium from all U.S. commercial nuclear power plants, including the regulatory requirements associated with the structures, systems, and components from which the releases emanated. The task force will identify and recommend areas for improvement that may be applicable to either or both the NRC and the industry. This task force is scheduled to complete its review in late summer 2006.

The NRC will remain vigilant to ensure that appropriate corrective actions are taken. Based on currently available information, the Commission does not believe that contamination issues at Braidwood and Dresden pose a hazard for public health and safety. If you have further questions or would like a briefing on this issue, please contact me.

Sincerely,

/RA/

Nils J. Diaz

May 15, 2006

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

Dear Mr. Chairman:

The Fiscal Year (FY) 2006 Energy and Water Development Appropriations Act, House Reports 109-86 and 109-275, directed the U.S. Nuclear Regulatory Commission (NRC) to provide a quarterly report on the status of its licensing and other regulatory activities. Previous reports were provided to you on a monthly basis, in accordance with the FY 2005 Energy and Water Development Appropriations Act, House Reports 108-554 and 108-792. The initial reporting requirement arose in the FY 1999 Energy and Water Development Appropriations Act, Senate Report 105-206. On behalf of the Commission, I am pleased to transmit the eighty-sixth report, which covers January - March 2006.

I am also providing in this cover letter additional information on several issues in order to keep you fully and currently informed of NRC's licensing and regulatory activities. The NRC recently identified several instances of unintended tritium releases from a few nuclear power plants. Even though information provided to date indicates there was no threat to the public health and safety, the NRC is reviewing these incidents to ensure that nuclear plant operators have taken appropriate action and to determine what, if any, changes are needed to the agency's rules and regulations. In March, the NRC assembled a task force to examine the issue of inadvertent, unmonitored releases of radioactive liquids containing tritium from U.S. commercial nuclear power plants. The task force is required to address several topics, including a general assessment of the potential public health impact from these releases; how the issue was communicated to the public, state and local officials, other Federal agencies, Congress, and other interested groups; a review of other inadvertent releases of tritium at nuclear power plants, including decommissioning sites, from 1996 to the present; industry actions in response to the releases, including the timing of remediation efforts; and NRC oversight of inadvertent releases, both under the Reactor Oversight Process (ROP) and the process in place prior to the ROP. A written report summarizing the task force's findings will be issued later this year. The NRC has also conducted and participated in several public meetings to discuss tritium levels in groundwater and the safety of public drinking water.

In FY 2001 and FY 2002 appropriations acts (P.L. 106-377 and P.L. 107-66), Congress provided funding to the NRC to provide financial assistance to the States for the remediation of formerly NRC-licensed sites. Subsequently, the NRC established a grant program to execute this financial assistance program for the purposes of reviewing files, conducting surveys, and characterizing and remediating (including regulatory oversight by States) sites formerly licensed by the Commission. All of the former sites under the grant program are located in States with which the NRC has entered into Agreements under Section 274 of the Atomic Energy Act.

Through cooperative efforts with the nine Agreement States eligible for grant assistance, action on the 133 former sites located in these States has been successfully completed. The NRC has been working to bring to closure the remaining four sites identified as contaminated, three of which are located in California and one in Colorado.

On February 1, 2006, the NRC issued Generic Letter 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power. The objective of the generic letter is to request information from nuclear power plant licensees to determine if compliance is being maintained with NRC regulatory requirements governing electric power sources and associated personnel training. The NRC staff is currently evaluating the responses and will report to the Commission on the results by the beginning of June.

On February 9, 2006, the National Academy of Sciences released a report on the transportation of spent nuclear fuel. The report, "Going the Distance? The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States," was released by the National Research Council, part of the National Academies. It was compiled by the Council's Committee on Transportation of Radioactive Waste. The report's principal finding is that there are "no fundamental technical barriers to the safe transport of spent nuclear fuel and high-level radioactive waste in the United States." Shipment of spent fuel by rail or truck is "a low-radiological-risk activity with manageable safety, health, and environmental consequences when conducted in strict adherence to existing regulations." The report also concluded that "the radiological risks associated with the transportation of spent fuel and high-level waste are well understood and are generally low." It attributed this conclusion in part to "rigorous international standards and U.S. regulations for the design, construction, testing, and maintenance of spent fuel packages." The committee recommended that the NRC conduct further research into the health and safety risks of long-duration fires engulfing spent fuel transportation casks. The report also recommended that "full-scale package testing should continue to be used as part of integrated analytical, computer simulation, scale model, and testing programs to validate package performance." This recommendation is also consistent with the goals of the NRC's Package Performance Study, which is now under development.

On February 16, 2006, the NRC announced the public release of its 2005 Safety Culture and Climate Survey results. According to the survey results, the NRC improved in essentially all areas as compared to the 2002 survey, with the largest gains in communication, mission and strategic planning, employee engagement, recruiting, developing and retaining staff, and management leadership. According to the survey, which had an impressive 70 percent response rate, workload and stress continue to be challenges for employees. Better knowledge transfer from staff who are retiring and use of the Differing Professional Opinion program are also areas of opportunity for continued improvement. The survey was conducted by the NRC's Office of Inspector General (OIG) with assistance from a contractor research firm to gain a better understanding of NRC's safety culture and climate. The 2005 survey is the third survey conducted to date; previous surveys were conducted in 1998 and 2002. The NRC is committed to taking additional actions to address the results of the 2005 survey.

As discussed in Section VI of the enclosed report, the NRC, having concluded its environmental and safety reviews and the adjudication of all contested issues, and having taken all other actions necessary for issuance of a license, issued Materials License No. SNM-2513 to Private Fuel Storage, L.L.C. (PFS) by letter dated February 21, 2006. That action constitutes the final agency action with respect to the PFS license application. Because final agency action has been taken on the PFS application, the NRC does not plan to provide future report updates on this topic.

On March 2, 2006, the NRC staff completed its review of the Vermont Yankee (VY) extended power uprate (EPU) application and approved the 20 percent power uprate. The licensee has begun power ascension of VY to the new EPU power level. Specific details on the uprate can be found in Section IX of the enclosed report.

The NRC has completed an Agreement with the State of Minnesota to assume part of the NRC's regulatory authority over certain radioactive materials in the state. The Agreement became effective March 31, 2006. The NRC transferred approximately 150 licenses, most for medical and industrial uses of radioactive material, to Minnesota's jurisdiction. Before approving the agreement, NRC reviewed Minnesota's radiation control program to ensure that it was adequate to protect public health and safety and was compatible with NRC's program for regulating the radioactive materials covered in the agreement. An announcement of the proposed agreement was made in November inviting comments from the public. No comments were received.

Effective April 1, 2006, the NRC has updated its Reactor Oversight Process (ROP) with the introduction of the Mitigating Systems Performance Index (MSPI), which tracks the availability and reliability of systems used to reduce the severity of incidents at a nuclear power plant. The NRC has worked with stakeholders on refining the MSPI through a pilot program since 2002. The development of the new indicator has included multiple public meetings and public comments, as well as input from the Advisory Committee on Reactor Safeguards and other nuclear regulators interested in using similar methods. The NRC and stakeholders have established a risk assessment methodology and have developed software and databases to provide the raw data necessary for evaluating the index.

On April 5, 2006, the NRC staff issued its final environmental impact statement on the proposed Early Site Permit (ESP) for the Grand Gulf site, about 25 miles south of Vicksburg, Mississippi. The report contains the NRC's finding that there are no environmental impacts that would prevent issuing the ESP. The ESP process allows an applicant to address site-related issues, such as environmental impacts, for possible future construction and operation of a nuclear power plant at the site. The Grand Gulf ESP application was filed on October 21, 2003, by System Energy Resources Inc. (SERI), a subsidiary of Entergy Nuclear. If approved, the permit would give SERI up to 20 years to decide whether to build a new nuclear unit on the site and to file an application with the NRC for approval to begin construction. The NRC staff's conclusion is based on its independent review of a report submitted by SERI, taking into account consultations with Federal, State, tribal, and local organizations and consideration of comments received during the public scoping process. Before the Commission can reach a final decision on issuing the permit, the NRC staff must complete revisions to the ESP's safety evaluation report. The Atomic Safety and Licensing Board Panel must also conduct a mandatory hearing on the matter.

I also want to inform you of the agency's progress in implementing the Energy Policy Act of 2005. On January 31, 2006, the NRC issued a Confirmatory Order requiring that backup power be available for the emergency notification system in accordance with Section 651(b). On February 10, 2006, NRC published in the Federal Register (71 FR 7349) its proposed fiscal year (FY) 2006 fee rule (10 CFR Part 170) in accordance with Section 623. On March 1, 2006, the NRC assigned Federal Security Coordinators and alternates in each NRC Region in accordance with Section 651(a)(3). On March 30, 2006, the NRC amended its Memorandum of Understanding with the State Department to cover health services for employees and dependents serving in foreign countries in accordance with Section 651(c)(3).

Please do not hesitate to contact me if I may provide additional information.

Sincerely,

/RA/

Nils J. Diaz

Enclosure:
Quarterly Status Report on the Licensing Activities
and Regulatory Duties of the U.S. NRC, January - March 2006

cc: Senator Thomas R. Carper

Identical letter sent to:

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

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

The Honorable Pete V. Domenici
Chairman, Subcommittee on Energy
and Water Development
Committee on Appropriations
United States Senate
Washington, D.C. 20510
cc: Senator Harry Reid

The Honorable David L. Hobson
Chairman, Subcommittee on Energy
and Water Development
Committee on Appropriations
United States House of Representatives
Washington, D.C. 20515
cc: Representative Peter J. Visclosky

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

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

QUARTERLY STATUS REPORT ON THE
LICENSING ACTIVITIES AND REGULATORY DUTIES OF THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

JANUARY - MARCH 2006

Enclosure

TABLE OF CONTENTS[1]

[1]Note: The period of performance covered by this report includes activities occurring between the first day of January and last day of March 2006. The transmittal letter to Congress accompanying this report may provide more recent information in order to keep Congress fully and currently informed of NRC's licensing and regulatory activities.

I Implementing Risk-Informed Regulations

The U.S. Nuclear Regulatory Commission (NRC) continues to make progress toward risk-informing its regulations for nuclear power reactors. On November 22, 2004, the NRC published a final rule,10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors." This risk-informed regulation establishes an alternate set of requirements incorporating up-to-date analytic tools and risk insights to enhance plant safety by enabling nuclear power plant licensees to determine more precisely the safety significance of reactor systems, structures and components and maintain these structures, systems, and components in a manner commensurate with their safety significance. To ensure the new regulation is properly implemented, the NRC published Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems and Components in Nuclear Power Plants According to Their Safety Significance," for trial use in January 2006. After receiving comments on the Regulatory Guide, the NRC staff began to clarify the guidance. A public meeting is planned for April 19, 2006, to discuss these revisions.

Risk-informed requirements for emergency core cooling systems are also being developed. The NRC published a proposed rule for risk-informing these requirements on November 7, 2005, with a 90-day public comment period. In response to a request from several industry groups, the NRC extended the comment period by 30 days to March 8, 2006. The NRC is now evaluating public comments and developing the final rule.

Broad efforts to transform the overall deterministic structure of NRC regulations into a new format based on the use of risk information are also in progress. Since 2003, the NRC has been working on a regulatory structure for new plant licensing that would result in risk-informed, technology-neutral regulations for licensing future nuclear power reactor designs. The NRC is also investigating whether this risk-informed, technology-neutral regulatory structure should apply or be available to risk-inform the current regulations on light water reactors (LWRs) in 10 CFR Part 50. A March 22, 2006 Commission directive instructed the staff to prepare an advance notice of proposed rulemaking seeking public input on ways to make the technical requirements for power reactors more risk-informed and performance-based. The notice will solicit public feedback on whether the focus should be on "technology-specific frameworks" for non-LWRs, whether development of a technology-neutral licensing framework is "premature," and how to prioritize rulemaking for various non-LWR technologies.

II Reactor Oversight Process

The NRC continues to implement the Reactor Oversight Process (ROP) at all nuclear power plants. The NRC continues to meet with interested stakeholders on a periodic basis to collect feedback on the effectiveness of the process and to consider feedback for future ROP refinements. Recent activities include the following:

- The staff hosted monthly Mitigating Systems Performance Index (MSPI) public meetings on January 25, February 22, and March 22, 2006. Meeting attendees discussed MSPI guidance clarifications and revisions, resolution of several open technical issues, and a process for conducting and resolving MSPI component outliers and generic issues. Attendees also discussed a schedule and timeline for completing the remaining milestones and activities before the scheduled April 1, 2006 implementation date of the MSPI.

- The staff hosted monthly ROP public meetings on January 26, February 23, and March 23, 2006. The meeting attendees discussed the ROP cross-cutting issues, the safety culture initiative, the significance determination process timeliness improvements, the performance indicator (PIs) improvements, and the open/new frequently asked questions on the PIs.

- The staff incorporated the recommended staff actions regarding agency guidance in the areas of Safety Conscious Work Environment and Safety Culture into NRC inspection procedures on March 24, 2006. The inspection procedures were sent out to NRC's regional offices for comments in accordance with the review process IMC 0040, "Preparing, Revising, and Issuing Documents for the NRC Inspection Manual."

- During the week of February 6, the staff participated in the NRC regional offices' end-of-cycle review meetings. The licensee's performance at each reactor site was assessed by utilizing the most recent quarterly performance indicators and inspection findings compiled over the previous twelve months. The output of these meetings was an end-of-cycle letter that communicates to the licensee which column of the Action Matrix the licensee is in during the assessment period, any substantive cross-cutting issues, and the inspection plan consisting of approximately 18 months of inspection activities.

III Status of Issues in the Reactor Generic Issue Program

On January 20, 2006, the NRC issued Generic Letter (GL) 2006-01, "Steam Generator Tube Integrity and Associated Technical Specifications," to all holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from their reactor vessels. The letter was issued because of the NRC concern that current Technical Specifications (TS) requirements may not be sufficient to ensure that steam generator tube integrity can be maintained in accordance with the current licensing and design basis. The Generic Letter requested that the affected plants either submit a description of their program for ensuring steam generator tube integrity for the interval between inspections or adopt alternative TS requirements for ensuring steam generator tube integrity. (Alternative TS requirements that address NRC concerns about the existing TS were previously developed by the industry and found acceptable by the NRC).

On January 17, 2006, the NRC issued Information Notice (IN) 2005-25, Supplement 1, "Additional Results of Chemical Effects Tests in a Simulated PWR Sump Pool Environment," to all holders of operating licenses for PWRs, except those who have permanently ceased operations and have certified that fuel has been permanently removed from their reactor vessels. The Supplement was issued to inform the affected licensees of recent NRC-sponsored research results related to chemical effects in a simulated PWR sump pool environment. It specifically provided information regarding test results related to chemical effects in environments containing dissolved phosphate and dissolved calcium.

IV Licensing Actions and Other Licensing Tasks

Operating power reactor licensing actions are defined as orders, license amendments, exemptions from regulations, relief from inspection or surveillance requirements, topical reports submitted on a plant-specific basis, notices of enforcement discretion, or other actions requiring NRC review and approval before they can be implemented by licensees. The fiscal year (FY) 2006 NRC Performance Plan incorporates two output measures related to licensing actions -- number of licensing actions completed per year and age of the licensing action inventory.

Other licensing tasks for operating power reactors are defined as licensee responses to NRC requests for information through generic letters or bulletins, NRC responses to 10 CFR 2.206 petitions, NRC review of generic topical reports, responses by the Office of Nuclear Reactor Regulation to regional office requests for assistance, NRC review of licensee 10 CFR 50.59 analyses and final safety analysis report updates, or other licensee requests not requiring NRC review and approval before they can be implemented by licensees. The FY 2006 NRC Performance Plan incorporates one output measure related to other licensing tasks -- the number of other licensing tasks completed.

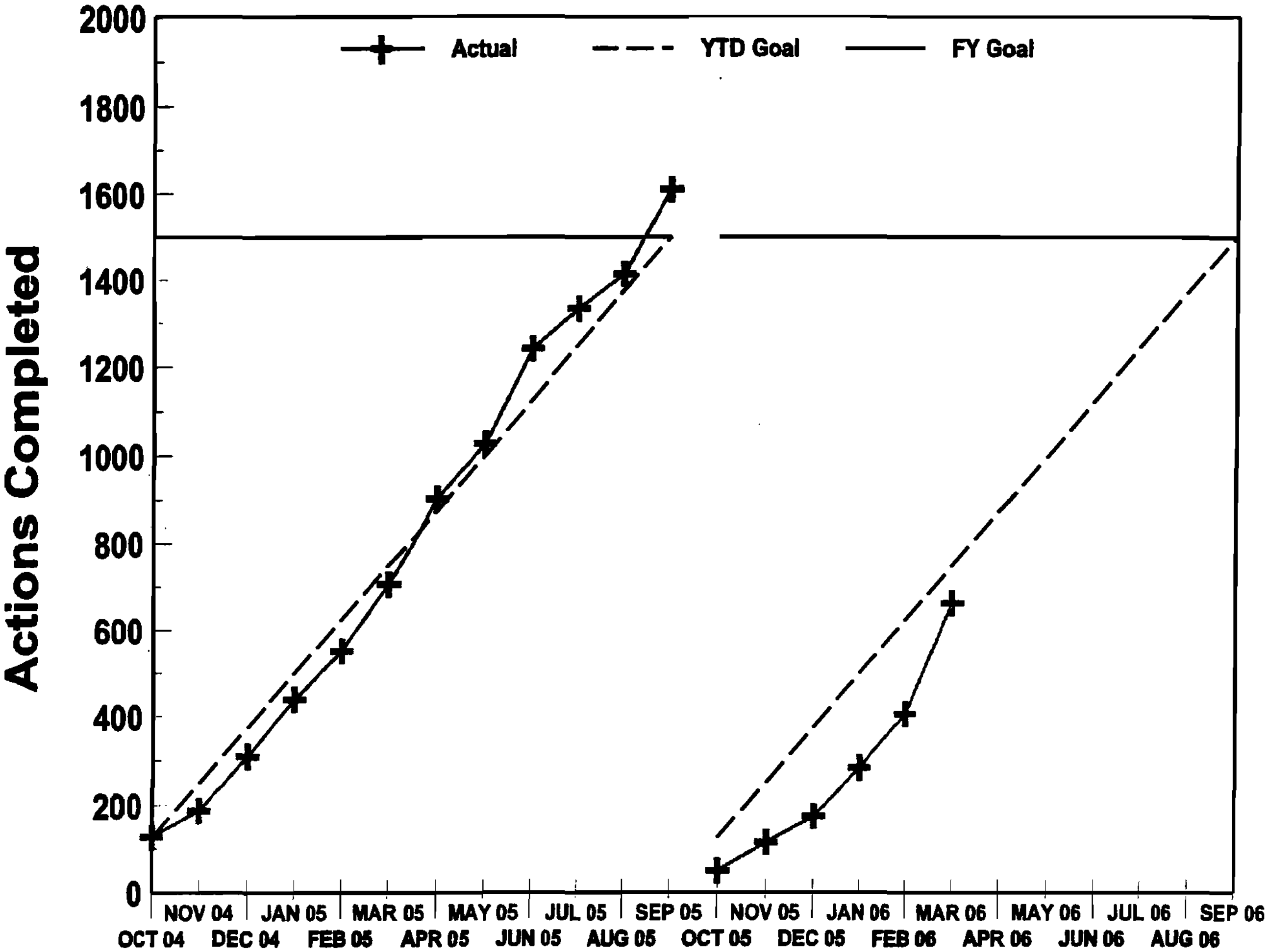
The actual FY 2004 and FY 2005 results, the FY 2006 goals, and the actual FY 2006 results for the three NRC Performance Plan output measures for operating power reactor licensing actions and other licensing tasks are shown in the following table.

PERFORMANCE PLAN				
Output Measure	**FY 2004 Actual**	**FY 2005 Actual**	**FY 2006 Goals**	**FY 2006 Actual (thru 03/31/2006)**
Licensing actions completed/year	1741	1609	≥ 1500	661
Age of licensing action inventory	91% ≤ 1 year; and 100% ≤ 2 years	92.6%≤ 1 year; and 99.9% ≤ 2 years	96% ≤ 1 year and 100% ≤ 2 years old	82.6%≤ 1 year and 99.4% ≤ 2 years
Other licensing tasks completed/year	671	715	≥ 500	400

The charts on the following pages show NRC's FY 2006 trends for the three operating power reactor licensing action and other licensing task output measure goals. The completion of licensing actions does not typically follow a straight line trend due to the inherent variability associated with the level of effort needed to complete individual licensing actions. For FY 2006, the value of completed licensing actions identifies a slight decrease relative to the value completed at this time in FY 2005. The increase identified in completed licensing actions in the second quarter of FY 2006 is attributable to increased management attention to avert an adverse trend.

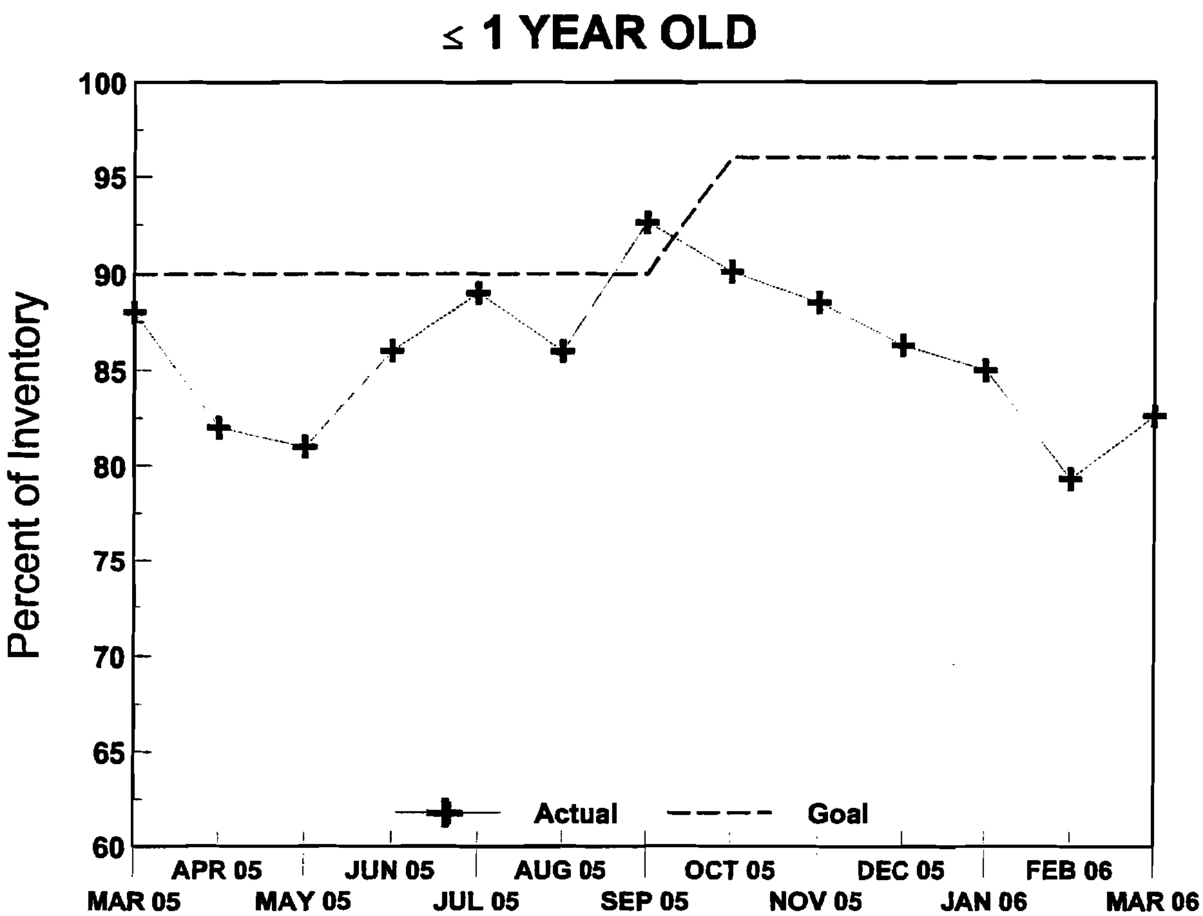
Nuclear Reactor Safety - Reactor Licensing

Performance Plan Target: Completed Licensing Actions



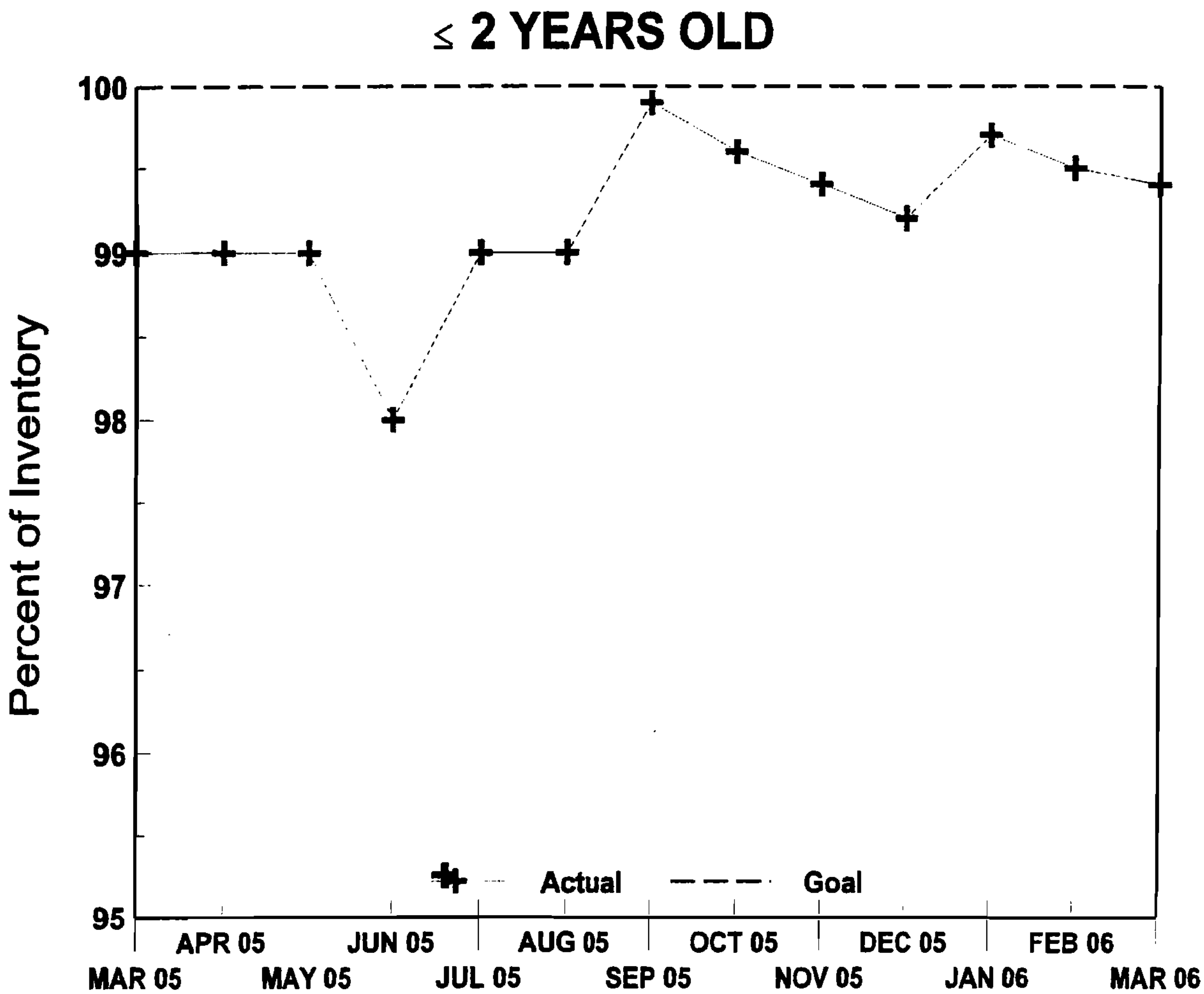
Nuclear Reactor Safety - Reactor Licensing

Performance Plan Target: Age of Licensing Action Inventory



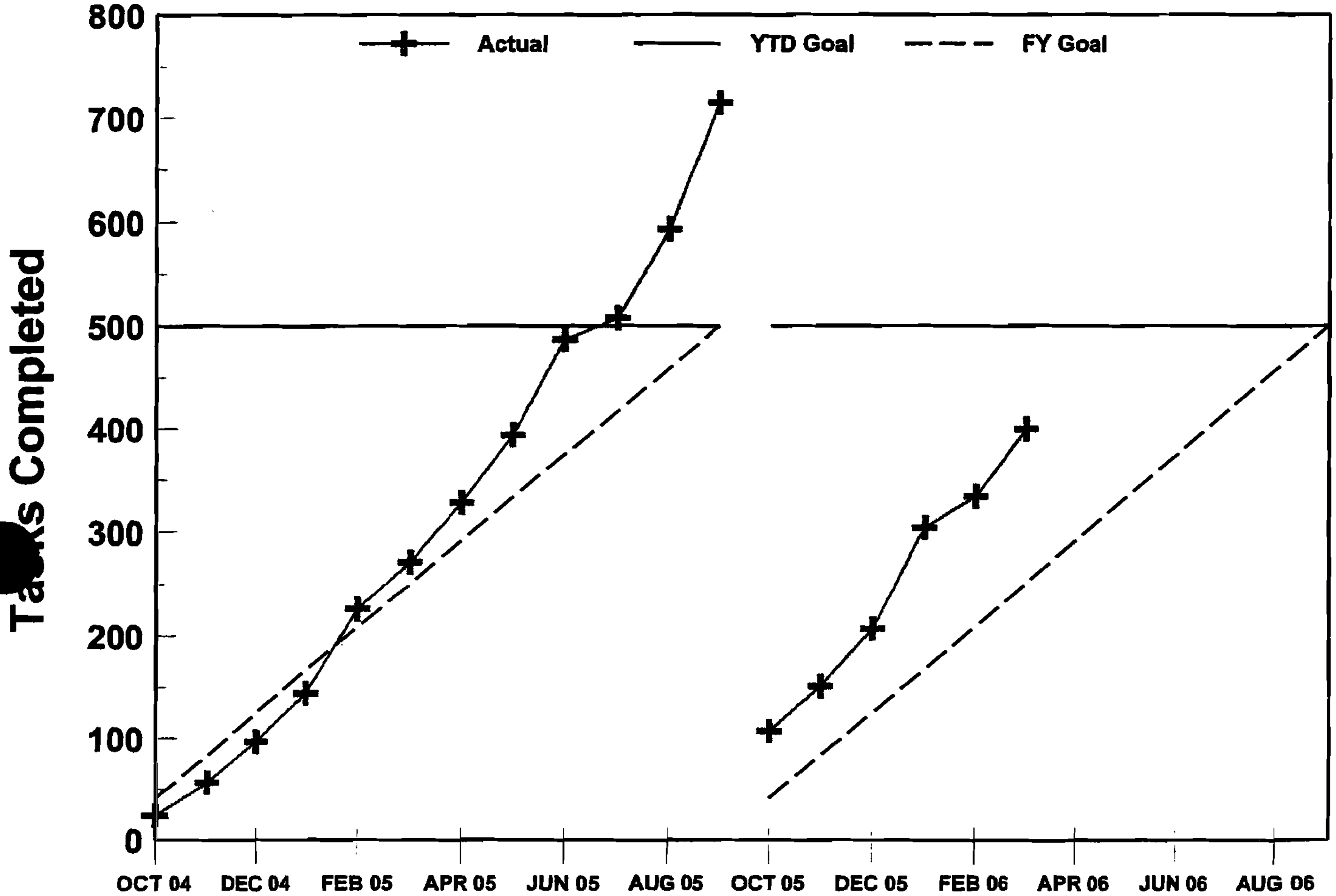
Nuclear Reactor Safety - Reactor Licensing

Performance Plan Target: Age of Licensing Action Inventory



Nuclear Reactor Safety - Reactor Licensing

Performance Plan Target: Completed Other Licensing Tasks



V Status of License Renewal Activities

The NRC has completed the review of license renewal applications for 39 of the 104 units licensed to operate. The extension of the licenses for these 39 units results in approximately 34 gigawatts-electric maximum dependable capacity remaining available for an additional 20 years past the initial license expiration dates.

Browns Ferry, Units 1, 2, and 3, License Renewal Application

The staff issued the final supplemental environmental impact statement (SEIS) in June 2005 and the final safety evaluation report (SER) in January 2006. A supplement to the SER is scheduled to be issued in April 2006. A decision on whether to issue the renewed licenses is scheduled for May 2006.

Nine Mile Point, Units 1 and 2, License Renewal Application

The staff is addressing the comments received on the draft SEIS and anticipates issuing the final SEIS in May 2006. The draft SER, identifying any remaining open items, was issued in March 2006, and the applicant's responses to the open items are due in April 2006.

Brunswick, Units 1 and 2, License Renewal Application

The staff is addressing comments received on the draft SEIS and anticipates issuing the final SEIS in April 2006. The initial draft SER was issued in December 2005, and the licensee's comments were received in January 2006. The final SER is scheduled to be issued in April 2006. A decision on the renewed licenses is scheduled for June 2006.

Monticello License Renewal Application

The draft SEIS was issued in January 2006, and the draft SER, identifying any remaining open items, is scheduled to be issued in April 2006. A request for hearing was received in response to the NRC's notice of opportunity for hearing, and an Atomic Safety and Licensing Board (ASLB) was established. The proceeding was terminated by the ASLB for lack of standing by the petitioners and inadmissable contentions. A subsequent appeal to the Commission was rejected.

Palisades License Renewal Application

The draft SEIS was issued in February 2006, and the draft SER, identifying any remaining open items, is scheduled to be issued in June 2006. A request for hearing was received in response to the NRC's notice of opportunity for hearing, and an ASLB was established. The ASLB determined that the petitioner did not submit an admissible contention and terminated the proceeding. The petitioner has appealed the ASLB's decision to the Commission.

Oyster Creek License Renewal Application

The Oyster Creek license renewal application is currently under review, and the staff is preparing requests for additional information and reviewing the licensee's responses. The draft

SEIS is scheduled to be issued in June 2006, and the draft SER, identifying any remaining open items, is scheduled to be issued in August 2006. A request for hearing was received in response to the NRC's notice of opportunity for hearing, and an ASLB was established. The Board has admitted one contention, and the hearing process is proceeding.

Pilgrim License Renewal Application

On January 27, 2006, the NRC received an application for renewal of the operating license for Pilgrim Nuclear Power Station. The staff has completed its acceptance review and has found the application acceptable for docketing and review. Until it is determined whether a hearing will be conducted, a 30-month review schedule has been established with a final decision on issuance of the renewed license scheduled for July 2008.

Vermont Yankee License Renewal Application

On January 27, 2006, the NRC received an application for renewal of the operating license for Vermont Yankee Nuclear Power Station. The staff has completed its acceptance review and has found the application acceptable for docketing and review. Until it is determined whether a hearing will be conducted, a 30-month review schedule has been established with a final decision on issuance of the renewed license scheduled for July 2008.

VI Status of Review of Private Fuel Storage, Limited Liability Corporation's Application for a License to Operate an Independent Spent Fuel Storage Installation on the Reservation of the Skull Valley Band of Goshute Indians

This proceeding involved an application from Private Fuel Storage, L.L.C. (PFS) to construct and operate an independent spent fuel storage installation on the reservation of the Skull Valley Band of Goshute Indians in Skull Valley, Utah. On September 9, 2005, the Commission issued a Memorandum and Order, CLI-05-19, in which it (a) denied the State of Utah's petition for review of ASLB's February 24, 2005, Final Partial Initial Decision and other decisions on aircraft crash issues, and (b) authorized the NRC staff, upon making the requisite findings on all non-contested issues, to issue a license to PFS to construct and operate its proposed facility.

On November 3, 2005, the State of Utah filed a motion with the Commission to reopen the record and to amend late-filed Contention Utah UU, based upon recent statements by officials within the U.S. Department of Energy (DOE) concerning DOE's current intention to accept spent fuel in multipurpose canisters at the proposed Yucca Mountain repository. On January 31, 2006, the Commission issued a Memorandum and Order, CLI-06-03, denying the State's motion in its entirety.

The NRC, the Bureau of Land Management (BLM), the Bureau of Indian Affairs, and the Surface Transportation Board have worked together to fulfill each agency's National Historic Preservation Act (NHPA) Section 106 obligations, leading to the development of a Memorandum of Agreement (MOA) for the protection of historic and cultural resources, and draft treatment and discovery plans to ensure the mitigation of any adverse impact to such resources. All necessary parties have signed the MOA, with the exception of BLM and the Utah State Historic Preservation Officer, who have declined to sign the MOA at this or any time in the foreseeable future. Accordingly, the NRC, by letter dated November 22, 2005, notified the

Advisory Council on Historic Preservation (ACHP) that NRC planned to terminate the Section 106 consultation process, pursuant to 36 C.F.R. § 800.7, and requested comments by the ACHP on such termination. By letter dated January 9, 2006, the ACHP provided its comments; therein, the ACHP concluded, *inter alia*, that the NRC's plan to include a condition in the PFS license to require implementation of the substantive provisions of the MOA constitutes a reasonable and appropriate means of concluding the NRC's responsibilities under the NHPA. In accordance with ACHP regulations, the NRC, by letter dated February 10, 2006, responded to the ACHP comments.

Having concluded its environmental and safety reviews and the adjudication of all contested issues, and having taken all other actions necessary for issuance of a license, the NRC, by letter dated February 21, 2006, issued Materials License No. SNM-2513 to PFS. That action constitutes the final agency action with respect to the PFS license application. Petitions for review of the NRC's issuance of the PFS license have been filed by the State of Utah and another Intervenor before the U.S. Court of Appeals for the District of Columbia Circuit.

Because final agency action has been taken on the PFS application, the NRC does not plan to provide future report updates on this topic.

VII Enforcement Process and Summary of Reactor Enforcement by Region

Reactor Enforcement by Region

Reactor Enforcement Actions						
		Region I	Region II	Region III	Region IV	TOTAL
Severity Level I	Quarter 2 FY 06	0	0	0	0	0
	FY 06 YTD Total	0	0	0	0	0
	FY 05 Total	0	0	2	0	2
	FY 04 Total	0	0	0	0	0
Severity Level II	Quarter 2 FY 06	0	0	0	0	0
	FY 06 YTD Total	0	0	0	0	0
	FY 05 Total	0	1[2]	2	0	3
	FY 04 Total	0	1	0	0	1
Severity Level III	Quarter 2 FY 06	0	0	1	0	1
	FY 06 YTD Total	0	0	4	0	4

[2]The FY 05 Total for Region II and the overall FY 05 Total were both increased by one to reflect a correction for a violation associated with a Severity Level II violation issued during July 2005. The violation and its associated finding will not be described because the issue is security related. This error was identified during an internal audit.

Reactor Enforcement Actions

	FY 05 Total	2	1	3	2	8
	FY 04 Total	1	2	4	0	7
Cited Severity Level IV or GREEN	Quarter 2 FY 06	3	0	1	1	5
	FY 06 YTD Total	3	0	1	1	5
	FY 05 Total	6	0	4	0	10
	FY 04 Total	1	0	2	3	6
Non-Cited Severity Level IV or GREEN	Quarter 2 FY 06	58	24	40	72	194
	FY 06 YTD Total	102[3]	58	120	127	407
	FY 05 Total	239	197	300	282	1018
	FY 04 Total	271	175	290	301	1037

* Numbers of violations are based on enforcement action tracking system data that may be subject to minor changes following verification. The numbers shown as Severity Level I, II, III or IV refer to the number of Severity Level I, II, III, and IV violations or problems. The monthly totals generally lag by 30 days due to inspection report and enforcement development.

Escalated Reactor Enforcement Actions
Associated with the Reactor Oversight Process

		Region I	Region II	Region III	Region IV	TOTAL
Notices of Violation Related to RED, YELLOW, or WHITE Findings	Quarter 2 FY 06 RED	0	0	0	0	0
	Quarter 2 FY 06 YELLOW	0	0	0	0	0
	Quarter 2 FY 06 WHITE	1	0	0	1[4]	2
	FY 06 YTD Total	1	0	2	1	4

[3]The FY 06 YTD Total for Region I and the overall FY 06 YTD Total were increased by two to reflect a correction in the December 2005 non-cited violation data.

[4]The violation and its associated finding will not be described because the issue is security related.

Escalated Reactor Enforcement Actions Associated with the Reactor Oversight Process					
	Region I	Region II	Region III	Region IV	TOTAL
FY 05 Total	5	5[5]	8[6]	2[7]	20
FY 04 Total	3	4	7	6	20

Description of Significant Actions Taken During the Second Quarter of FY 06

AmerGen Energy Company, LLC (Oyster Creek Generating Station) EA-05-199 – On January 9, 2006, a Notice of Violation was issued for a violation of 10 CFR 50.54(q), 10 CFR 50.47(b)(4), and the Oyster Creek Generating Station Emergency Plan. This violation was associated with a WHITE significance determination process (SDP) finding involving the licensee's failure to utilize properly the Emergency Plan emergency action level (EAL) matrix during an actual event. Specifically, operators did not recognize that plant parameters met the EAL thresholds for declaring an Unusual Event and a subsequent Alert. Since an Alert was not declared, licensee personnel did not activate their emergency response organization to assist operators in mitigating the event. Additionally, State and local agencies, which rely on information provided by the facility licensee, might not have been able to take initial response measures in as timely a manner had the event degraded further.

Entergy Nuclear Operations, Inc. (Indian Point Units 2 and 3) EA-05-190 – On January 31, 2006, an immediately effective Confirmatory Order Modifying License was issued to Entergy Nuclear Operations, Inc., Indian Point Units 2 and 3. The licensee consented to modifying its operating licenses for Indian Point Units 2 and 3 to meet the criteria in Section 651(b) of the Energy Policy Act of 2005 that directs the Commission to require that backup power is to be available for the emergency notification system of a power plant, including the emergency siren warning system, if the alternating current within the 10-mile emergency planning zone of the power plant is lost.

Exelon Generation Company, LLC (LaSalle County Station) EA-06-022 – On March 31, 2006, a Notice of Violation was issued to Exelon for a willful Severity Level III violation involving three

[5] The FY 05 Total for Region II and the overall FY 05 Total were both increased by one to reflect a correction for a violation associated with a WHITE SDP finding issued during December 2004. The violation and its associated finding will not be described because the issue is security related. This error was identified during an internal audit.

[6] The FY 05 Total for Region III and the overall FY 05 Total were both increased by three to reflect a correction for three violations associated with a previously issued RED SDP finding. A description of the violations was included the April 2005 Congressional Report, but the April 2005 totals were not updated. This error was identified during an internal audit.

[7] The FY 05 Total for Region IV and the overall FY 05 Total were both increased by one to reflect a correction for a violation associated with a WHITE finding issued on April 15, 2005. A description of this event is also included the Addition to Description of Significant Actions Taken During April 2005 section. This error was identified during an internal audit.

contract employees who violated radiation protection procedures associated with entry into high radiation areas.

Addition to Description of Significant Actions Taken During April 2005[8]

Omaha Public Power District (Fort Calhoun Station) EA-05-038 – On April 15, 2005, a Notice of Violation was issued for a violation associated with a WHITE SDP finding involving the licensee's failure to identify and correct a failed fuse during emergency diesel (EDG) generator surveillance testing, which resulted in the EDG being inoperable for 29 days. The associated violation was cited against 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," because the licensee failed to identify and correct the issue associated with the failed fuse, which resulted in the EDG being inoperable for a period of time longer than allowed by the plant's technical specifications.

VIII Power Reactor Security Regulations

In response to the terrorist attacks on September 11, 2001, the NRC and the nuclear industry have taken many actions to ensure the security at nuclear power plants. A series of Advisories, Orders, and Regulatory Issue Summaries have been and, as needed, continue to be issued to strengthen further the security of NRC-licensed facilities and control of nuclear materials.

The NRC is codifying through rulemaking the actions taken to enhance security of NRC power reactor licensees. The public comment period for a proposed rule on fitness-for-duty (10 CFR Part 26), including both drug/alcohol testing and fatigue-related provisions, ended on December 27, 2005. This rulemaking will update the drug and alcohol testing provisions and establish enforceable requirements of the management of worker fatigue. The public comment period for a proposed rule on the Design Basis Threat (DBT) (10 CFR 73.1) ended on January 23, 2006. The DBT rulemaking specifies the adversary characteristics that nuclear power plants and certain related facilities must be able to defend against with high assurance and would amend the NRC's regulations to include, among other things, the supplemental security requirements previously imposed by the Commission's DBT Orders of April 29, 2003. This rulemaking is also addressing specific threat attributes identified in Section 651 of the Energy Policy Act of 2005. Also currently under development is a comprehensive proposed rule on Requirements for Physical Protection (10 CFR 73.55) incorporating safety/security interface requirements that will be published for public comment later this year.

The NRC is now conducting full force-on-force exercises at each site on a normal, three-year cycle using the expanded adversary characteristics that were developed as a result of the increased post 9/11 threat. The purpose of the force-on-force exercises is to assess and improve, as necessary, performance of defensive strategies at licensed facilities. The NRC retains responsibility for establishing exercise scenarios, oversight of the mock adversary force, and evaluation of licensee performance. Measures have been established to minimize any possibility of a conflict of interest between the mock adversary force and the licensees'

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[8]This event description was added in order to reflect a correction in the April 2005 data. The FY 05 Total for Region IV and the overall FY 05 Total were also increased by one in order to reflect the same correction. This error was identified during an internal audit.

responsibilities for physical protection. To date, mock adversary force personnel have performed adequately in the force-on-force exercises in which they have participated.

In February 2006, NRC staff participated in an industry-sponsored workshop on force-on-force security that provided opportunities for members of industry and government to discuss force-on-force exercise processes and other security initiatives affecting licensees. The NRC staff also made a presentation on the Joint Conflict and Tactical Simulation (JCATS) system, including a simulation exercise, and requested voluntary participants for future JCATS activities.

NRC has established a review team to evaluate the Remotely Operated Weapons System (ROWS) deployed at one power reactor site. This is the first application of ROWS technology at a power reactor. The licensee has submitted a revised security plan that incorporates the ROWS, which offers a response capability at reduced cost, into the site protective strategy. This is a first-of-a-kind effort, and the NRC review team is developing a standard review plan to be used to evaluate the licensee's submittal and any similar future requests.

The NRC continues to support the U.S. Department of Homeland Security (DHS) / Homeland Security Council (HSC) initiative to enhance integrated response planning for power reactor facilities. The staff is continuing to work with HSC, DHS, the Federal Bureau of Investigation (FBI), and others to develop plans to address recommended actions. Working closely with licensees and DHS, the staff also developed Emergency Action Levels specifically for events involving credible imminent threats. An emergency preparedness, industry-identified, frequently-asked questions (FAQ) process was implemented in September 2005, and in January 2006, NRC held the initial public meeting with industry representatives to discuss FAQs and proposed resolutions dealing with EAL guidance. In February 2006, NRC issued the summary and analysis of more than 700 comments received during the August 31 - September 1, 2005 emergency preparedness public meeting held to obtain stakeholder input to enhance emergency preparedness regulations and guidance.

In December 2005, the NRC designated Regional Federal Security Coordinators (primary and alternate) in each of the NRC Regional Offices. Their responsibilities are delineated in Section 651 of the Energy Policy Act of 2005. NRC staff will assess effectiveness after one year.

On January 24, 2006, the NRC conducted a successful tabletop exercise at DHS headquarters with representatives from DHS, the Department of Defense, and the FBI. The tabletop focused on the interrelationships between NRC and DHS, consistent with the National Response Plan and annexes, in responding to incidents at nuclear power plants. The interactive discussion among participants resulted in reconfirmation of the respective responsibilities of the NRC and DHS for nuclear plant incidents. A follow-on, NRC-sponsored, interagency tabletop exercise, focused on a terrorist aircraft attack on a nuclear power plant, was conducted at NRC headquarters on March 16, 2006.

The NRC has completed the site-specific spent fuel pool assessments that were begun on July 5, 2005, and issued the last of the assessment reports on December 16, 2005. NRC conducted these assessments to identify additional mitigation strategies to enhance the spent fuel pool cooling safety function under severe circumstances challenging the functional capabilities of the plant. In January 2006, the industry responded with generic strategies that could be used at all plants. The NRC staff is evaluating safety benefit of the proposed strategies. In addition, the NRC has completed structural analysis of one spent fuel pool and is continuing with the structural analysis of an additional pool to provide further insight into spent fuel pool structural safety margin. The remaining analysis will be completed in May 2006.

IX Power Uprates

There are three types of power uprates. A measurement uncertainty recapture (MUR) power uprate is a power uprate of less than 2 percent and is based on the use of more accurate feedwater flow measurement techniques. Stretch power uprates (SPUs) are power uprates that are typically on the order of less than 7 percent and are within the design capacity of the plant. SPUs require only minor plant modification. Extended power uprates (EPUs) are power uprates beyond the design capacity of the plant and, thus, require major plant modification.

Licensees have been applying for and implementing power uprates since the 1970s as a way to increase the power output of their plants. The NRC staff has been conducting power uprate reviews since then and has completed 108 such reviews to date. Approximately 13,797 megawatts-thermal (MWt) or 4,599 megawatts-electric (MWe) to the Nation's electric generating capacity or an equivalent of about 4.6 nuclear power plant units has been gained through implementation of power uprates at existing plants. The NRC staff currently has 10 plant-specific power uprate applications under review. The ten applications under review are for four MUR power uprates and six EPUs.

On March 2, 2006, the NRC staff completed its review of the Vermont Yankee (VY) EPU application and approved the 20 percent power uprate. Regarding litigation issues, the Atomic Safety and Licensing Board is expected to establish the final hearing schedule in the near future. Regarding the power ascension of VY to the new EPU power level, VY suspended the power ascension process after the first 5 percent increase in power on March 5, 2006, when certain plant data reached an administrative limit specified in the VY steam dryer monitoring plan. VY remained at the 105 percent power level until March 31, 2006, when the NRC headquarters staff completed its review of the licensee's engineering evaluation, which justified further power ascension. As documented in the NRC staff's Safety Evaluation for the EPU, the licensee has formally committed not to increase power above the applicable hold point if any safety concerns are identified during the NRC staff review of the power ascension data. The power level of VY as of April 3, 2006, is 110 percent of the previous licensed thermal power.

On February 10, 2006, the Hope Creek licensee withdrew its EPU application. The NRC allowed the licensee to withdraw the application because it was incomplete.

Regarding the Calvert Cliffs 1 & 2 and Fort Calhoun MUR power uprates, which were submitted on January 31 and March 31, 2005, respectively, the NRC did not complete the reviews within six months, which is the timeliness goal for MUR power uprates that are based on the use of NRC-approved methodologies for feedwater flow measurement. The scheduled reviews have been extended because the licensees chose not to use NRC-approved methodologies.

In March 2006, the NRC staff surveyed licensees to obtain information on whether they plan to submit power uprate applications over the next 5 years. Based on this survey, licensees plan to request power uprates for 23 nuclear power plant units over the next 5 years. If approved, these power uprates will result in an increase of about 3,795 MWt or approximately 1,265 MWe.

X New Reactor Licensing

The NRC expects to license the next generation of nuclear power plants using Part 52 to Title 10 of the *Code of Federal Regulations*(10 CFR Part 52). 10 CFR Part 52 governs the issuance of standard design certifications, early site permits (ESPs), and combined licenses (COLs) for nuclear power plants.

Design Certifications and Pre-Application Meetings

On December 30, 2005, the Commission approved the final design certification rule for the Westinghouse AP1000 standard plant design. On January 27, 2006, the AP1000 final design certification rule was issued in the *Federal Register* (71 FR 4464). This final rule amends 10 CFR Part 52 to certify the AP1000 standard plant design. Applicants or licensees intending to construct and operate an AP1000 design may do so by referencing the AP1000 design certification rule. A revised final design approval based on Revision 15 of Westinghouse's design control document was issued on March 10, 2006. The certification was the fourth issued under Part 52 and is valid for 15 years.

On August 24, 2005, General Electric (GE) submitted its design certification application for the Economic Simplified Boiling Water Reactor (ESBWR) design. By letter dated December 1, 2005, the NRC staff informed GE that the ESBWR design certification application, as supplemented by GE on October 24, 2005, was sufficiently complete to be accepted formally as a docketed application for design certification. The NRC staff also informed GE that a schedule had been established for the design certification review. Based on GE's commitments to provide additional supporting information, a milestone of October 11, 2007, was established for issuance of the SER with open items. Based on experience with previous design certifications, a 15 month period is assumed for closure of the open items and issuance of the final design approval, and a 12 month period is assumed for the design certification rulemaking. In a letter to GE dated January 5, 2006, the staff emphasized the importance of the Request For Additional Information process and the need to provide timely responses to ensure that schedules would not be adversely impacted.

On March 23, 2006, the staff briefed Senate Energy and Natural Resource Committee staff members on the ESBWR design certification review project. This briefing also covered infrastructure development efforts, including the COL Application Regulatory Guide development, the Standard Review Plan update, and the Part 52 rulemaking.

On January 10, 2006, the NRC staff met with representatives of Framatome ANP (FANP) to discuss the pre-application review for the Evolutionary Power Reactor (EPR). FANP plans to submit three topical reports over the next several months and also discussed a proposal for early submittal of information during the pre-application review period to facilitate early review, resolution of issues, and NRC approval. FANP also described topics that it believes would benefit from the application of the Multinational Design Approval Program. On February 23, 2006, the staff met with FANP regarding possible design acceptance criteria (DAC) for the EPR design. FANP stated that its goal is to set a high threshold for use of DAC for the EPR design certification and proposes to submit design process descriptions for piping, instrumentation and controls, and human factors in the third quarter of this calendar year. NRC review of these submittals would yield a defined level of design completion and detail required to close out design issues during the design certification review, with the intent of minimizing or eliminating

DAC in the final design control document to be cited in the certification. Framatome plans to provide a letter to NRC describing its proposal.

Pebble-Bed Modular Reactor (PBMR) (Pty) Ltd. continues to engage the NRC staff in planning discussions to prepare for the pre-application review of the PBMR design. PBMR (Pty) Ltd. intends to pursue a design certification under 10 CFR Part 52. The company has also stated that it intends eventually to seek deployment of the PBMR in the U.S. PBMR (Pty) Ltd. expects to submit detailed white papers on a number of technical topics and support the submittals with educational sessions and topical workshops for the NRC staff. PBMR (Pty) Ltd.'s most recent schedule projections show the pre-application phase to extend to the end of 2007 or early 2008, followed by submission of a design certification application in 2008. On February 28 - March 2 and March 15 -16, 2006, PBMR (Pty) Ltd. representatives met with the NRC staff for familiarization sessions on plant layout and systems, safety design and analysis, and plant operations and events for the PBMR reactor.

Early Site Permits

The staff is currently reviewing three ESP applications. Dominion Nuclear North Anna, LLC (Dominion) submitted an ESP application in September 2003 for its North Anna site, located in Louisa County, Virginia. The final SER for the North Anna ESP was issued on June 16, 2005. On October 25, 2005, Dominion notified the staff that it was changing the design of the cooling system for proposed Unit 3 from a once-through cooling system to a closed cooling system. The change was made to address the water usage concerns expressed by the Commonwealth of Virginia and local citizens. The change requires revisions to the application, the Environmental Impact Statement (EIS), and the final SER. On January 13, 2006, Dominion Nuclear North Anna LLC submitted a stand-alone supplement to the North Anna ESP application to address the safety and the environmental changes in the application resulting from a modified approach to the proposed Unit 3 cooling. On February 10, 2006, the staff issued a letter to Dominion identifying key areas in which the supplement is deficient and requested the applicant to provide a complete and comprehensive revised ESP application adequately addressing the deficiencies. Also, in the letter, the staff provided an updated schedule for the supplemental final SER and EIS to be issued. On February 22, 2006, the NRC staff briefed the Senate Committee on Energy and Natural Resources staff regarding the North Anna ESP application review. The NRC staff discussed the status of its review of the recent design change initiated by Dominion and the key areas in which additional information is needed. The NRC staff held a public meeting with Dominion on March 10, 2006, to discuss the North Anna ESP supplemental submittal.

In September 2003, Exelon Generation Company, LLC submitted an ESP application for its Clinton site, located in Harp Township, DeWitt County, Illinois. The NRC staff issued the draft SER for the Exelon ESP application for the Clinton site on February 10, 2005. The staff issued the supplemental draft SER with open items on August 26, 2005. On February 17, 2006, the staff issued its final SER for the Clinton ESP application.

System Energy Resources Inc. (SERI) submitted an ESP application in October 2003 for its Grand Gulf site, located in Claiborne County, Mississippi. On October 21, 2005, the staff issued the final SER for the Grand Gulf ESP application. On December 23, 2005, the ACRS issued its final letter on the Grand Gulf ESP final SER, and on February 7, 2006, the staff sent a letter to SERI requesting that the applicant provide a supplement to the application further addressing potential hazards along the Mississippi River. On March 1, 2006, the staff received SERI's supplemental information. The staff is reviewing this information and will revise the SER as necessary.

All three applications require an EIS. The North Anna draft EIS was issued on December 10, 2004; the Clinton draft EIS was issued on March 2, 2005; and the Grand Gulf draft EIS was issued on April 21, 2005. The staff is scheduled to issue the final EIS in for the Grand Gulf site in April 2006 and for the Clinton site in July 2006.

Combined License

On August 17, 2005, Southern Nuclear Operating Company notified the NRC staff that Georgia Power Company had directed them to pursue an ESP/COL at the Vogtle Electric Generating Plant site, located near Waynesboro, Georgia. Southern is scheduled to submit an ESP application in August 2006 and a COL application in March 2008. On January 27, 2006, Southern announced that it will pursue the Westinghouse AP1000 as the reactor technology for potential new nuclear units at the Vogtle site. On March 20 - 22, 2006, the staff toured Southern's Vogtle and Hatch sites in support of the Vogtle ESP application.

AREVA and Constellation Energy announced on September 15, 2005, the formation of UniStar Nuclear. This joint enterprise is intended to provide a single source for design, construction, and operation of new nuclear plants. UniStar Nuclear will market the EPR design. AREVA and Constellation each own half of Unistar. By letter dated November 4, 2005, Constellation Energy and Framatome notified the NRC staff that an application for certification of the EPR is planned at the end of 2007, with a COL application referencing the EPR design following about 6 months later. An additional COL application is planned about a year later. On January 25, 2006, the NRC staff met with representatives of UniStar/Constellation to discuss pre-application activities for a potential COL application. UniStar/Constellation discussed potential schedules for early submittals of information necessary to obtain approval from the NRC for limited work authorizations. UniStar/Constellation also stated that it is scheduling to begin site characterization activities at Calvert Cliffs, which is one of several potential UniStar sites.

By letter dated February 1, 2006, Progress Energy notified the NRC staff that it plans to submit two COL applications, one for a site located in the Carolinas and one for a site in Florida, and that it has selected the Westinghouse AP1000 as the reactor technology and the Harris Nuclear Plant as the site for the Carolinas. The Florida site for the COL application will be determined in the near future. On February 21, 2006, the NRC staff met with Progress Energy to discuss their preparations for submitting a COL application. Progress is scheduled to submit its first COL application in late September or early October 2007 for the Harris site and a second application for a Florida site in late 2007 or first quarter 2008.

On November 15, 2005, the NRC staff met with Entergy Nuclear to discuss planning related to COL applications for its Grand Gulf and River Bend sites. The Grand Gulf application is scheduled to be submitted in either the 4th quarter of 2007 or the 1st quarter of 2008, and the River Bend application is scheduled for approximately 6 weeks after the Grand Gulf submittal. The Grand Gulf application will be a joint venture with NuStart and will reference the ESP, and both submittals will reference the GE ESBWR. Entergy stated that it is working with Dominion Nuclear, which is also referencing the ESBWR design, to submit a standardized COL application, and is working with GE on the certification of the ESBWR design. On December 5, 2005, Entergy Nuclear submitted a letter to the NRC staff to initiate pre-application activities.

On September 22, 2005, NuStart Energy announced that it had selected Grand Gulf and Bellefonte as the two sites it will use for its applications for COLs for new nuclear plants. The Grand Gulf site was designated for the GE ESBWR design and the Bellefonte site for the Westinghouse Advanced Passive 1000 reactor design. In its letter dated November 17, 2005, NuStart announced that it would be preparing a dual unit COL application for the Bellefonte site, which is scheduled to be submitted during the fourth quarter 2007, and a single unit COL application for Grand Gulf site, which is scheduled for fourth quarter 2007 or first quarter 2008. On February 7, 2006, the NRC staff held a public meeting with NuStart to discuss the Bellefonte COL pre-application activities. NuStart stated that it is planning on using some of the existing structures at the Bellefonte site, such as the cooling towers, intake structure, switchyard, and tower. During the meeting, the NRC staff and NuStart discussed the concept of the design-centered approach and standardization of COL applications among other applicants referencing the AP1000 design.

On December 5, 2005, South Carolina Electric and Gas (SCE&G) submitted a letter of intent to pursue new nuclear capacity. A COL application will be for two units and is targeted for submittal in the third quarter of 2007. In a February 10, 2006 letter to the NRC staff, SCE&G stated that it has chosen the Westinghouse AP1000 as the reactor technology and has selected the existing Virgil C. Summer Nuclear Station site as the location.

On March 13, 2006, the NRC staff received a letter of intent from an unannounced Advanced Boiling Water Reactor (ABWR) applicant. The applicant intends to submit an ESP application before the last quarter of 2007 and a COL application as soon thereafter as practicable. The letter contains proprietary information submitted under 10 CFR 2.390.

On March 16, 2006, Duke Energy announced that it had selected the former Cherokee site, near Gaffney, South Carolina for the development of a COL application utilizing two AP1000 units. Duke also announced the designation of two additional sites for possible future ESP development in Davie County, North Carolina, and Oconee County, South Carolina.

Regulatory Infrastructure

On November 3, 2005, the Executive Director for Operations issued SECY-05-0203 requested Commission approval to publish in the *Federal Register* revised proposed revisions to 10 CFR Part 52, as well as changes throughout the NRC's regulations to enhance the NRC's regulatory effectiveness and efficiency in implementing the licensing and approval processes in Part 52 and to clarify the applicability of various requirements to each of the regulatory processes in Part 52 (SECY-05-0203, "Revised Proposed Rule to Update 10 CFR Part 52, Licenses, Certifications, and Approvals for Nuclear Power Plants"). This rulemaking to enhance 10 CFR Part 52 is based on lessons learned during design certification and ESP reviews and on discussions with stakeholders about the ESP, design certification, and combined license review processes. This revised proposed rule would withdraw and supersede the Commission's July 3, 2003 (68 FR 40026) proposed rule on 10 CFR Part 52. On January 30, 2006, the Commission approved the withdrawal of the previously proposed rule and publication of the revised notice of proposed rulemaking. The Commission directed the staff to give high priority to complete this rulemaking activity on schedule and provide the proposed final rule to the

Commission no later than October 2006. The proposed 10 CFR Part 52 rule was published in the Federal Register on March 13, 2006 (71 FR 12781). On March 14, the NRC staff held a public meeting with stakeholders to discuss the proposed 10 CFR Part 52 changes and rulemaking.

On December 1 and 2, 2005, the NRC staff participated in a public meeting with the NEI Combined License Task Force. During the meeting, the NRC staff stated that it is developing a COL application regulatory guide based on Regulatory Guide 1.70, "Standard Form and Content of Safety Analysis Reports for Nuclear Power Plants." A draft of the regulatory guide is scheduled to be issued in June 2006 and the final in early 2007. Work-in-progress versions of each chapter of the regulatory guide are being placed on the NRC website between February and June 2006. The NEI Combined License Task Force has requested periodic meetings to discuss draft chapters after they are placed on the NRC website. On March 15, 2006, the NRC staff held a public workshop with stakeholders to discuss the draft Regulatory Guide (DG-1145) and its contents. There are three additional public meetings scheduled prior to DG-1145's scheduled issuance in June 2006.

In January 2006, the NRC staff posted the schedule for updating NUREG-0800, "Standard Review Plan," on the NRC extemal website at the following address: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/srp-schedule.pdf.

On January 12 and March 8, 2006, the NRC staff met with representatives from the Department of Energy (DOE) to discuss the use of a Laboratory Consortium to support new reactor licensing. The DOE Laboratory Consortium consists of the major DOE Office of Science Laboratories (Argonne National Laboratory, Brookhaven National Laboratory, Oak Ridge National Laboratory and Pacific Northwest National Laboratory). The NRC staff and National Labs are working to establish a collaborative approach with regard to leveraging multiple laboratories' resources to assist the staff in future new reactor licensing application reviews.

In February 2006, the NRC staff traveled to Finland and France to meet with regulatory counterparts regarding the Multinational Design Approval Program. Discussions were focused on possible cooperation in review of the EPR reactor design. Counterpart nations expressed interest in information exchange and cooperation, indicating topics other than the EPR review where NRC may be able to provide assistance.

On February 21, 2006, the NRC staff met with the Nuclear Energy Institute (NEI) to discuss the design-centered approach and standardization of a combined license application (COLA). The staff and industry representatives discussed various aspects of a reference COLA, including when one would be identified and submitted, and what portions of a reference COLA would be considered standard.

On March 6, 2006, NRC staff hosted a public meeting with NEI on the proposed rulemaking for security design expectations for new reactors. Industry representatives indicated their intentions to develop several documents that may help with the development of staff's guidance documents in support of the rule. Staff and industry agreed to continue to interact throughout the rulemaking process.

In March 2006, the Commission approved the NRC staff's recommendation to issue an Advanced Notice of Proposed Rulemaking (ANPR) on approaches for making technical requirements for power reactors risk-informed, performance-based, and technology neutral (10 CFR Part 53). The Commission directed the staff to complete the ANPR stage by December 2006 and to provide a recommendation by May 2007 on whether and, if so, how to proceed with rulemaking.

New Reactor Licensing Activities
As of March 31, 2006

Organization	Designs endorsed or under consideration	Sites under Consideration	Planned Applications	Date	Basis
General Electric	ESBWR	N/A	Design Certification	8/24/2005	8/24/05 Application Submitted
Framatome ANP	EPR	N/A	Design Certification	12/2007	Letter 11/4/05
Southern Nuclear Operating Company	AP1000	Vogtle	ESP and COL	8/2006: ESP 3/2008: COL	Letters 7/26 and 8/17/05 Mtg Summary (ML052710018)
Constellation	EPR	Nine Mile Point Calvert Cliffs, plus 2	COL	6/2008 and 6/2009	Press Release 11/2/05 Mtg Letter 11/4/05
Dominion	ESBWR	North Anna	COL	9/2007	DOE solicitation award and press release Letter 11/22/05
Duke	AP1000	Cherokee (2)	COL	Late 2007 or Early 2008	Letters 3/4/05, 10/25/05 and 3/16/06
Progress Energy	AP1000	Harris (2) Florida (2)	COL COL	Sept or Oct 2007 Late 2007 or 1st Qtr 2008	Letters 8/24/05 and 2/1/06 11/1/05 Mtg Press Release
NuStart Energy	AP1000 ESBWR	Bellefonte (2) Grand Gulf	COL COL	4th Qtr 2007 4th Qtr 2007 or 1st Qtr 2008	Letters 12/7/2004 and 11/17/2005, press release
Entergy	ESBWR	River Bend	COL	Early 2008	Press Release 11/15/05 Mtg Letter 12/5/05
South Carolina Electric and Gas	AP1000	Summer (2)	COL	3rd Qtr 2007	Letters 12/5/05 and 2/10/06
Unannounced ABWR Applicant	ABWR	TBD (2)	ESP and COL	3rd Qtr 2007:ESP (COL: soon after)	Letter 3/13/06

STATEMENT SUBMITTED

BY THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

TO THE

COMMITTEE ON ENERGY AND NATURAL RESOURCES

UNITED STATES SENATE

CONCERNING

NUCLEAR POWER PROVISIONS

ENERGY POLICY ACT OF 2005

PRESENTED BY

DR. NILS J. DIAZ

CHAIRMAN

SUBMITTED: MAY 22, 2006

Introduction

Mr. Chairman and Members of the Committee, it is a pleasure to appear before you today to discuss, on behalf of the Commission, the U.S. Nuclear Regulatory Commission's programs for new reactor regulation. We appreciate the support that we have received from the Committee, and we look forward to working with you in the future. We would also like to take this opportunity to thank Congress for the additional budgetary support that was provided last year. These resources are allowing the Agency to achieve earlier completion of safety and security programs and to begin structuring the Agency for reviewing new reactor applications. On a personal note, Mr. Chairman, I am grateful for the opportunity to serve this great country of ours for almost 10 years, first as a Commissioner and then as Chairman of the best nuclear regulatory agency in the world, and during extraordinary times. It has been my privilege to have worked with you to better serve the well-being of our people.

The NRC is dedicated to the mission mandated by Congress - - to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment - - in the application of nuclear technology for civilian use. We are committed to exercise this mandate with a regulatory framework that is effective, predictable, and that continues to meet the changing demands of the country. To achieve this goal, we have made preparations and continue to put in place the infrastructure needed to review the announced new reactor licensing and certification work, including the 13 announced combined license (COL) applications beginning in 2007. I would like to highlight our current and anticipated new reactor regulatory activities, a new system for licensing reviews, and new human capital and space planning initiatives designed to meet the new challenges posed by the dynamic nature of today's nuclear arena. The continued safe and secure operation of the current fleet of operating nuclear power plants remains the Agency's top priority; therefore, the new reactor licensing activities are being carefully planned to ensure the continued safe operation of these facilities.

New Reactor Licensing Workload

The Commission's Strategic Plan establishes a fundamental objective to:

Enable the use and management of radioactive materials and nuclear fuels for beneficial civilian purposes in a manner that protects public health and safety and the environment, promotes the security of our nation, and provides for regulatory actions that are open, effective, efficient, realistic, and timely.

Consistent with this objective and our statutory responsibility, the NRC has been conducting reviews of Early Site Permit (ESP) and Design Certification (DC) applications, and is developing an efficient infrastructure to conduct the review of anticipated combined license (COL) applications in the future.

As a result of the passage of the Energy Policy Act of 2005 and concurrent developments in U.S. energy demands, the NRC is preparing for an increased number of potential COL, ESP and DC applications. The Energy Policy Act incentives for new reactor construction established a highly dynamic environment in which new nuclear power plants are being seriously considered to meet future generation capacity, the need for which is expected to increase by the year 2015. Last year at this time, the NRC had been notified of three potential COL applications in the next few years. Today, the number of expected COL applications is 13 for a total of 19 units, and the number of applications is expected to increase in the near future. Some of these applications are expected to reference reactor designs already certified by the NRC, while others are expected to reference designs that are currently under NRC review. We also expect to be conducting reviews of additional ESP applications, or equivalent environmental reviews. We are preparing to review and act on applications anticipated to be submitted in the 2007-2008 time frame, and are organizing accordingly. We continue to assess our resource needs, which have increased significantly, in light of the very substantial increase in the number of anticipated COL applications and related work. The attached graph 1 shows the anticipated work schedule based on industry submittals, public announcements, and expected but as yet unannounced applications.

Current New Reactor Licensing Activities

Current new reactor licensing activities are expected to follow the processes established under 10 CFR Part 52. Part 52 establishes the framework to review ESP, CD, and COL applications.

The Commission recently proposed a revision to 10 CFR Part 52, to clarify it and enhance its usability. The proposed amendments incorporate the lessons learned from previous regulatory reviews, to enhance regulatory predictability at the COL stage. Furthermore, in the Part 52 rulemaking, the Commission is soliciting comments on an approach that would facilitate amendments to design certification rules after the initial certification. With such a provision, a detailed standard certified reactor design would be able to incorporate additional features that are generic to the design and thereby encourage further standardization. Also, changes to the limited work authorization process are being considered to expand the ability to initiate site preparation work in advance of COL issuance. The Commission plans to issue the final rule by January 2007.

NRC's licensing reviews are supported by regulatory guides and standard review plans. The NRC staff is reviewing and revising the regulatory guidance documents associated with new reactor licensing. These guidance documents include a planned combined license application regulatory guide which contains the information that COL applicants need to provide in their applications, and an update of pertinent standard review plan (SRP) sections for use by NRC staff reviewing COL applications. The Draft Regulatory Guide, which has been the subject of numerous public meetings and workshops, will be formally issued for comment in June 2006. The NRC staff estimates that the final regulatory guide will be completed by December 2006, to support prospective applicants who are planning to submit COL applications in late 2007 and 2008. This schedule is consistent with the schedule for the promulgation of the revised Part 52 rule. Complementary to the COL application regulatory guide, the NRC staff is updating the standard review plan to support the anticipated new site and reactor licensing applications. The staff is working with the industry to complete the standard review plan updates by the Spring of 2007.

To date, the NRC has received three ESP applications, focusing on environmental implications and emergency preparedness, for sites in Virginia, Illinois, and Mississippi which currently have operating reactors on them. The NRC staff has prepared safety evaluation reports for all three sites, and has issued draft environmental impact statements for public comment for two of the sites and has issued a final environmental impact statement for one of the sites. The agency will complete its remaining regulatory reviews in an effective, efficient, timely, and predictable manner. I note that additional work is being performed in connection with one application that was recently significantly revised and resubmitted by the applicant. Adjudicatory proceedings associated with the ESP applications are currently ongoing. From our experience with the ESP reviews, we have identified numerous lessons learned, for both the NRC and industry, that will be used to improve the staff's new reactor licensing process in the future and will be implemented prior to the next ESP application, expected during the summer of 2006.

The agency's work on new reactor standardized design certification has also intensified. Three designs were previously certified: General Electric's Advanced Boiling Water Reactor, Westinghouse's AP600, and System 80+ designs. The NRC recently certified the Westinghouse AP1000 reactor and codified it in the NRC's regulations, as Appendix D to 10 CFR Part 52. The NRC is currently reviewing the General Electric Economic Simplified Boiling Water Reactor (ESBWR) design certification application and is on schedule with respect to its review. The NRC is conducting pre-application activities for AREVA's U.S. Evolutionary Power Reactor (EPR) design whose design certification application is expected in 2007. The NRC is also conducting limited pre-application work for the Pebble Bed Modular Reactor (PBMR) and the International Reactor Innovative and Secure (IRIS), and is expecting additional design certification applications in the future.

To effectively review multiple COL applications in parallel, the staff is planning to implement a design-centered review approach. We believe this approach is crucial to achieving effective, efficient, and timely reviews for multiple applications. This approach is founded on the concept of "one issue-one review-one position for multiple applications" to optimize the

review effort and resources needed to perform these reviews. The NRC staff would use a single technical evaluation for each reactor design to support reviews of multiple COL applications for the same technical area of review, assuming that the relevant components of the applications are standardized. The design-centered approach will focus its reviews by: 1) using standardization and coordination of approaches and applications; 2) requiring complete and high-quality applications; 3) increasing the use of the DC rulemaking to codify issue closure; and 4) using single technical evaluations to support multiple COL applications.
In addition, to achieve consistency of the staff reviews, the process for implementing the design-centered review program will require a multi-layered project management team for each design, and will use dedicated technical review resources. The plans and schedules of these reviews include an increased level of detail and integration to achieve the requisite level of control and documentation. The benefits of this approach would be enhanced by the full participation of multiple entities in ensuring that pertinent components of the applications are standardized. A schematic representation of the sequencing and use of the design-centered review approach is shown in graph 2. Significant efficiencies are expected to be gained through the use of the design-centered approach.

New Reactor Construction Oversight

To prepare for the construction of new reactors licensed in accordance with 10 CFR Part 52, a new construction inspection program (CIP) is being developed. The new CIP builds on the lessons learned from the construction of the existing fleet of operating reactors. The CIP comprises four different parts, early site permit inspections; pre-combined license (Pre-COL) inspections; inspections, tests, analyses and acceptance criteria (ITAAC) inspections; and non-

ITAAC Inspections. These inspections will cover all aspects of new plant construction and operation from early site preparation work, through construction, to the transition to inspections under the reactor oversight process (ROP) for operating reactors. Half of the associated inspection procedures are in place and the remaining procedures are under development and are scheduled to be in place well before the start of on-site construction activities.

Successful implementation of the CIP will require four main functions: 1) day-to-day inspections at the construction site by resident construction inspectors; 2) on-site inspections by specialist inspectors; 3) off-site inspections (e.g., vendor inspections); and 4) documentation of inspection results and public notification of the successful completion of the ITAAC. ITAAC are part of the combined license and define specific requirements to be met prior to operation. To gain staff efficiencies and facilitate knowledge transfer, all construction inspection management and resources will be located in a single region which will schedule all construction inspectors nationwide.

The NRC performed an initial assessment of the existing ROP for use with new reactor designs which confirmed that the overall ROP framework could be used, including utilizing performance indicators and the significance determination process for evaluating inspection findings. The Construction Inspection Program will specifically address each new reactor to be built, detailing the steps that will be employed to integrate that plant into the ROP as it transitions from the construction phase into the startup and operations phase.

Multinational Design Approval Program (MDAP)

The NRC is working with international regulators on a multinational design approval program intended to leverage worldwide nuclear knowledge and operating experience in a cooperative effort to review reactor designs that have been or are being reviewed and approved in other countries. The first stage of the MDAP has already begun. It involves enhanced cooperation with the regulatory authorities in Finland and France to assist NRC's future design certification review of the US EPR. Follow-on stages of the MDAP could foster the safety of reactors in participating nations through convergence on safety codes and standards, and other technical matters while maintaining full national sovereignty over regulatory decisions. Preliminary work to more fully develop the framework for consideration of a Stage 2 is underway at the NRC and the Organization for Economic Co-operation and Development's Nuclear Energy Agency.

Challenges to Success

The NRC recognizes that many challenges for new reactor licensing activities exist. Key challenges include effective communication between the NRC and the applicants, and the interrelationship between the technical review and the associated adjudicatory process. To successfully complete the reviews within the anticipated schedule, continuous clear, effective, and timely communication between the NRC and the applicant must occur. Delays in providing or responding to requests for information must be avoided and any modifications to the application need to be conveyed immediately so that products can be appropriately coordinated. In addition, the technical review and adjudicatory process for the application are interrelated and both are required for the final decision making process. Multiple products are also needed to maximize the early resolution of issues leading to a final determination, including an ESP, DC and COL. An applicant may decide to submit a license application in a manner different from the originally contemplated sequence, such as choosing not to apply for an ESP prior to applying for a COL or selecting a design that has not been certified through rulemaking. In such cases, the technical review and adjudicatory process performed for an ESP or DC review will need to be included in the COL review and could challenge the predictability of the process and the application review schedule. To meet these challenges, we have implemented organizational changes in our legal and technical organizations, recruited personnel, and are developing an integrated planning tool to assist in coordinating the applicant schedules.

The NRC has completed substantial preparation activities and executed reviews of supporting elements for COL applications. We continue to incorporate the lessons learned from current reviews into the regulatory process to create a stable and predictable regulatory process. As such, the NRC is preparing to conduct thorough and timely reviews of ITAAC and, therefore, the use of the Energy Policy Act Risk Insurance Program, due to NRC delays should not be necessary. As noted previously, when COL applications are submitted, they should be high quality, essentially standardized applications that contain the safety case and other required components in the level of detail that will support staff review and the adjudicatory process. Anything less may challenge the predictability of the licensing process.

The NRC understands and accepts its role in new reactor licensing, the success of which depends on many factors, most notably the submittal of high quality applications by the industry. With the continued support of Congress, we will carry out our responsibilities and meet the challenges ahead.

Human Capital and Space Planning

As you know, the NRC's ability to accomplish its mission depends on the availability of a highly skilled and experienced work force. In a recent ranking of the Top 10 Federal Work Places by the Partnership for Public Service and American University's Institute for the Study for Public Policy Implementation, the NRC was designated one of the top three places to work in the Federal government. In addition, the NRC was ranked first by people surveyed who are under 40 years of age. The Commission is very proud of these rankings and strives to improve the quality of the work environment for NRC employees. Nonetheless, the NRC continues to be challenged by the substantial growth in new work at a time when increasing numbers of experienced staff are eligible to retire. To address these challenges, the agency has developed human capital strategies to find, attract, and retain staff with critical-skills and has developed a space acquisition plan to accommodate these additional employees.

The NRC is aggressively recruiting a mixture of recent college graduates and experienced professionals to meet the agency's emergent work activities. The current projection is that over 400 additional FTEs will be devoted to new work by FY 2008. The Commission is striving to hire approximately 350 new employees in FY 2006 to cover the loss of personnel and to support growth in new work. To date during this fiscal year, we have already succeeded in recruiting and hiring almost 300 new employees toward this goal. Our aggressive efforts to recruit, hire, and develop staff will continue throughout Fiscal Year 2007 as we prepare for receipt of the first COL applications. The agency expects to have a critical hiring need for at least the next five years.

The NRC closely monitors its voluntary attrition rate including retirements, which has historically been below six percent, and will continue to monitor this rate because it could increase as industry competition for skilled individuals increases and as eligible staff retire. The agency uses a variety of recruitment and retention incentives to remain competitive with the private sector. We continue to experience success utilizing the provisions of the Federal Workforce Flexibility Act of 2004 and the Energy Policy Act of 2005. The NRC has budgeted for continued and increased use of these recruitment and retention tools in the coming years.

Our steady growth and accelerated hiring program have exhausted available space at our Headquarters buildings. We have developed and are implementing strategies to obtain adequate space to accommodate our expanding work force. We are creating additional workstations within our Headquarters buildings, including building workstations in conference rooms, and are moving our Professional Development Center off-site to use the space it currently occupies for new employees. We are also seeking additional office space in the immediate vicinity of our headquarters complex to support the expected growth of the agency.

The NRC will be continually challenged to maintain adequate infrastructure and the personnel needed to accomplish its mission. However, with Congress' help, the Commission is poised to meet these challenges successfully through the ongoing human capital planning, implementation, and assessment process, the space planning program, and the various tools provided by the Energy Policy Act of 2005.

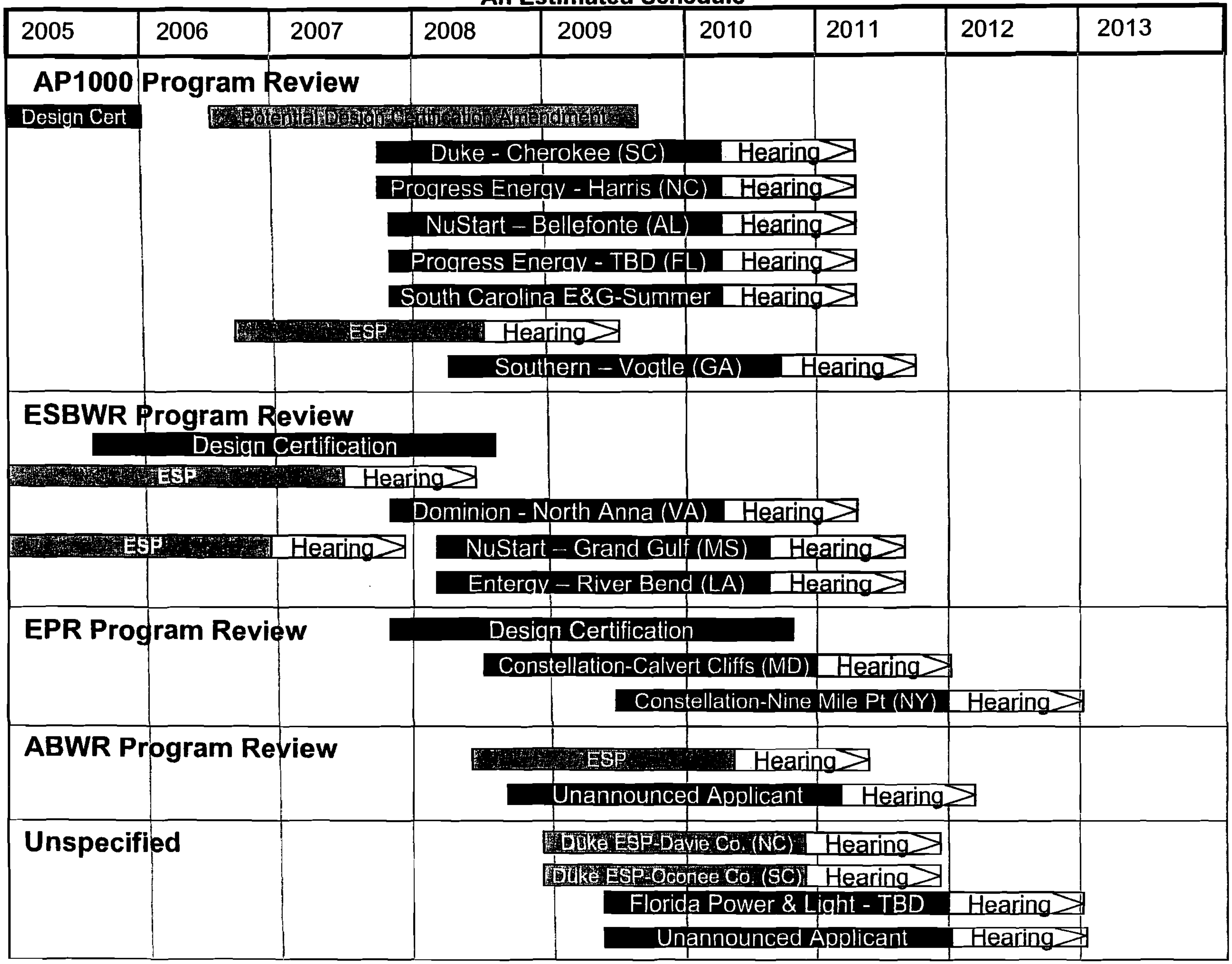
Conclusion

The Commission continues to be committed to ensuring the adequate protection of public health and safety and promoting common defense and security in the application of nuclear technology for civilian use. To that end, the Commission is dedicated to ensuring that our agency is ready to meet the expected demand for new reactor licensing. NRC's Part 52 processes are safety focused and are stable, efficient, and predictable. We have taken action to clarify Part 52, to ensure a clear regulatory and oversight framework; to reorganize the Agency and put in place the processes to ensure timely review; to meet the NRC's human capital and office space needs, and to seek additional funding as necessary. The Agency is prepared to meet the challenge associated with new reactors while maintaining strong oversight of the current operating reactors. I am convinced that the Agency has the technical and legal know-how to make the right decisions in a timely manner.

I appreciate the opportunity to appear before you today, and I look forward to continuing to work with the Committee. I welcome your comments and questions.

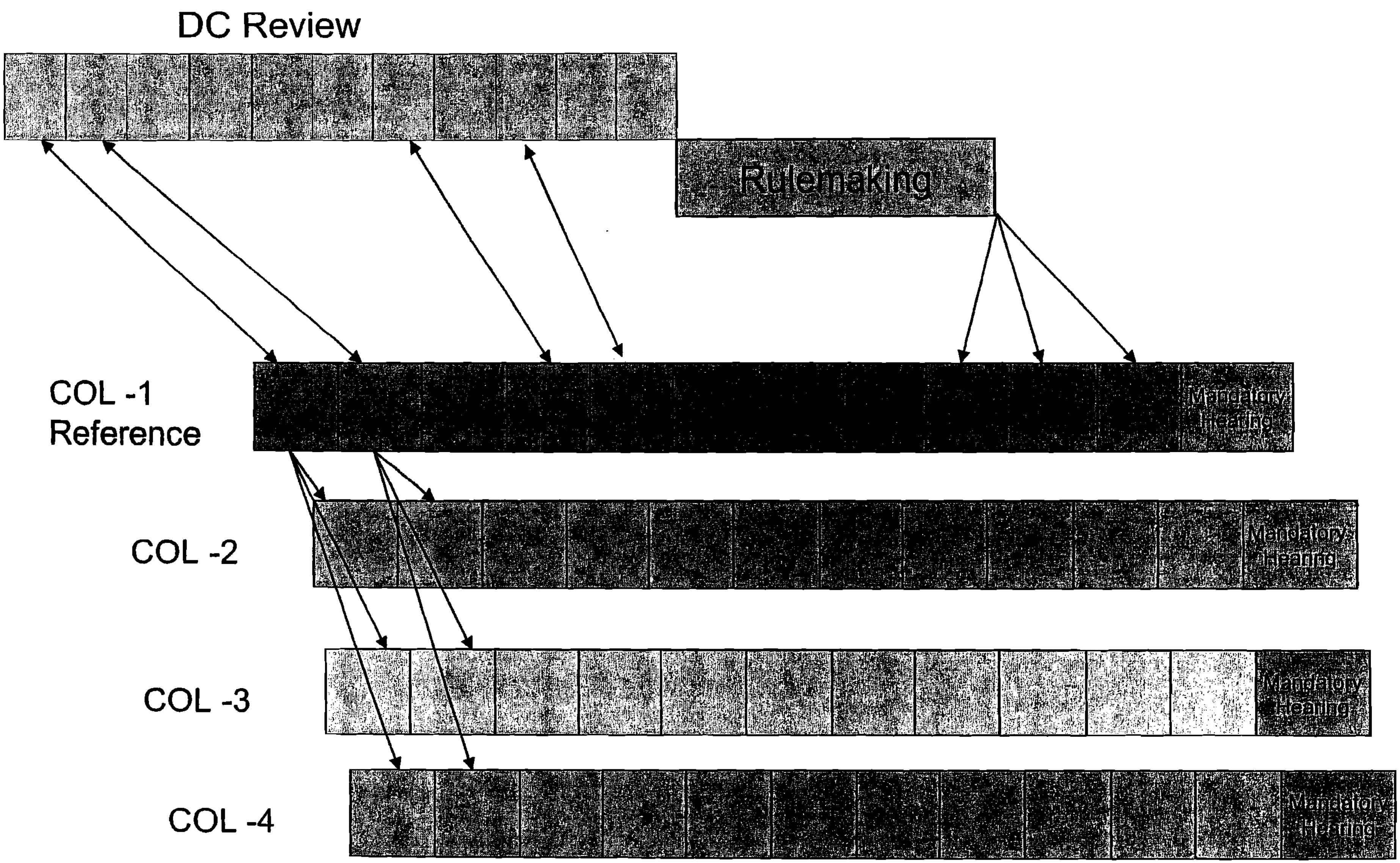
New Plant Licensing Applications

An Estimated Schedule



Graph 1

Design-Centered Review Approach



Graph 2

Inside NRC

Volume 28 / Number 10 / May 15, 2006

House panel boosts NRC funding for reactor reviews in FY-07

NRC's budget would get a $50 million boost on top of its $776.6 million fiscal 2007 request under a House Appropriations panel recommendation approved last week.

The NRC would receive approval to collect an additional $40 million from its licensees for work on license application reviews for new reactors and $10 million from DOE for oversight of the Hanford vitrification plant.

The NRC allowance is included in a $30 billion energy and water funding bill that received subcommittee approval last week. The DOE portion of the bill — which also funds US Army Corps of Engineers water projects, some Interior Department activities, and independent energy-related agencies — totals roughly $24.4 billion, $299 million below the request. Neither the subcommittee bill nor the accompanying report was publicly available last week. NRC has been planning a range of work to support the nuclear industry's growing interest in building new power reactors. The NRC budget request released in February said the agency was planning for technical reviews and mandatory hearings for five early site permit applications in FY-07, which begins October 1 (INRC, 20 Feb., 4). In addition, it expected to continue review work on one design certification application

and conduct pre-application
reviews on other advanced reactor
designs.

Since the budget document was prepared,
the planning assumptions have
changed dramatically. Previously, NRC
expected four combined construction
permit-operating license applications
(COLs). Now NRC says the industry has
indicated plans for 13-15 COLs between
2007-1009, NRC told key lawmakers last month.
Representative David Hobson, the Ohio Republican who
chairs the Appropriations subcommittee that controls energy
spending, lashed out last week at DOE's handling of the vitrification
plant project at the department's Hanford site in
Washington state. That project, which also is known as the
Waste Treatment Plant and which prepares for disposal
radioactive waste retrieved from underground storage tanks
at Hanford, is $6 billion over budget and six years behind
schedule, Hobson said.

"We propose major management changes to the Waste
Treatment Plant: 90 percent of design must be complete
before construction of major facilities [begins], a tighter linkage
between contract payments and contract performance,
and, most importantly, oversight by the Nuclear Regulatory
Commission of nuclear safety," Hobson said in his opening
remarks at the May 11 budget markup. The subcommittee
bill would set FY-07 funding for the project at $600 million,
$90 million below the budget request.

Subcommittee staffer Kevin Cook said NRC would not
license the facility but would have what he called "over-theshoulder
oversight" of work on the plant.

Elsewhere in the bill, the subcommittee reduced funding
for DOE's Global Nuclear Energy Partnership program to $150
million,from the $250 million sought, to launch the fuel-cycle
initiative next fiscal year. The suggested allocation matches the
level authorized in the Energy Policy Act of 2005 for spent fuel
reprocessing and recycling, the subcommittee said.

Hobson said he thought the Bush administration was
moving ahead too quickly with the program, calling the
information available about GNEP's waste streams and lifecycle
costs "inadequate."

But Hobson, who supports spent fuel reprocessing and
recycle, added that he doesn't want to kill GNEP because he

believes it includes some technology development work that is warranted. As envisioned by DOE, GNEP would close the nuclear fuel cycle in the US and abroad in part through a fuel lease-takeback program aimed at discouraging some countries from implementing their own uranium enrichment and spent fuel reprocessing programs. The initiative is aimed at encouraging an expansion of the use of nuclear power while also reducing the global proliferation risk and minimizing the radiotoxicity of waste slated for disposal in a repository. Funding from the cut to GNEP would be used, in part, to provide financial assistance to university nuclear engineering programs in the US and to help fund the weatherization of homes for low-income citizens. DOE had not requested any funds for university aid in FY-07, claiming that enrollments in university nuclear engineering programs had collectively met a goal of 1,500 students. The program received $26.7 million this fiscal year for various matching grants, fellowships, scholarships, and research grants. The program is aimed at attracting students to nuclear engineering, a field that could face workforce shortages in the next five years.

Also cut was DOE's mixed-oxide fuel project at the Savannah River Site in South Carolina. The bill would eliminate all funding for that program in FY-07 and would use roughly two-thirds of the $368 million sought for it for plutonium immobilization activities and cleanup work at Savannah River. The NRC already has issued construction authorization for the Savannah River plant but has not yet received an application for an operating license.

Hobson stressed that he believes that since Russia has signaled it does not intend to move forward with its own MOX project, there is no compelling reason for the US to build what was supposed to be a parallel MOX plant at Savannah River. The US-Russian program, in which work in the two countries was supposed to move forward at roughly the same pace, is aimed at the disposition of surplus weapons plutonium as MOX fuel. The long-standing US-Russian plan calls for Russia to load MOX into LWRs, but Russian insistence that the US and other countries pay for both construction and operation of the MOX fabrication plant has stalled progress. US officials are now giving a more sympathetic hearing to Russian proposals to use fast reactors instead of LWRs (NuclearFuel, 10 April, 3).

The action by the House Appropriations subcommittee last week is at odds with language in defense authorization legislation that the House approved May 11 in a 396-31 vote. In that bill, authorizers decoupled the US and Russian MOX programs, saying the US could proceed with the construction of a MOX plant at Savannah River regardless of the status of the Russian program.

Separately, the appropriations bill would fully fund the DOE repository project at Yucca Mountain, Nevada at $544.5 million. It also would give the program an additional $30 million for interim storage, subject to congressional authorization. Nuclear waste legislation, which has not yet been introduced in the House, would be the likely vehicle for such authorization, according to Hobson, who told reporters he is working with Representative Joe Barton on the interim storage issue. Barton, a Texas Republican, chairs the House Energy and Commerce Committee that has jurisdiction over the DOE repository program.

The full Appropriations Committee is expected to take up the bill May 16.—***Elaine Hiruo, Washington***
INSIDE NRC MAY 15, 2006

Commission, industry discuss future of risk-informed regulation

Maximizing the benefits of, and removing obstacles to, the expanded application of risk-informed approaches to regulation were the focus of a May 3 commission briefing by NRC staff and industry representatives.

Chairman Nils Diaz, a staunch proponent of riskinformed regulation, said in an opening statement that the NRC has "done well" in adopting a risk-informed approach, But "it seems like things slow down" due to "communication and implementation, rather than the principles," he said. The agency and industry "agree on the principles, and then we start disagreeing and spending an enormous amount of time in what the details are, and that's because people still don't realize that the technique is powerful, that it's flexible, [and] that it's safety-focused ..."

Commissioner Jeffrey Merrifield said that frustration regarding slow progress "may be reflective of the fact that we have really been dealing with some of the low-hanging fruit early on" and began "grappling with some of the tougher issues more recently, and have even tougher ones to deal with going forward."

James Levine of Arizona Public Service, who chairs industry's working group for risk-informed regulation, said that "momentum has slowed significantly" and "development and implementation of the major risk-informed rulemakings has taken far too long. While probabilistic insights give clarity to what is truly safety-significant, existing deterministic barriers that some perceive continue to be difficult hurdles to cross."

"Absent strong management oversight, there is a tendency amongst some to move towards supplemental use of risk insights on top of the existing deterministic methods or to make very minimal changes to the existing requirements while at the same time requiring extensive risk analysis," Levine said. "It is important to reinforce the commission's policy on using risk insights to focus on matters of high safety significance as many of these issues that have complicated these rulemakings have been associated with residual risk," he said.

Stephen Floyd, vice president for regulatory affairs at the Nuclear Energy Institute, agreed, noting that "if we continue to struggle with low risk-significant items on top of all the

risk assessments that will be required, we believe we will have lost ground." He said that "one of the things that has been most disappointing in working in the risk applications area over the last 10 years is that we seem to have to repeat that lesson on every application. And maybe it's because of the low-hanging fruit approach where we are picking a cherry over here and one over here and one over here."
(Floyd will retire from NEI at the end of June, and a successor is expected to be named this week when NEI holds its annual Nuclear Energy Assembly. Speculation is that Anthony Pietrangelo, who is currently NEI's senior director of risk regulation, will take over for Floyd.)

Commissioner Edward McGaffigan expressed his frustration that the NRC had not done more to spur licensees to develop probabilistic risk assessments, or PRAs. "I'm deeply frustrated that ... the infrastructure for risk-informed regulation is so, I don't know, threadbare," he said. "We are suffering from the half-measures of previous commissions, and we could have required high-quality PRAs," but "because of the lack of infrastructure, I have been less than enthusiastic about some of the initiatives — the 50.69 and the 50.46(a) initiatives — that have come along," McGaffigan said.
Some at the briefing characterized this issue as a "catch-22" or "chicken-and-egg" question: Industry is sometimes reluctant to expend resources needed to develop high-quality PRAs absent a specific regulatory application with clear benefits, but such applications cannot move forward until the quality of PRAs has improved.

"I think the thing that moves industry most is success. We're pretty good copycats. If something works well for somebody, and it seems to have a lot of benefits to it, everybody piles on and figures out how to make it happen for them," Floyd said. He urged a plan to facilitate adoption and implementation of two voluntary, risk-informed rules — 50.69 on categorization of systems, structures and components (INRC, 6 Feb., 15) and 50.46(a) on core cooling requirements (INRC, 14 Nov. '05, 1). "[S]how that they can work, and I think people will follow. If it's a successful application and a successful implementation with the pilots, people will make the investment in the PRAs by following the standards to be able to reap the benefits," he said.
"When things take years and years and years, and you don't see an end coming, then you start to lose interest ... that's part of the problem we have with a good part of the industry not jumping on board if you will is because they don't see the payback quick enough for the effort that's being put into it. So, again, I don't think it's the application. I think it's, what are the results?" Levine said.

Industry and the commission agreed that better communication will facilitate the process. Levine said industry proposes "semi-annual meetings of the NEI risk-informed regulatory working group with senior NRC management to discuss the process of the rulemakings associated with guidance and implementation of the pilots, or issues for the pilots" implementing risk-informed regulations. One goal of such meetings would be to "develop and publish schedules for final rules and pilot plant implementation," he said.

Commission briefings on risk-informed regulation should be held "at least on an annual basis," Levine said. "[T]he underlying theme of this meeting is that riskinformed and performance-based regulation is part of the fabric of the NRC. It is not going to go away," Diaz said at the meeting's conclusion. "I think that sometimes we are risk-averse ourselves and in many ways, you know, try to go to too many levels. That provides a safe regulatory path. But if we really are risk-informed, we should at times take those small risks that will put us on the right path," Diaz said.

Presentation slides and a transcript of the briefing are posted on NRC's web site at **http://www.nrc.gov/readingrm/doc-collections/commission/tr/2006/.**
—Steven Dolley, Washington

Diaz's multinational review proposal laid out in more detail

A newly created steering committee would start work right away on shaping a global licensing approach for advanced reactors and so-called Generation IV designs under the next phase of a plan envisioned by Chairman Nils Diaz. Diaz began floating the idea of a Multinational Design Approval Program (MDAP) in 2004 and formally presented the proposal at an IAEA general conference in Vienna in September 2005 after the NRC commissioners approved the first phase of his plan. While Diaz has discussed his proposal at international forums, the full-fledged plan has never been laid out publicly in the US — until now.

An overview of Diaz's MDAP proposal (COMNJD-5-06) was released last year after the commissioners agreed to move ahead with Stage 1. The other two phases were sketched out in a May 2005 white paper that was never released. But a three-page, updated outline dated March 28, 2006 was posted on NRC's electronic library Adams May 5 (accession number ML060900337).

Whereas his original proposal of Stage 1 called for multinational cooperation on NRC certification reviews of possibly Areva NP Inc.'s EPR, General Electric's ESBWR, and Atomic Energy of Canada Ltd.'s Advanced Candu Reactor series 700 or 1200, Diaz's March paper appears to have narrowed the review to only the EPR reactor, which is now under construction in Finland and has just been formally ordered in France (Nucleonics Week, 11 May, 1). The paper notes that Areva anticipates submitting to the NRC a design certification application for the 1,600-MW EPR in late 2007.

"Initial bilateral meetings were held in January and February 2006 between NRC and its regulatory counterparts in Finland and France, STUK and ASN, respectively," the paper said. "The first steps in Stage 1 will be centered around exchanging information on the breadth and depth of the ongoing EPR design reviews being conducted by the French and Finnish governments."

Areva officials have embraced the MDAP concept and have made clear their willingness to cooperate with NRC in Stage 1. At a January 10 meeting with the NRC staff, Areva submitted a list of possible topics for the first phase, including severe accident mitigation design features, probabilistic risk assessment sequences and technical assessments, engineered safety features, and fuel design.

Joseph Williams, a senior project manager in NRC's new reactor licensing branch, told Areva in an April 7 letter that the staff is considering how MDAP interactions with foreign regulators will impact the EPR design review. He said the cooperation might "shorten portions" of the review but might not condense the overall schedule since "all technical areas at or near the critical path for completion of the review need to be shortened to have a significant effect." NRC has generally estimated design certification review to take between 42 and 60 months.

Diaz's paper says that the "level of cooperation achievable" in the first phase would depend upon how standardized the design is among the three countries — Finland, France and the US. Applying what was learned in this phase to other new reactor designs would be considered on a caseby-case basis, he wrote.

Next two stages

The second stage, as Diaz sees it, would coincide with the first stage, which primarily focuses on foreign regulators assisting NRC in the review of new LWR designs that have been approved or are under review in other countries. The MDAP process would be formalized in Stage 2, Diaz said in his paper. The group of "core countries" that had been participating would form a steering committee to set the policy direction and establish working groups for specific "technical modules," according to the March white paper.
One technical module might examine "regulatory reciprocity on the manufacturing oversight of international reactor suppliers and components," the paper said. "Other possible technical modules for consideration include the design criteria, codes, and standards associated with the quality assurance, risk assessment, and severe accident mitigation features."

Jeffrey Jacobson of NRC's Office of International Programs said the commissioners have approved beginning "exploratory" work in Stage 2. There will be more discussion about MDAP at a June meeting of the OECD Nuclear Energy Agency, or NEA, in Paris, he said.

NEA would serve as secretariat during the Stage 2 developments, Diaz suggested in his white paper and at a meeting in Vienna last year (INRC, 3 Oct. '05, 1). Diaz noted in his paper that NEA serves now as secretariat for the Generation IV International Forum, or GIF, a DOE-led program involving international cooperative research and development of six highly promising next-generation nuclear technologies. He said using NEA in this capacity would be beneficial because of GIF's established infrastructure, and it would provide an opportunity to link the work in GIF and MDAP.

But Diaz emphasized that NEA would not step outside its role as secretariat. The technical direction, he said, would come from the MDAP steering committee.
The final phase, Stage 3, would expand the multinational regulatory reviews to Gen IV designs, or next-generation technology, based on the results of the GIF program. The Gen IV designs are expected to be ready for commercial deployment around 2030 or beyond.

Results unknown
NRC said last month that it was "too early" to tell whether MDAP interactions could shorten reactor design review schedules. The statement was in response to a question from Ohio Senator George Voinovich (Republican), who chairs the Environment and Public Works Subcommittee on Clean Air, Climate Change, and Nuclear Safety. In written correspondence following a March 9 hearing, Voinovich asked whether MDAP could reduce the agency's review schedule, estimated to take up to five years. He also questioned the costs and benefits of such international cooperation.

NRC responded in an April 19 letter, released April 26, that the benefits would depend on various factors. "The first factor is the degree of similarity among the designs proposed in the US and internationally," NRC said. "The second contingent factor is the level of review undertaken by the participating regulatory agencies to meet their national standards and how similar these standards are to those of the NRC."

NRC said the key to shortening the review would depend on whether it could use other regulators' work in technical areas. It said that the EPR design review would likely benefit from MDAP but other near-term design reviews, including the ESBWR review, would not use a multinational approach.

NRC has budgeted two full-time equivalents, or FTEs, for MDAP work in fiscal 2006 and 2007. An FTE is the government's measure for the number of workers equal to the pay of a full-time staffer. NRC's Jacobson said the efficiencies expected to be achieved through the program will "more than offset" the FTE.—***Jenny Weil, Washington***

DGSNR blesses EDF's program to backfit sump strainers by 2009

Electricite de France's program to backfit containment sump strainers on all of its PWR by 2009 is "acceptable," nuclear regulatory authority DGSNR said in an information note posted May 4 on its internet site (http://www.asn.gouv.fr/data/information/18_2006_ano4.asp).

The note related the history of the sump-clogging issue, notably DGSNR's Oct. 9, 2003 letter to EDF requiring the utility to study the risk of sump clogging for all 58 of its PWRs. It related that on Dec. 24, 2003, EDF replied that in certain low-probability accident sequences, such as complete rupture of main primary piping, clogging of the sump strainers could not be excluded, but it could be excluded for smaller-diameter pipe breaks. Some of EDF's reactors were more vulnerable to the phenomenon than others because of smaller strainer screen surfaces and use of types of thermal insulation that would generate more debris that could be entrained to the sumps, the utility said.

EDF had identified a reference case for the safety studies on the issue, including assumptions for analyzing phenomena important to sump clogging, DGSNR said. After consulting its reactor safety advisory group, DGSNR asked EDF to conduct additional studies, notably to test assumptions on chemical interactions between coolant and debris, but said that shouldn't prevent beginning of backfit work.

EDF had committed in 2004 to action designed to remedy the situation, DGSNR said. The utility proposed to replace existing strainers with ones featuring much larger screen surfaces. In 2005, three reactors, at Dampierre, Gravelines and Fessenheim, were backfit, DGSNR said (Nucleonics Week, 20 Dec. '05, 4).

Given the satisfactory experience with those backfits, it said, more reactors will be modified this year.
EDF projects completing the backfits on all of its reactors by the end of 2009, with priority given to the most vulnerable units, which are to be backfit by the end of next year, DGSNR said.

While it has accepted EDF's backfit proposal, DGSNR said it remains "attentive" to research that continues on the international level, notably by French expert organization IRSN, to better understand the sump clogging phenomenon. EDF officials have said that the sump strainer replacement was the single most effective measure in a list of potential backfits analyzed for safety gains compared to expenditure (INRC, 28 Nov. '05, 8).—***Ann MacLachlan, Paris***

Inside NRC

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Chemical effects, coating issues complicate PWR sump evaluations

Despite limitations of available data and evaluation methodologies, PWR licensees will be expected to account for chemical and downstream effects and possible coating failures in pending upgrades of their containment sumps, NRC staff said at a workshop last week.

The three-day workshop at a conference center in Rockville, Maryland was convened so that NRC staff could convey to industry the agency's expectations for adequate completion of evaluations and upgrades being pursued at all 69 PWRs in response to generic letter 2004-02 to cope with the potential impact of debris blockage on emergency recirculation during a design basis loss-of-coolant accident, or LOCA (INRC, 20 Sept. '04, 5). During the conference, agency staff also met with five vendors to discuss enhanced sump strainers that will be installed at almost all US PWRs by the end of 2007.

Technical issues still to be resolved include effects of chemical reactions in a post-LOCA containment on the amount of debris generated; so-called "downstream effects" on plant systems of fine debris that might pass through sump screens; debris transport issues; and quantifying the effects of debris accumulation on screens on recirculation performance. another issue is determining the extent to which various coatings applied in containment might fail to adhere during a LOCA and thereby contribute to debris generation, Michael Scott of NRC's Office of Nuclear Reactor Regulation said in his May 23 presentation at the meeting. Detailed presentations and discussions on each of these issue areas followed.

NRC staff and industry do not fundamentally disagree on the importance of evaluating and upgrading PWR sumps or on the general approach to resolving the issue by December 2007, as requested in the 2004 generic letter. Licensees have largely completed evaluations of their sumps using a methodology developed by the Nuclear Energy Institute and approved by the NRC, and the agency is reviewing those responses (INRC, 11 July '05, 1).

Differences of opinion now mostly revolve around technical issues related to the degree of detail that must be included in licensee evaluations of chemical, downstream and coating effects, and how to account for uncertainties in those evaluations in the design and implementation of sump upgrades.

These complex issues have proven to be recurring sore spots ever since NEI submitted its methodology for NRC review nearly two years ago. At the time, the methodology provided scant detail on how to evaluate these issues, primarily because little research had been conducted on them. Since that time, NRC and industry have both launched research programs on chemical and downstream effects (INRC, 18 April '05, 3), and data from these efforts are being incorporated into evaluations as they become available. Each licensee is expected to perform a plant-specific evaluation of these issues, though they may draw upon generic methodologies and data as relevant.

NRC staff is reviewing a Westinghouse topical report that details methodologies to evaluate and quantify chemical effects, WCAP 16530-NP, and plans to issue requests for additional information on the report in July, Paul Klein of NRR said at the meeting. The draft target date for a safety evaluation on this issue is May 2007, which was set "with the recognition that additional testing may be necessary to address staff RAIs," and the "thoroughness and timeliness" of industry's RAI responses "will influence" when the evaluation is issued, Klein said.

Klein's presentation also outlined a series of detailed technical questions on chemical effects that remain to be resolved. John Butler, senior project manager at NEI, provided industry's perspective on these issues in his presentation, addressing the technical questions and noting that "licensee strainer testing, including chemical effects, is being conducted on a schedule to support the current resolution schedule" for the sump issue.

"Almost all licensees" are using a second Westinghouse topical report, WCAP-16406P, as their evaluation methodology for downstream effects, Thomas Hafera of NRR said. Agency staff has completed a "preliminary review" of the report and provided comments to the Westinghouse Owners Group. Discussions with industry on this approach continue,

but most issues have now been addressed, and a "path forward" identified that potentially offers "something amenable to both parties," Hafera said. A revision of the topical report will be submitted to the NRC for review this month.
NRC review of the downstream issues topical report is "ongoing" and "more items will most likely be found" requiring discussion and possible revision, Hafera said. Once the topical report is approved, licensees will "have the responsibility to make sure that RAIs are answered by the new methodologies and issues addressed," he said.
Tim Andreychek of Westinghouse responded for industry on downstream issues, pointing out that "further explanation will be incorporated" into the topical report as it is revised and reviewed. "We're satisfied we've identified the path forward," Hafera said.

Coatings issue remains contentious
One point of disagreement at the meeting was appropriate treatment of containment coatings in sump evaluations.
In a January 16 letter to NEI, Brian Sheron of NRR reiterated staff's previously expressed position that "licensees would need to identify and institute a coating testing program to assure that previously applied coatings continue to meet the qualification standards necessary to assure they will not fail during a LOCA." In the absence of such a program, staff's position is that "all coatings inside containment should be assumed to fail during a LOCA and be available for transport to the sump," Sheron said.

Agency staff has also challenged the adequacy of visual evaluations of the coatings. "While a coating may appear visually sound, there is no assurance that the coating continues to meet adhesion and other qualification requirements that provide the assurance that the coating won't fail during a LOCA," Sheron said in his letter.

In his March 31 reply, Anthony Pietrangelo, NEI senior director for risk regulation, said that "while there have been some noted instances in which qualified coatings have experienced problems, these instances have been infrequent and have been thoroughly investigated by both the industry, and, in some cases, by NRC." Pietrangelo pointed out that the use of visual inspections of coatings is endorsed by NRC regulatory guidance in RG 1.54, Rev. 1 and the generic aging lessons learned report, Nureg-1801, used to assess certain plant systems, structures and components during license renewal reviews.

"An industry-wide review by [the Electric Power Research Institute] of coatings assessment records could find no documented instances of degradation of reactor containment coating systems, or any other industrial coatings systems, that did not first exhibit visual precursors that could be detected and investigated by qualified personnel during periodic examinations," Pietrangelo said.

In his April 26 reply, Sheron said that "the NRC staff interprets observed visual degradation to mean that the coating systems in fact failed to meet their design requirements before visual indications existed, and had physical testing of the containment coatings been performed on a routine basis, the degradation would have been identified before visual indication appeared. Coatings that exhibit visual signs of degradation most likely have been in a degraded state for an extended period, representing a source of debris in a design-basis accident.

"Because a fundamental difference in opinion appears to exist on the adequacy of visual coatings inspections" and "further debate on this topic will not generate a timely resolution," NRC staff believes "a different approach may be necessary," Sheron said. This approach "may involve either the implementation of a physical testing program to ensure the adherence of the coatings to the substrate, or transport and/or sump strainer testing with representative coating debris to demonstrate that coating debris will not challenge strainer performance," he said.

These positions did not change much at the May 23 meeting. Ervin Geiger of NRC's Office of Nuclear Regulatory Research (RES) presented research results that showed that coatings debris from 1/64 inch to two inches in size "have limited potential for transport" to sump screens at specified test stream velocities. However, "if dropped onto the water surface, alkyd coatings debris and a fraction of the heavier coatings debris may remain buoyant and transport," Geiger said in his presentation. Therefore, "licensees must be able to justify the characteristics — size, density, and shape — of their coating debris in order to take credit for lack of debris transport to the sump," he said.

Matthew Yoder of NRR said that staff will review results of tests by two vendors and coatings research by EPRI, and RES is analyzing the results of NRC's coating transport tests, to be published this fall. Staff is "interacting with industry to resolve concerns about assessment of qualified coatings for degradation in service," Yoder said.

Noting staff's concerns about the adequacy of visual inspection, Yoder said that "perhaps the largest challenge our staff faces at this time is methodologies licensees are using to assess their qualified coatings." NRC staff "don't need new data; we need a new approach by industry in time to resolve" the sump debris issue, he said.

NEI's Butler and Dan Cox of the Nuclear Utility Coatings Council, an EPRI forum for sharing of technical information on coatings, defended visual inspections at the meeting. Cox noted that "visual indications" of coatings degradation "occur early in the degradation process; they're not necessarily terminal events." Visual indications of degradation "occur early on" and "a trained assessor will pick them up," he said. Periodic visual inspections during outages therefore "provide reasonable assurance that coatings will function as designed," Cox said.

"Nothing presented to us supports that," NRR's Scott responded, and Yoder said he hasn't seen any data to show that visual inspections detect coatings degradation early in the process.

Cox detailed an industry research plan to examine four categories of coatings at four to six plants during fall 2006 outages to determine that "visually sound design basis accident-qualified coatings meet their original adhesion criteria." The research plan has not yet been formally approved but will likely be carried out as part of EPRI's aging research program, Cox said.

NRC staff and industry representatives agreed to hold another meeting on the coatings issues once test protocols for the research plan are fully developed. Cox noted that the standards development organization ASTM International will host a workshop on July 18 and 19 to provide information and training on condition assessment of coatings in containments. Information on the workshop will soon be posted on ASTM International's web site **(http://www.astm.org).—*Steven Dolley, Washington***

The New York Times
nytimes.com

/la , 2006

:DITORIAL

[he Greening of Nuclear Power

Jot so many years ago, nuclear energy was a hobgoblin to environmentalists, who feared the potential for catastrophic accidents ınd long-term radiation contamination. But this is a new era, dominated by fears of tight energy supplies and global warming. ;uddenly nuclear power is looking better.

[he nuclear industry recently trotted out two new leaders of its campaign to encourage the building of new reactors. They are :hristie Whitman, the former administrator of the Environmental Protection Agency, and Patrick Moore, a co-founder of ;reenpeace. This campaign is the latest sign that nuclear power is getting a more welcome reception from some environmentalists vho have moved on to bigger worries.

[rue, most environmental organizations remain adamantly opposed to any expansion of nuclear power and instead look to :onservation and renewable energy to get us out of the fossil fuel age. But when the ecologist James Lovelock — creator of the Gaia ıypothesis, which holds that Earth and all its organisms behave as if they were a single living system — urges his colleagues to irc r "wrongheaded opposition" to nuclear energy, it is clear that fissures are developing.

[here is good reason to give nuclear power a fresh look. It can diversify our sources of energy with a fuel — uranium — that is both ıbundant and inexpensive. More important, nuclear energy can replace fossil-fuel power plants for generating electricity, reducing :he carbon dioxide emissions that contribute heavily to global warming. That could be important in large developing economies ike China's and India's, which would otherwise rely heavily on burning large quantities of dirty coal and oil.

3ut nuclear power should not be given a free pass in our frantic quest for energy and environmental security. Making any real dent n carbon emissions could require building many hundreds or even thousands of new nuclear plants around the world in coming lecades, a hugely ambitious undertaking fraught with challenges.

\s nuclear expertise and technologies spread around the world, so does the risk that they might be used to make bombs. Jnfortunately, the Bush administration erred badly when it signed a nuclear pact with India that would undercut the Nuclear Jonproliferation Treaty, the cornerstone of international efforts to prevent the spread of nuclear weapons. That misguided deal ıeeds to be repudiated by the Senate. We can only hope that it does not undercut a more promising administration plan to keep he most dangerous fuel-making technologies out of circulation by supplying developing nations with uranium and taking the pent fuel rods back.

'here remains the unsolved problem of what to do with the radioactive waste generated by nuclear plants. Many people are ıwilling to see a resurgence in nuclear power without some assurance that the spent fuel can be handled safely. The Energy epartment's repeated setbacks in efforts to open an underground waste repository at Yucca Mountain in Nevada do not inspire ›nfidence, but there is no reason why the spent fuel rods can't be stored safely at surface sites for the next 50 to 100 years.

:ore problematic is the administration's long-term solution for waste disposal. It wants to recycle the spent fuel in a new eneration of advanced reactors that would use technologies that don't yet exist, following a timetable that many experts think ırealistic. Its current approach is apt to be costly and would leave dangerous plutonium more accessible to terrorists.

uclear power has a good safety record in this country, and its costs, despite the high initial expense of building the plants, are ›oking more reasonable now that fossil fuel prices are soaring. How much impact it could really have in slowing carbon emissions as yet to be spelled out, but there is no doubt that nuclear power could serve as a useful bridge to even greener sources of energy.

Final Draft Generic Letter "Post-Fire Safe-Shutdown Circuits Analysis Spurious Actuations"

Sunil Weerakkody, Chief
Fire Protection Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation

ACRS Meeting
Rockville, MD
May 31, 2006

2

Purpose of Meeting

- To present the final draft Generic Letter 2006-XX: "Post-Fire Safe-Shutdown Circuits Analysis Spurious Actuations"
- To obtain ACRS endorsement to issue the proposed generic letter

Outline

- Staff Introduction
- Overview
- Probability of Spurious Actuations Due to Fires
- Summary and Objective of the Generic Letter

Overview

- Industry/NRC views on credibility of multiple spurious actuations
- NEI/EPRI cable fire test results
- Risk-informed inspections
- Re-establishing compliance

Fire-Induced Cable Faults

Presenter
Daniel Frumkin
Fire Protection Engineer

ACRS Meeting
Rockville, MD
May 31, 2006

Outline

- Background
- Test Objective
- Test Details
- Test Results
- Conclusion

Background

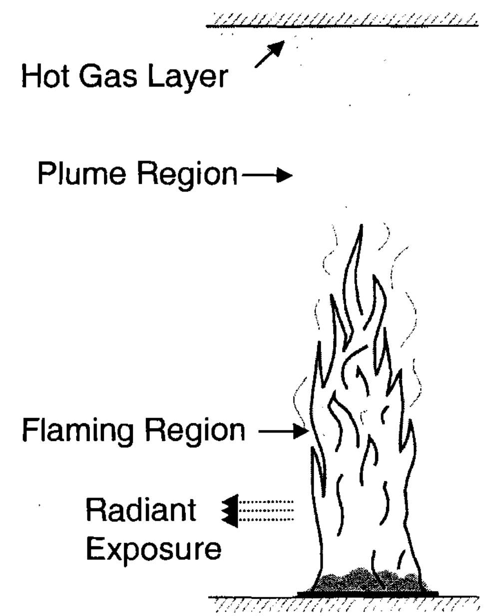
- The NEI/EPRI testing intended to address fire-induced circuit failure issues of concern to NRC staff, principally the potential spurious operation of equipment.
- NRC witnessed test and Sandia National Laboratory performed some insulation resistance testing during the NEI/EPRI Testing.
- Documents produced following the testing:
  - Characterization of Fire-Induced Circuit Faults – Results of Cable Fire Testing, EPRI 1003326, December 2002©
  - Circuit Analysis – Failure Mode and Likelihood Analysis, NUREG/CR-6834, September 2003
  - EPRI/NRC-RES, Fire PRA Methodology for Nuclear Power Facilities, EPRI 101989, NUREG/CR-6850, September 2005
  - Spurious Actuation of Electrical Circuits Due to Cable Fires: Results of an Expert Elicitation, EPRI 1006961, May 2002©

Test Objectives

- Research and test effort undertaken jointly by EPRI and NEI to investigate, characterize, and quantify fire-induced circuit failures.
- To better understand the electrical response of typical nuclear plant cables and circuits to fires.
- NRC and Sandia National Laboratory provided input to the test plan and witnessed testing

Test Details

- 18 cable fire tests were conducted between January 9, 2001 and June 1, 2001 at the Omega Point Laboratories test facility in San Antonio, Texas.
- The following types of exposure are included within the Test Program scope:
  - Hot gas layer exposure
  - Plume exposure
  - Radiant exposure



Test Results

- Over 80% of the hot shorts for the multi-conductor cable involved multiple conductors
- The spurious actuation data shows that a single internal hot short within a multiconductor cable usually affected both actuation devices simultaneously.

Test Results

Spurious Actuation Probability “Best Estimates” Given Cable Damage

	From EPRI Report Results	NUREG CR/6850
Thermoset (TS) Tray Intracable/Intercable without Control Power Transformer (CPT)	0.6 / 0.2	0.6 / 0.4
TS Conduit Intracable/Intercable with CPT	0.075 / 0.05	0.075 / 0.05
Thermoplastic (TP) Tray Intracable/Intercable without CPT	0.3* / 0.2*	0.6 / 0.4
TP Conduit Intracable/Intercable without CPT	No results for TP conduit	0.15 / 0.1
Armored Cable Intracable with CPT	0.075	0.075

* Use of CPTs is not identified in EPRI Results Table

Test Results (Cont.)

- Use of CPTs reduce spurious actuation probability by half, lack of CPT doubles probability
- All intercable interactions are between two single conductor cables, intercable spurious actuations between multiconductor cables are lower

Conclusion

- A review of the test data readily illustrates that hot shorts often involve more than one conductor.
- Concurrent hot shorts within a cable are probable and should be considered during circuit analysis.

Final Draft Generic Letter "Post-Fire Safe-Shutdown Circuits Analysis Spurious Actuations"

UNITED STATES NUCLEAR REGULATORY COMMISSION

Presenter
Robert Wolfgang
Fire Protection Engineer

ACRS Meeting
Rockville, MD
May 31, 2006

Presentation Summary

- Purpose of Issuing Generic Letter
- Requested Information From Licensees
- Background Since 1997
- Basis for Generic Letter
- Issue Clarified in Generic Letter
- Public Comments
- Summary

Purpose of Issuing the Generic Letter

- Clarify how the NEI/EPRI cable fire test program re-affirms long-held regulatory positions.
- Provide part of the foundation for licensees planning to transition to NFPA 805.

Purpose of Issuing the GL (Cont.)

- Respond to Agency's need to provide clarification and closure of outstanding fire protection issues.
- Respond to licensees' request to provide clarification of regulatory expectations.
- Respond to Regions' request to provide clarification of regulatory expectations for circuit inspections (resumed Jan. 2005).

Requested Information from Licensees

- Within 90 days, evaluate licensing basis and information in GL regarding multiple spurious post-fire safe-shutdown circuit analyses. Conclude whether the NPP is in compliance with regulatory requirements.
  - If not in compliance, submit functionality assessment of affected systems, structures, and components (SSCs).
  - If not in compliance, submit description of compensatory measures put in place.

Requested Information from Licensees (Cont.)

- Within 6 months, submit the plan to return all affected SSCs to compliance with regulatory requirements (plant mods, license amendments/exemption requests, etc.).
- Within 30 days, provide notification if cannot meet requested completion date (state why and proposed schedule/course of action).

Background Since 1997

- Multiple LERs brought lack of consensus concerning circuits to the staff's attention, which led to a moratorium on inspection of circuits issues (1997)
- NEI/EPRI cable fire tests demonstrated that multiple spurious actuations can occur and that they can occur in rapid succession without sufficient time for mitigation. Therefore, if a licensee does not account for multiple spurious actuations in its circuits analysis, the licensee may not be in compliance with 10 CFR 50.48 and 10 CFR Part 50, Appendix A, GDC 3, which require that a licensee provide and maintain free from fire damage one train of systems necessary to achieve and maintain safe shutdown.

Background Since 1997 (Cont.)

- Developed risk-informed approach to inspections to focus on risk-significant configurations (based on cable fire tests) (RIS 2004-003).
- Held public meeting in Atlanta to discuss staff positions and solicit stakeholder feedback (2004).
- Worked with NEI to finalize an acceptable industry guidance document for circuit analysis (NEI 00-01) (2005).
- Issued RIS 2005-30 to clarify regulatory requirements for circuit analyses. Addresses "associated circuits," "any-and-all," and emergency control stations.
- Draft GL issued for public comment (October 2005)
- Public meeting held (March 2006).
- Pertinent public comments incorporated into final draft GL.
- Received CRGR approval to issue the GL.

Basis for Generic Letter

- Review of NRC regulations, generic communications, correspondence, etc., related to this issue (references are identified in the GL).
- Results of NEI/EPRI cable fire test program.
- Input from inspectors on issues that need to be addressed.

Issue Clarified in Generic Letter

- Multiple Spurious Actuations
  - Some licensees claimed that only a single spurious actuation must be assumed in circuit analysis based on a misinterpretation of GL 86-10 response to question 5.3.10.
  - Some licensees claimed that multiple spurious actuations occur with sufficient time between actuations to take mitigating actions.

Issue Clarified in Generic Letter (Cont.)

- NRC letter from Sam Collins to NEI on March 11, 1997 stated that multiple spurious actuations caused by fire-induced hot shorts must be considered and evaluated.
- Byron and Braidwood have SERs approving assumption of a single spurious actuation per fire event (If staff position is applied to them, it would be a compliance backfit).
- The GL clarifies the regulatory requirement that multiple spurious actuations must be considered and evaluated.

Issue Clarified in Generic Letter (Cont.)

- The staff position on multiple spurious actuations presented in the GL is consistent with Section 9.5.1 of the Standard Review Plan.

Public Comments

- Significant public comment was that GL constituted a backfit to licensees
- Staff addressed comment and obtained CRGR approval

Summary

- The GL is a request for information from licensees.
- Industry cable fire test program re-affirmed staff interpretation of regulatory requirements.
- The GL is necessary to ensure that all risk-significant circuit situations are identified and addressed.

(3)

Bounding the Fire Risk from Circuit Spurious Actuations at Nuclear Power Plants

Raymond H.V. Gallucci, Ph.D., P.E.

U.S. Nuclear Regulatory Commission (NRC), MS O-11 A-11, Washington, D.C. 20555, rhg@nrc.gov

INTRODUCTION[a]

The U.S. NRC has requested that nuclear power plant (NPP) licensees review their fire protection (FP) programs to confirm compliance with regulatory requirements related to the phrase "one-at-a-time" for multiple spurious circuit actuations [1]. The Electric Power Research Institute (EPRI)/Nuclear Energy Institute (NEI) cable fire tests showed a relatively high probability of simultaneous or rapidly successive multiple spurious actuations during or after a fire [2]. This paper presents a bounding analysis on the potential fire risk in terms of core damage frequency (CDF).

BASELINE

The Individual Plant Examination for External Events (IPEEE) at a typical "older" NPP reported a fire CDF = 3.3E-5/y [3]. This included the modeling of "hot short" failures (i.e., spurious openings/closures of motor- or air-operated valves [MOVs or AOVs]), for which a maximum failure probability of 0.1 was assumed.[b] A review of the importance measures for the 24 hot short basic events that appeared in cut sets above the truncation level (1E-10/y) indicates a summed Fussell-Vesely (FV) importance of 0.0547, corresponding to a fire CDF contribution of (3.3E-5/y)(0.0547) = 1.8E-6/yr.[c] Among these 24 hot short basic events are 10 that correspond to five pairs of components, i.e., systemically symmetric components in redundant trains for which the failure characteristics, locations, and, presumably, cable run locations are similar. The summed FV contribution from these 10 events is 0.0320, corresponding to a fire CDF contribution of (3.3E-5/y)(0.0320) = 1.1E-6/y.[d]

For these 10 paired hot short events, the cut sets in which they appeared are assumed to be of the following forms:

- For "A" train hot short basic events – $CDF_A = F_A \bullet A \bullet \sum(B_j' \bullet X_j)$
- For "B" train hot short basic events – $CDF_B = F_B \bullet B \bullet \sum(A_k' \bullet Y_k)$

where:

- CDF_i = fire CDF contribution from cut sets containing i = A or B, each representing a hot short basic event for that train (A or B)
- F_i = fire initiator that induces hot short failure i
- A' or B' = non-hot-short-induced basic event failure corresponding to hot short failure for train A or B, i.e., A' pairs with B and B' with A
- X or Y = non-fire-induced failures that complete the cut sets for CDF_A or CDF_B, respectively, i.e., X pairs with A•B' and Y pairs with B•A'

Probabilistically, A = B = 0.1.[e] We can further express $\sum(B_j' \bullet X_j)$ as $\underline{B}' \bullet \sum(X_j)$, where $\underline{B}'$

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[a] This paper was prepared by an employee of the U.S. NRC. The views presented do not represent an official staff position.

[b] The value of 0.1 was assumed for all MOV and AOV control cable hot shorts; 0.001 was used for hot shorting of multi-phase AC power cables for MOVs.

[c] The sum of the individual FV's represents an upper bound on the total contribution from all hot short basic events because there may be cut sets where multiple hot short basic events appear, such that summing their individual FV's produces some "double-counting." Given that the maximum individual FV is 0.0109 for PORV failure (two such events), the effect of any double-counting is believed to be small and the sum of the individual FV's reasonably representative of the total hot short contribution to fire CDF.

[d] The same caveat as in the immediately preceding footnote regarding double-counting applies here as well.

[e] We ignore the contributions from those hot shorts for AC power cables for MOVs, where the probability is 0.001, since cut sets from these will likely contribute

$= \sum(B_j' \bullet X_j)/\sum(X_j)$. Doing likewise, we obtain $\sum(A_k' \bullet Y_k) = \underline{A}' \bullet \sum(Y_k)$, where $\underline{A}' = \sum(A_k' \bullet Y_k)/\sum(Y_k)$. Because of the symmetry involved with these paired hot short events, we can further assume $\underline{A}' = \underline{B}'$ and $\sum(X_j) = \sum(Y_k)$ in probabilistic terms.

With these simplifying assumptions, the contribution to fire CDF from the 10 paired hot short events becomes the following:

- $CDF_A + CDF_B = (F_A + F_B) \bullet A \bullet \underline{B}' \bullet \sum(X_j) = 1.1E\text{-}6/y$

which we can express as $(F_A + F_B) \bullet \sum(X_j) = (1.1E\text{-}6/y)/(A \bullet \underline{B}')$. We already know that A = 0.1, so the ratio on the right will be minimized for a maximum value of $\underline{B}'$, which is a weighted average of the various values of B' that appear in the cut sets. Since we are dealing with hot shorts for MOVs and AOVs, the non-hot-short-induced failures that comprise the various values of B' are the familiar "random" component failures, such as valve failure to open/close. Unreliabilities or demand failure probabilities for these tend to peak around 0.001. So, assuming $\underline{B}'= 0.001$ will minimize the above ratio, such that $(F_A + F_B) \bullet \sum(X_j) = (1.1E\text{-}6/y)/([0.1][0.001]) = 0.011$.

BOUNDING ANALYSIS

For the 10 paired events, any dual failures caused by a pair of hot shorts would appear in cut sets of the following forms:

- If initiated by $F_A - s \bullet F_A \bullet A \bullet B \bullet \sum(X_j)$
- If initiated by $F_B - s \bullet F_B \bullet A \bullet B \bullet \sum(Y_k)$

where s = fire severity factor reducing the likelihood of the more extreme fire (i.e., $s \bullet F_i$ [i = A or B]) assumed necessary to cause dual hot shorts.[f] Probabilistically, we can employ the previously assumed equivalences to express the total contribution to fire CDF from these paired hot shorts as follows:

- $CDF_{pairs} = s \bullet (F_A + F_B) \bullet A^2 \bullet \sum(X_j)$

To approximate s, we note that the Fire Protection Significance Determination Process (FPSDP) uses a value of 0.1 to reflect the fraction of fires of a particular type that will produce the 98th (vs. the 75th) %ile heat release rate, characteristic of an extreme fire of that particular type [4]. To approximate A, we note that the FPSDP assumes a maximum probability of hot shorting of 0.6 for non-conduit thermoplastic or thermoset cables where intra-cable or inter-cable hot shorts are possible. NUREG/CR-6850, the basis reference for the FPSDP, reduces this value to 0.3 if the cable is protected by a control power transformer, which is the typical case [5]. Since this typical "older" plant likely has a mix of thermoplastic and thermoset cables, 0.3 seems a reasonable assumption for A as the hot short probability. Therefore, assuming s = 0.1, A = 0.3, and using the quantification from above for the remaining terms, we obtain the following bounding estimate for fire CDF due to simultaneous or rapidly successive multiple spurious actuations:

- $CDF_{pairs} = (0.1)(0.011)(0.3)^2 = 9.9E\text{-}5/y \approx 1E\text{-}4/y$.[g]

CONCLUSION

There likely are some conservative assumptions in this estimate, especially in terms of fire characteristics and cable layout. However, it is instructive to note that, even if the

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negligibly compared to those resulting from control cable hot shorts.

f An implicit assumption here is that a fire of lower intensity but higher frequency, characterized by F_A (or F_B) alone (i.e., without the fire severity factor "s"), would not be extreme enough to cause dual hot shorts, but only hot short "A" (or "B"). Thus, without the factor "s" present to characterize the fire of higher intensity but lower frequency (i.e., $s \bullet F_A$ [or $s \bullet F_B$]), assumed extreme enough to cause dual hot shorts, it was not possible to have both A and B (or B and A) in a cut set initiated by F_A (or F_B) alone as on the previous page. With the factor "s" present, both A and B can be caused by either fire ($s \bullet F_A$ or $s \bullet F_B$). This is a surrogate approach used in lieu of actual fire modeling for this analysis since the details required to perform fire modeling are not available. The factor "s" is assumed to be the same for either fire initiator.

g If we employed the re-evaluated fire CDF discussed in the first footnote (1.1E-5/y), this value would be reduced by a factor of ~3 to 3.3E-5/y.

estimate is an order of magnitude too high, it would still be fairly significant at ~ 1E-5/y.[h] Thus, at least for a typical "older" plant, one cannot *a priori* dismiss multiple hot shorts as being of low risk significance.

REFERENCES

1. USNRC, *Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations,* Generic Letter 2006-xx, Washington, D.C., June 2006.

2. EPRI, *Spurious Actuation of Electrical Circuits Due to Cable Fires: Results of an Expert Elicitation*, EPRI TR-1006961, Palo Alto, California, May 2002.

3. "Response to Request for Additional Information on IPEEE," Letter to Guy S. Vissing, USNRC, from Robert C. Mecredy, Rochester Gas & Electric Corp., R.E. Ginna Nuclear Power Plant, July 30, 1999.

4. USNRC, *Inspection Manual Chapter 609, Appendix F, "Fire Protection Significance Determination Process,"* Washington, D.C., February 2005.

5. USNRC, *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities*, NUREG/CR-6850, Washington, D.C., September 2005.

[h] Or ~ 3E-6/y, following the preceding footnote.

BOUNDING RISK ANALYSIS FOR MULTIPLE SPURIOUS ACTUATIONS

ACRS Presentation on Generic Letter 2006-xx, *Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations*

BASELINE

- "Typical older NPP" -- fire CDF = 3.3E-5/y
  - IPEEE modeled "hot shorts" of MOVs and AOVs with probability = 0.1
  - 24 hot short basic events above truncation (CDF = 1E-10/y) contributed 0.0547 to fire CDF, or (3.3E-5/y)(0.0547) = 1.8E-6/y
    - Ten corresponded to five paired components, i.e., systemically symmetric in redundant trains
      - Contributed 0.0320 to fire CDF, or (3.3E-5/y)(0.0320) = 1.1E-6/y

BASELINE (cont'd)

- Assume that the components within each pair have similar failure characteristics and locations, including their cable runs
  - These comprise full set of candidates for multiple spurious actuations (hot shorts), not specifically modeled in traditional fire IPEEEs
    - Perform bounding analysis to estimate potential maximum CDF due to multiple spurious actuations for this typical older NPP

HOT SHORT CUT SETS

- Per pair, one hot short corresponds to train A and other to train B, and appear in symmetrically paired cut sets, as follows:
  - $CDF_A = F_A \bullet A_{HOT} \bullet \sum(B_{j,RANDOM} \bullet X_{j,RANDOM})$
  - $CDF_B = F_B \bullet B_{HOT} \bullet \sum(A_{k,RANDOM} \bullet Y_{k,RANDOM})$
    - CDF_i = fire CDF from cut sets with train i hot shorts
    - F_i = fire initiator that induces hot shorts in train i
    - A, B = hot short or random (non-fire-induced) failure of one of paired components in train A or B
    - X, Y = random failures of other cut set components

SIMPLIFICATIONS

- IPEEE assumed $A_{HOT} = B_{HOT} = 0.1$
- Express $\sum(B_{j,RANDOM} \bullet X_{j,RANDOM}) = \underline{B}_{RANDOM} \bullet \sum(X_{j,RANDOM})$, where $\underline{B}$ is an average
  - Likewise for A and Y, based on symmetry
- From symmetry, assume $\underline{A}_{RANDOM} = \underline{B}_{RANDOM}$ and $\sum(X_{j,RANDOM}) = \sum(Y_{k,RANDOM})$
- Total fire CDF from 10 paired hot shorts:
  - $CDF_A + CDF_B = (F_A + F_B) \bullet A_{HOT} \bullet \underline{B}_{RANDOM} \bullet \sum(X_{j,RANDOM}) = 1.1E\text{-}6/y$

SIMPLIFICATIONS (cont'd)

- Rewrite previous as $(F_A + F_B) \bullet \sum(X_{j,RANDOM}) = (1.1E\text{-}6/y)/(A_{HOT} \bullet \underline{B}_{RANDOM})$
  - Since A_{HOT} is fixed (0.1), right-side ratio is minimum when $\underline{B}_{RANDOM}$ is maximum
    - Typical component random failure probabilities are ≤ 0.001, so assume maximum $\underline{B}_{RANDOM} = 0.001$
  - Based on above assumed values
    - $(F_A + F_B) \bullet \sum(X_{j,RANDOM}) = (1.1E\text{-}6/y)/(0.1 \bullet 0.001) = 0.011$ [at a minimum]

MULTIPLE HOT SHORTS

- Dual failures of any of the 10 paired hot shorts would appear in cut sets as follows:
  - F_A initiates: $s \bullet F_A \bullet A_{HOT} \bullet B_{HOT} \bullet \sum(X_{j,RANDOM})$
  - F_B initiates: $s \bullet F_B \bullet A_{HOT} \bullet B_{HOT} \bullet \sum(Y_{k,RANDOM})$
    - Severity factor "s" reduces likelihood of more extreme fire deemed necessary for dual hot shorts
  - Based on previous equivalences:
    - $CDF_{PAIRS} = s \bullet (F_A + F_B) \bullet (A_{HOT})^2 \bullet \sum(X_{j,RANDOM})$
      - Since $(F_A + F_B) \bullet \sum(X_{j,RANDOM}) = 0.011$ [at a minimum], then $CDF_{PAIRS} = (0.011) \bullet s \bullet (A_{HOT})^2$ [at a minimum]

BOUNDING ANALYSIS

- Fire Protection SDP
  - 10% of fires produces 98th %ile heat release rate, characteristic of "extreme" fire → s = 0.1
  - Maximum hot short probability = 0.6
    - NUREG/CR-6850 (*Fire PSA Methodology*) reduces this to 0.3 for more "typical" case
      - Since "typical older NPP" likely has mix of thermoplastic and thermoset cables, assume A = 0.3
- **$CDF_{PAIRS} = 0.011 \bullet 0.1 \bullet 0.3^2 \approx 1E\text{-}4/y$**

CONCLUSIONS

- CDF estimate of 1E-4/y is considered bounding because of likely conservatism:
  - Closely located (within same cable tray?) cables for paired components in redundant trains A and B subject to dual hot shorts
  - Minimum, if any, cable protection against fire
    - 10% of fires severe enough for dual hot shorts
    - No fire development analysis or suppression credit
  - **Even if 10x too high, CDF is still significant**

At least for a "typical older NPP," one cannot *a priori* dismiss multiple hot shorts as being of low risk significance

BOTTOM LINE

BOUNDING RISK ANALYSIS FOR MULTIPLE SPURIOUS ACTUATIONS

ACRS Presentation on Generic Letter 2006-xx, *Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations*

ACRS Meeting
Rockville, MD
May 31, 2006

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- Rewrite previous as $(F_A + F_B) \bullet \sum(X_{j,RANDOM}) = (1.1E\text{-}6/y)/(A_{HOT} \bullet \underline{B}_{RANDOM})$
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    - Typical component random failure probabilities are $\leq$ 0.001, so assume maximum $\underline{B}_{RANDOM} = 0.001$
  - ✧Based on above assumed values
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BOTTOM LINE

At least for a "typical older NPP," one cannot *a priori* dismiss multiple hot shorts as being of low risk significance

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Presentation to ACRS
1:30 p.m. May 31, 2006
Generic Letter on
Inaccessible or Underground Cable Failures

Thomas Koshy, Senior Electrical Engineer
George Wilson, Chief, Electrical Engineering Br.
Michael Mayfield, Director, Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Agenda

- Purpose
- Safety Concerns
- Background
- Scope
- Requested Information
- Public Comments & Responses
- GL Modifications
- CRGR Endorsement
- NEI White Paper
- Questions

PURPOSE

❖ To obtain ACRS recommendation on the issuance of a generic letter to assess the operational readiness of inaccessible or underground power cables.

SAFETY CONCERNS

❖ Some of the underground /inaccessible cables supply power to safety related & risk significant equipment. (offsite power to safety buses, EDG feeder, Emergency service water pumps, etc.)

❖ Failure of one of these power cables from offsite power source could disable multiple systems.

❖ Since most of the cables in this application are not qualified for the moist environment, **there is an increasing possibility of more failures, and the possibility more than one cable failing simultaneously.**

Background

- Since 1989, 17 sites have experienced medium cable failures and over 100 medium voltage cables were replaced.
- Most of the faulty cables were not discovered until an operational failure occurred. **(EPRI Data indicated 65 cable failures with about half the plants reporting)**
- Licensee Event Reports (LERs) provided under 10 CFR 50.73 on cable failures do not reflect a representative sample. Most LERs are issued when the cable failure results in a plant shutdown or a RPS activation. LER is not required if a redundant component is still available to perform the required safety function.
- Most of the cable failures have two things in common:
  - The cable was about 12 years old
  - The cable was submerged or exposed to moisture for sometime
- A typical plant has 6 to 8 underground power cables that could cause significant safety challenge

Scope of Generic Letter

- Power cables that are within the scope of the maintenance rule, including cables connected to offsite power, emergency service water pumps, Emergency Diesel generator, Service water pumps etc.,

AND

- Those routed through underground or inaccessible locations such as buried conduits, cable troughs, above ground and underground duct banks

Benefits of a Monitoring Program

- Gain confidence in the capability of the cable to respond to design bases events of significant duration (Prevent failures during accident mitigation)
- Prevent unanticipated failures that cause plant transients
- Plan convenient outages for cable repair or replacement.

EXAMPLES

- OCONEE
  - Shows the benefits of using non-destructive cable testing data
  - The test showed that only 2 out 6 cables were degraded to the point of replacement
  - Developed a plan to track the degradation and replace it during a scheduled refueling outage.
- PEACH BOTTOM
  - Shows one extremely conservative approach of dealing with a cable failure was a global replacement of 60 cables over a 3 month period)
  - Testing could have detected the cables that did not need replacing
- OYSTER CREEK
  - Experience proves that just replacing cables does not prevent repeat failures
  - Shows the number of cable failures that do not meet the reporting criteria of LER required under 10 CFR 50.73

Requested Information

- Provide a history of inaccessible or underground power cable failures within the scope of maintenance rule
- Provide a description and frequency of all inspection, testing, and monitoring programs to detect the degradation of power cables used for EDG, offsite power, ESW, …
- If a monitoring or surveillance program is not in place, explain why a program is not necessary.

Public Comments

- Four Nuclear Utility Organizations
  - TVA, Progress Energy, AmerGen, Strategic Teaming and Resource Sharing (11 Nuclear Units: TXU Power, AmerenUE, Wolf Creek, Pacific Gas & Electric, STP Nuclear Operating Company, Arizona Public Service Company)
- One Industry Organization
  - NEI
- Two submittals from a cable testing company
  - Imcorptech

Public Comments & Responses

- Comment: Cable failures are random and therefore no NRC action is required.
  - **Response:** Cables failures in inaccessible and underground application have been at a higher rate based on available data and the plant impact. Therefore the GL requests info. on failure rate and monitoring programs
- Comment: Low Voltage cables and cables included in the maintenance rule should not be within GL scope
  - **Response:** The scope is limited to power cables that have the most plant safety impact were included in the scope. Staff has knowledge of some plants to have 480V power cables for EDGs and ECCS equipment, and DC power cables that had failures.

Public Comments & Responses

- Comment: Surveillance tests are adequate testing for cables
  - **Response:** Brief cycles of operation during surveillance testing or maintenance cannot detect power cable insulation degradation. A program that can detect the degradation could prevent unanticipated failures while responding to design bases events. Surveillance tests could be considered adequate for instrumentation cables that operate at much lower voltage and current in relation to its insulation rating.

Public Comments & Responses

- Comment: Regulatory Basis for the cable monitoring
  - **Response:** Primarily, 10 CFR 50 Appendix A Criterion 18, "... inspection and testing of important areas .... such as wiring ... assess the continuity of the systems and the **condition** of their components."
- Comment: What is the basis for considering multiple cable failure?
  - **Response:** At Davis Besse an insulation degradation created a transient that faulted at 13.8kV causing the trip of a circ. water pump and two substations. This event was discussed to illustrate the potential for fault current to cause further equipment or cable failures.

GL Modifications

- Most changes were of editorial nature
- Revised the scope to include above ground and below ground duct banks
- Removed broadband impedance spectroscopy as an available technique for testing
- Revised the requested information to include manufacturer, type of service and date of failure to identify repeated failures
- Revised the time for information collection to 60 hrs. from 40 hrs.

CRGR Endorsement

- CRGR has reviewed the GL and provided two comments
  - Specify the focus on power cables
  - Add an example of safety related cable failure that had a significant plant impact
- Changes were incorporated into the package sent to ACRS

NEI White Paper (6.6)

- Graded approach for monitoring and replacement of cables
  - Many cables do not power safety-related or important to safety loads
  - Graded approach to replacement and monitoring is best for safety and business reasons

Response: Cables within the scope of GL are within the scope of 10 CFR 50.65 Maintenance rule (safety-related systems and their support systems, accident mitigation systems, systems that could fail to cause a scram) and therefore, classified as most significant because of greater reliance for preventing plant transients or mitigating an accident.

NEI White Paper (8)

- Recommendations
  - Provide dry environment
  - Prepare for cable failures
  - Share failure resolutions

Response: Providing dry environment with periodic pumping out could increase cable life but not prevent failures. These cable failures could affect many systems and cable replacement is not practical in few hours.

Failure in cable service can be prevented with the use of current technology to ensure continued operation of accident mitigation systems and prevent plant transients

GL on Inaccessible Or UG Cables

- Questions?

Proposed License Renewal Interim Staff Guidance LR-ISG-2006-01: Plant-Specific Aging Management Program for Inaccessible Areas of BWR Mark I Steel Containment Drywell Shell

Presented by
Linh Tran, Division of License Renewal
Hans Ashar, Division of Engineering
May 31, 2006

Purpose

Provides guidance for future applicants with BWR Mark I steel containment design on the information that should be included in the license renewal applications

Proposed LR-ISG

- Imposes no new technical requirements
- Identifies information needed in LRA for staff to perform its review

Concerns

- Water seeping through inaccessible spaces

- Staff issued numerous requests for additional information to obtain the information

Recommendation

Plant-specific aging management program for inaccessible areas of Mark I steel containment drywell shell

Recommendation (con't)

- The applicant should:
  - Develop a corrosion rate
  - Demonstrate that responses to GL 87-05 consistent with developed corrosion rate
  - Provide evaluation that address the inaccessible areas if corrosion is identified in the accessible areas

Recommendation (con't)

- Demonstrate that moisture accumulation does not exist in the exterior portion of the drywell shell

- If moisture is suspected in the inaccessible areas:
  - Include SSCs identified as source of moisture in scope

Recommendation (con't)

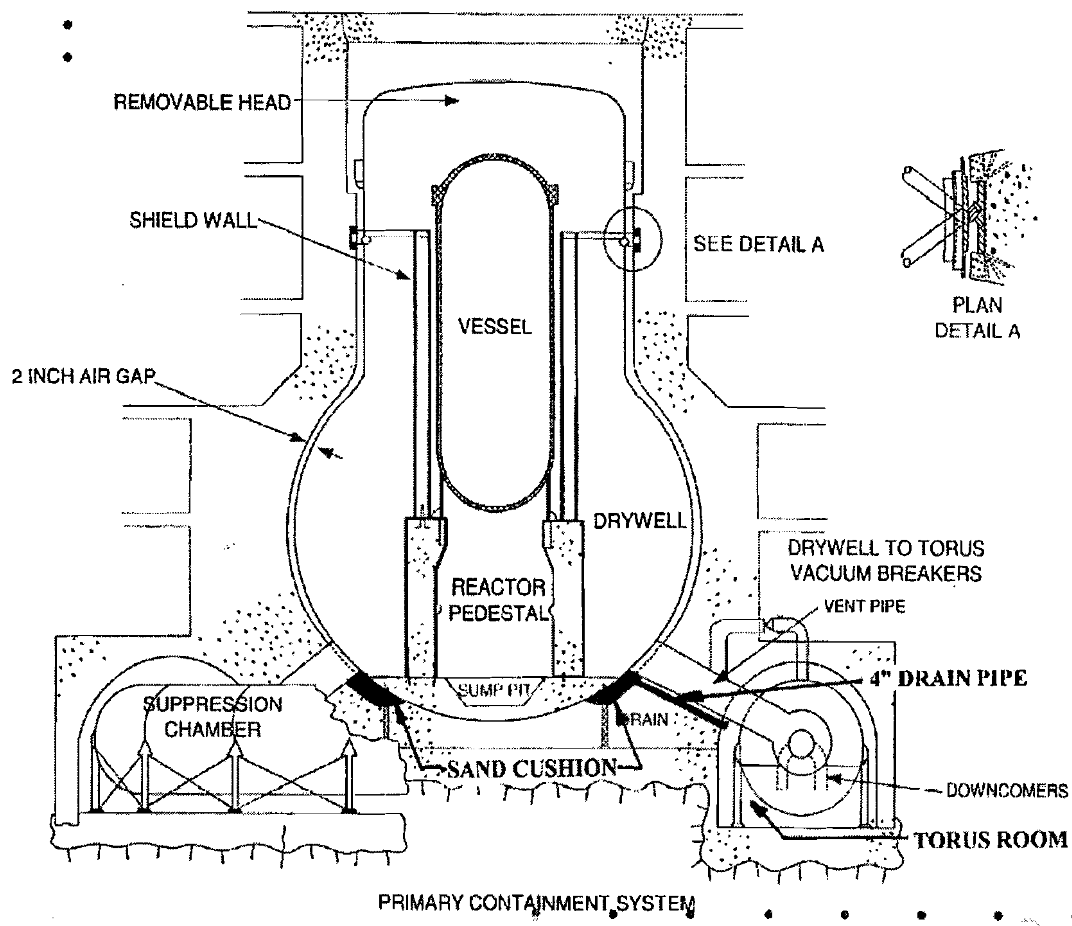
- Describe the augmented inspection plan, in accordance with ASME Section XI, Subsection IWE
- Demonstrate corrosion is not occurring or is at a manageable rate

– Identify actions that will be taken to ensure that the intended function of drywell shell will be maintained

Status of Review

Review Completed	Review in Progress	Future Review
Dresden 2, 3	Nine Mile Point 1	Cooper
Hatch 1, 2	Brunswick 1, 2	Duane Arnold
Peach Bottom 2, 3	Monticello	Fermi 2
Quad Cities 1, 2	Oyster Creek	FitzPatrick
Browns Ferry 1, 2, 3	Vermont Yankee	Hope Creek 1
	Pilgrim 1	

Typical Containment Drywell



UNITED STATES NUCLEAR REGULATORY COMMISSION

Challenges and Strategies for Licensing New Reactors

Phillip M. Ray
John R. Tappert
Division of New Reactor Licensing
Office of Nuclear Reactor Regulation

June 1, 2006

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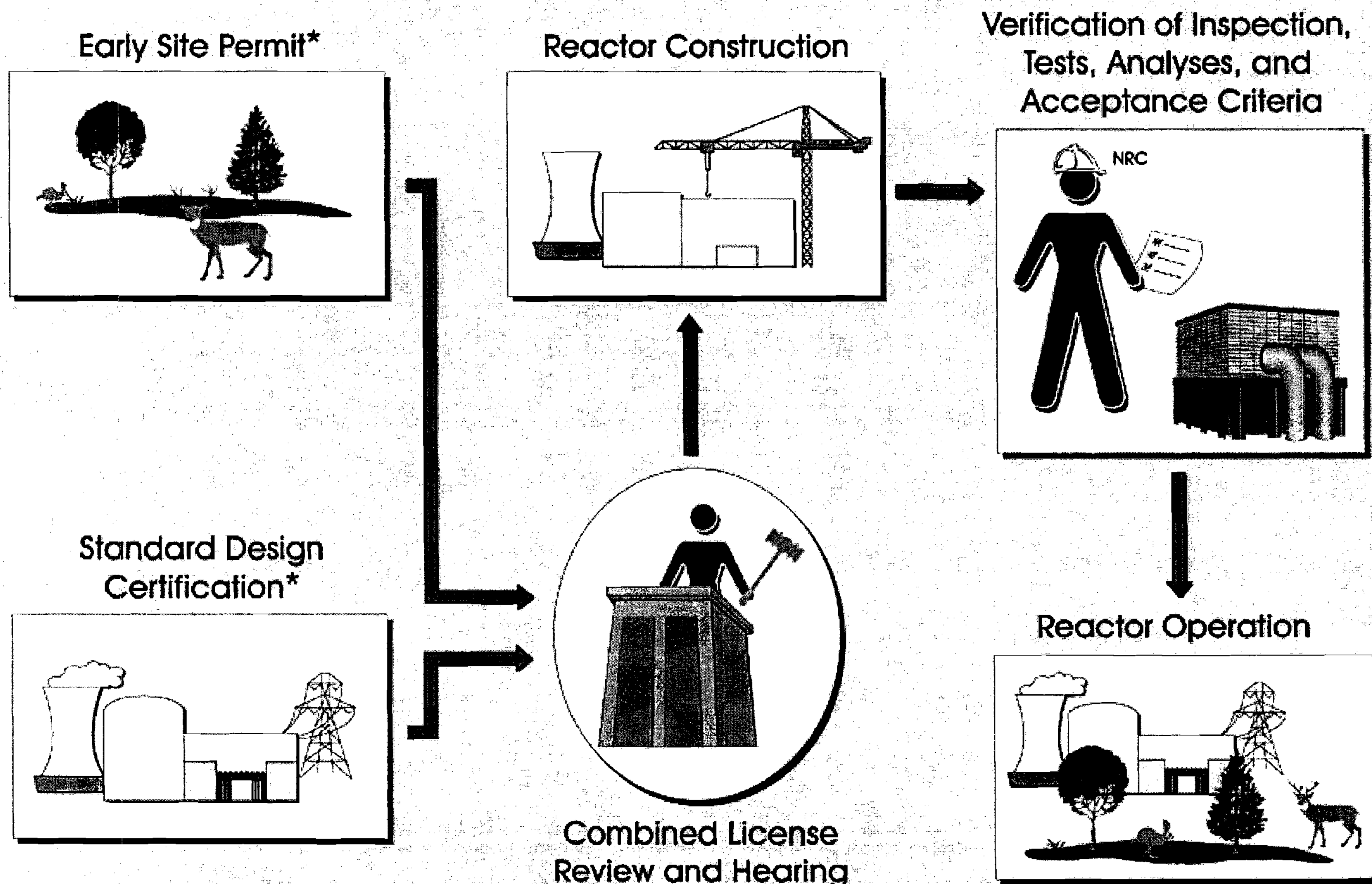
Topics for Discussion

- Challenges
  - Level of expected licensing activities
  - Schedules and expectations
  - Resources
- Strategies for new reactor licensing activities
- Key infrastructure development activities

Combined Licenses, Early Site Permits, and Standard Design Certifications



** or equivalent process*

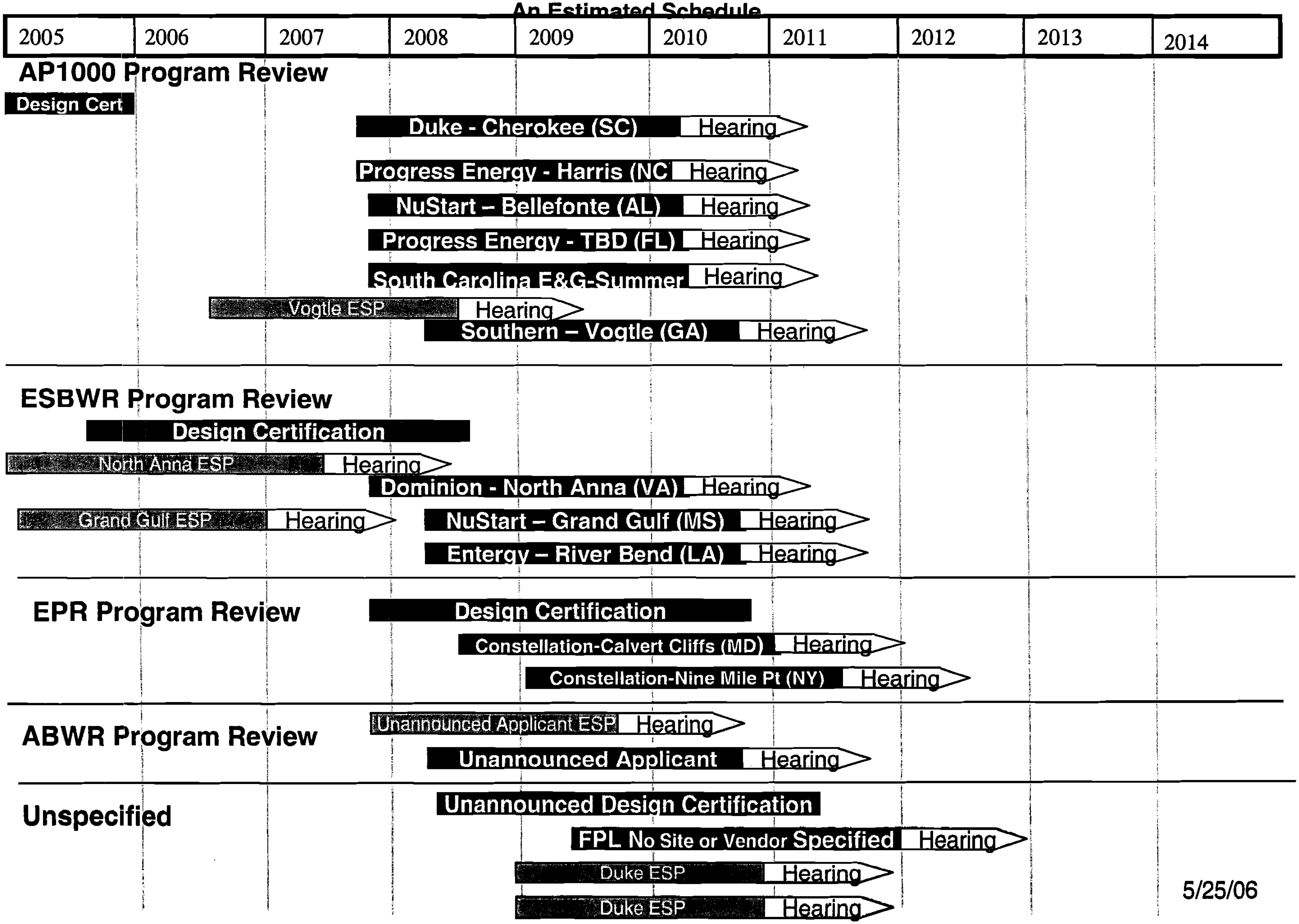
New Reactor Licensing Activities Forecasted

(As of May 19, 2006)

Organization: Docket Number or Project Number	Sites Under Consideration (# Units)	Applications Planned Submittal
AP 1000 Certified Design		
Duke (742)	Cherokee (2)	COL: Late 2007 or Early 2008
NuStart (740)	Bellefonte (2)	COL: 4th Qtr 2007
Progress Energy (738)	Shearon Harris (2)	COL: Sept or Oct 2007
	Florida (2)	COL: Late 2007 or 1st Qtr 2008
South Carolina Electric and Gas (743)	Summer(2)	COL: 3rd Qtr 2007
Southern Nuclear Operating Company (737)	Vogtle	ESP: 8/2006 COL: 3/2008
ESBWR Design		
Dominion (741)	North Anna	ESP: submitted in 2003 COL: 2007
Entergy (745)	River Bend	COL: Early 2008
NuStart (744)	Grand Gulf	ESP: submitted in 2003 COL: early 2008
EPR Design		
Constellation (746)	Calvert Cliffs Nine Mile Point	COL: 6/2008 and 6/2009
ABWR Certified Design		
Unannounced	TBD (2)	ESP: 3rd Qtr 2007 COL: TBD
Unannounced Design		
Florida Power and Light	TBD	COL: 2009
Duke	Davie County, NC Oconee County, SC	ESP (TBD) ESP (TBD)
Advanced Technology		
Unannounced Applicant	N/A	Design Certification: 1st Qtr 2008

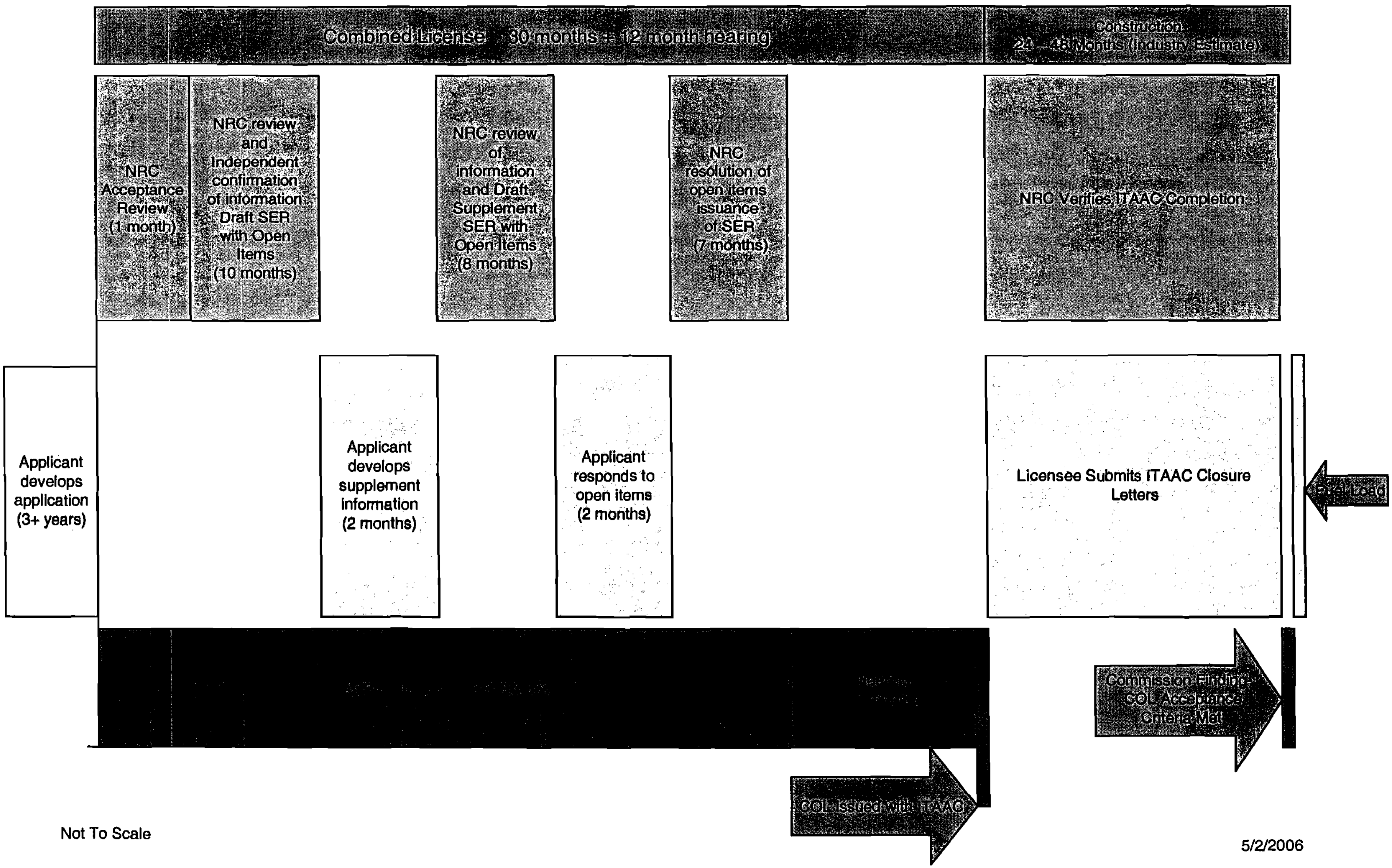
New Plant Licensing Applications

An Estimated Schedule



5/25/06

Combined License
Safety Review Process



Not To Scale

5/2/2006

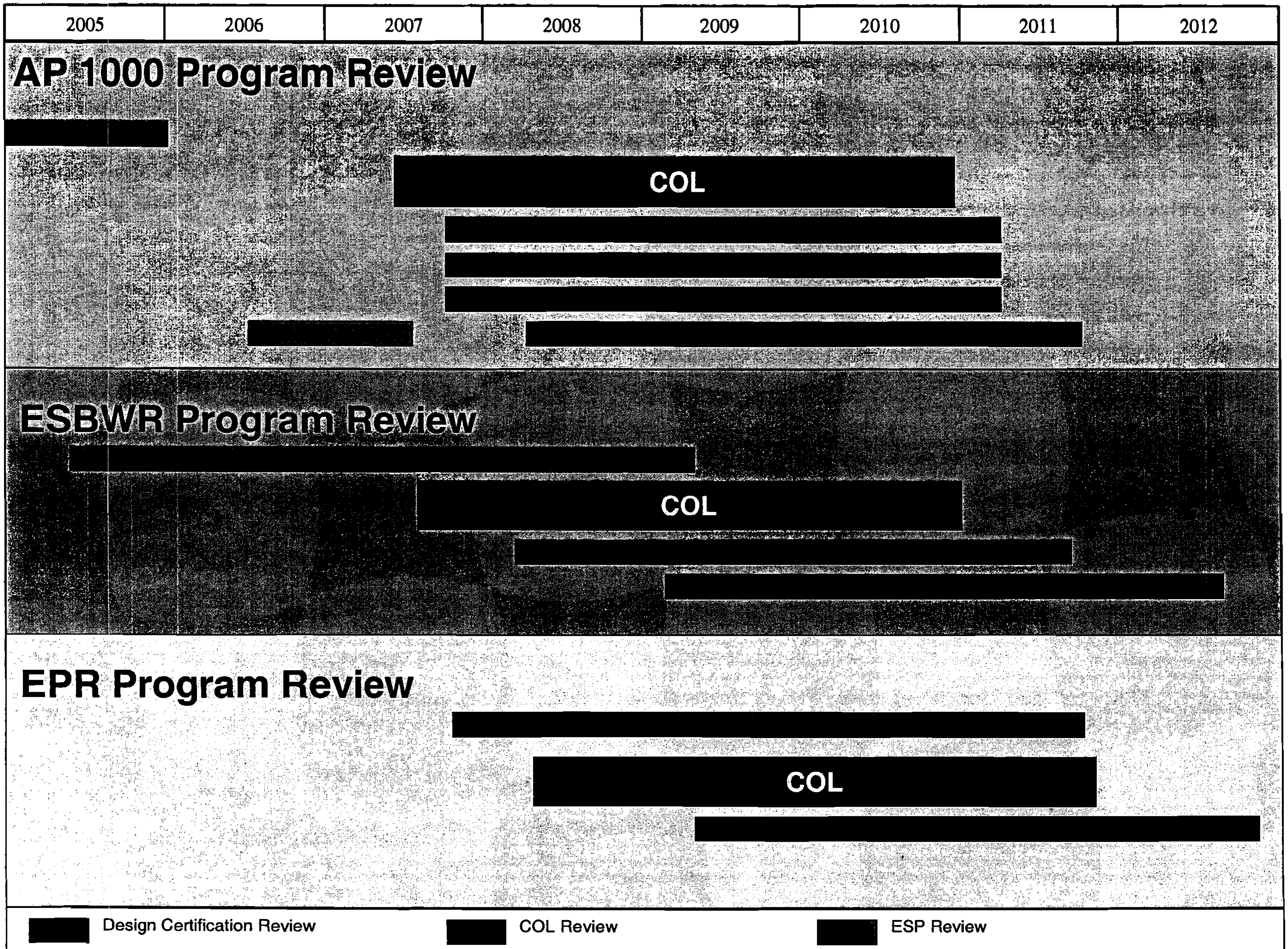
NRC Human Capital

- Recruiting
- Training
- Knowledge Management

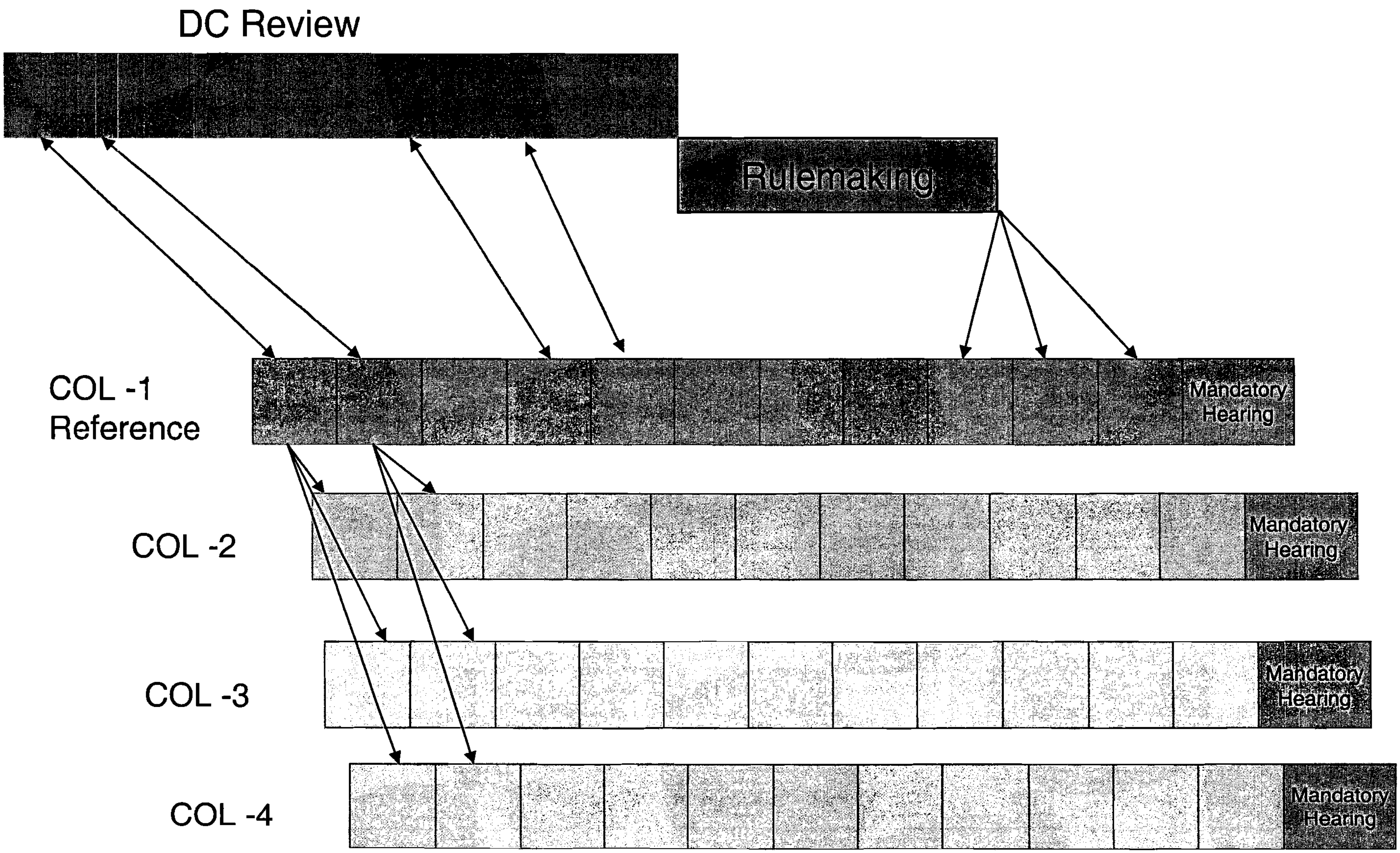
Strategies for New Reactor Licensing

- **Design Centered Approach**
  - Maximize Standardization
- **Optimize Review Process**
  - Infrastructure Development
  - Detailed Planning
  - Pre-application Reviews
  - Accountability - quality and schedule
  - Management Attention
- **Increase Qualified Resources (internal and external)**

Design-Centered Review Approach



One Decision – Multiple Applications



Similar approach used on site reviews (environmental and safety)

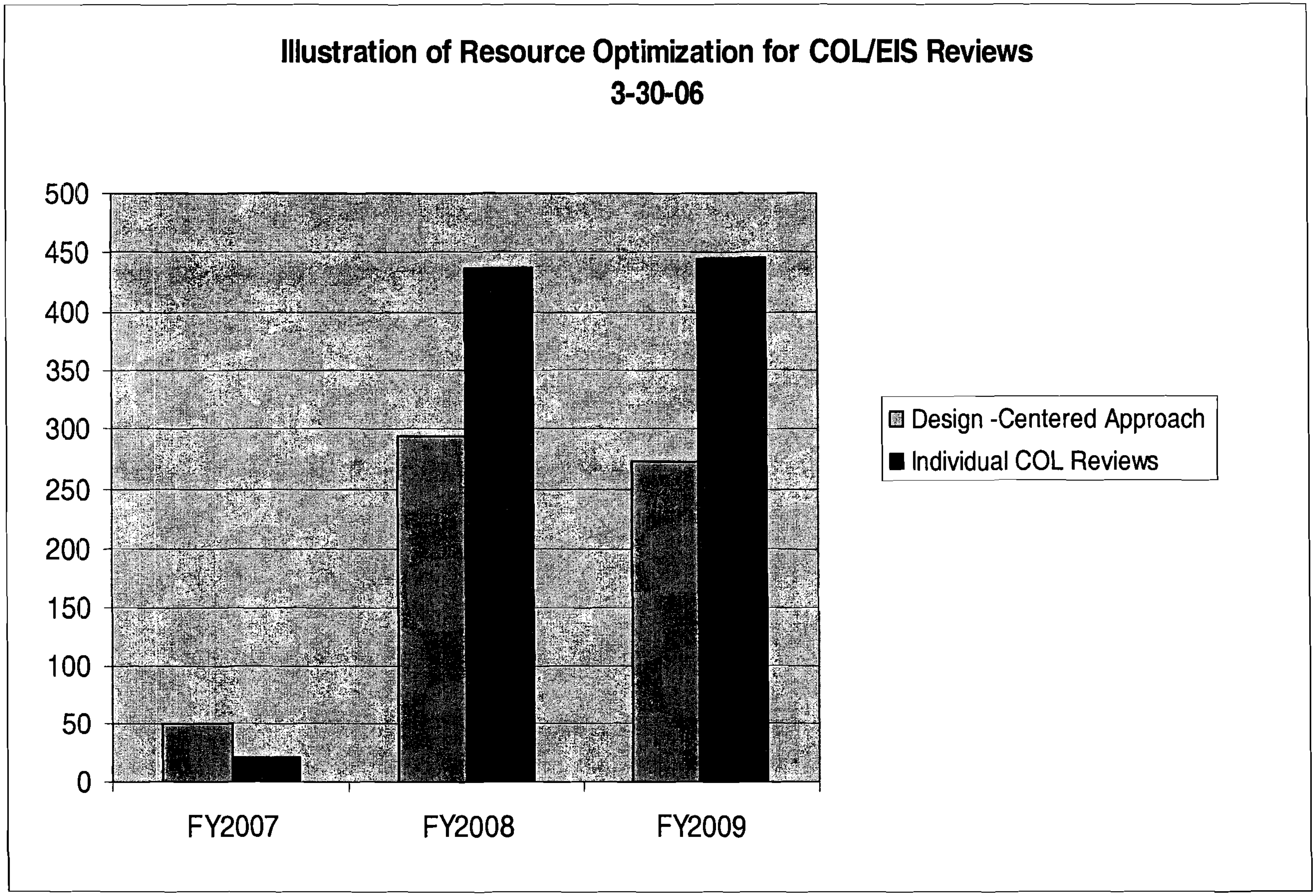


Illustration of Resource Optimization for COL/EIS Reviews
3-30-06
500
450
400
350
300
250
200
150
100
50
0
FY2007
FY2008
FY2009
Design -Centered Approach
Individual COL Reviews

Key Infrastructure Activities

- Develop Combined License (COL) Regulatory Guide
- Update and Revise NUREG-0800, "Standard Review Plan"
- Develop Construction Inspection Program

Draft Guide DG-1145

Combined License Applications for Nuclear Power Plants (LWR Edition)

Objective: Regulatory guide which provides application content and process guidance for combined license applications submitted under 10 CFR Part 52 that reference a certified design, an early site permit, both, or neither

Structure: 4 Parts - estimate length: 500 total pages

- Standard form and content
- Additional Information
- Applications referencing certified designs or early site permits
- Miscellaneous topics related to part 52 processes and application content

DG-1145
(continued)

Schedule

- Started January 2006
- Individual sections being drafted by new reactor staff, technical branches, contractor
- Post on NRC website as technical content sections receive branch concurrence
- Accept public comments on these “draft work-in-progress” sections
- Discuss in subsequent monthly public workshops (first workshop March 15)
- Draft scheduled for June 2006
- Final guide to accompany final Part 52 rulemaking

DG-1145
(continued)

Benefits:

- Providing COL applicants avenue to dialogue with NRC on COL applications
- Identifying needs for standard review plan updates
- Informing design certification review
- Preparing staff for review of COL applications

Challenges:

- Supporting monthly public meetings
- Completeness of draft sections of guide
- Defining scope of COL review
- Scope and timing of first-of-a-kind engineering (FOAKE) inspections
- Environmental finality of EIS from ESP in COL application review

SRP Update
Revised Plan

- SRP by March 2007
  - SRP section revisions need to be in effect 6 months prior to the docket date of an application - 10 CFR 50.34(h)
  - Design centered approach increases need to have guidance in place at the start of the COL and remaining DC reviews
  - COL applications anticipated starting September 2007

SRP Update
(continued)

- Issuance of LIC-200, Rev 1, May 8, 2006
  - Detailed instruction on "how to" do the update
  - Scoping process on front end
  - Issuing SRP section revisions for use and comment instead of Draft SRP sections
- Opportunities to engage ACRS
  - Use scoping results to facilitate planning
  - Evaluate which sections ACRS would like to consider and timing of consideration

Regulatory Guides

- Coordinate with SRP updates
- Prioritize Regulatory Guides to be updated
  - Affirm RGs which do not need to be updated
  - Scheduling those which can be completed within the March 2007 timeframe
  - Qualifying RG applicability within the specific SRP sections, as appropriate
  - Identifying RGs in which the technical basis is still being developed beyond March 2007

Construction Inspection Program Development

Mary Ann M. Ashley
Team Leader, Construction Inspection Program Development
Division of Inspection and Regional Support, NRR

CIP Development Challenges

- Inspecting construction activities worldwide
- Scheduling inspections based on licensee schedules
- Informing our ITAAC inspection sample selection
- Linking construction activities to ITAAC
- Linking an inspection finding to an ITAAC
- Issuing timely inspection reports
- Ensuring an Enforcement Policy that reflects Part 52 needs
- Ensuring the inspection staff is adequately trained

CIP Program Structure

- Inspection Manual Chapters – keyed to organization of approach in 10 CFR Part 52
  - IMC-2501: Early Site Permits
    - 5 Inspection Procedures - issued and in use
  - IMC-2502: Inspections to Support Issuing a COL
    - 9 Inspection Procedures issued and ready for use
  - IMC-2503: Inspections of ITAAC-Related Work
    - 25 Inspection Procedures – addressing the specific attributes of the different kinds of ITAAC – to be issued over next 12 months
  - IMC-2504: Inspections of Non-ITAAC Work
    - Approximately 150 Inspection Procedures – addressing the programs and processes common to all work activities; pre-op and startup testing; operational programs – to be issued over the next 18 months

CIP Inspector Resources

- Three types of inspectors are needed
  - Off-site fabrication inspectors
  - Construction resident inspectors
  - Construction specialist inspectors
- Inspectors are needed at different times
  - Off-site fabrication
  - Resident – 2 years after COL application submitted
  - Specialist – when on-site construction begins
- How many inspectors are needed?
  - Up to 3 full-time resident inspectors needed for each plant and 1 full-time scheduler with technical knowledge
  - Up to 3 specialist inspectors are needed for each plant

Conclusions

- NRC is preparing for an unprecedented level of new reactor licensing activity
- In order for the NRC to accomplish our mission to ensure adequate protection of public health and safety for new reactors licensed under 10 CFR Part 52 (given resource constraints, schedule pressures, and stakeholder expectations), a standardized, uniform, design entered approach to both COL application development and NRC review is essential.

(8)

G:\pphandout.wpd

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
533^{rd}	10	10.1

Title: PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES

Authors:
JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

10

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

From Staff Person
JOHN T. LARKINS

INTERNAL USE ONLY

SUMMARY/MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
May 30, 2006

The ACRS Subcommittee on Planning and Procedures held a meeting on May 30, 2006, in Room T-2B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 11:00 a.m. and adjourned at 12:15 p.m.

ATTENDEES
G. Wallis
W. Shack
J. Sieber

ACRS STAFF
J. T. Larkins
S. Duraiswamy
H. Nourbakhsh
M. Afshar-Tous
R. Caruso
J. Flack
E. Thornsbury
M. Junge
D. Fischer
M. Snodderly
A. Thadani
R. Savio
S. Meador
J. Gallo

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

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

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through September 2006 is attached (pp. 8-10). The objectives are to:

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

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action (pp. 11-12).

RECOMMENDATION

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

3) Appointment of New Members to the ACRS

On May 9, 2006, the ACRS Member Candidate Screening Panel sent a list of candidates to the Commission, recommending appointment of three new members to the ACRS. The Commission has unanimously approved the Panel's and Committee's recommendation on appointing three new members to the ACRS, subject to security and conflict of interest reviews. We are awaiting final SRM on this matter. These individuals will be invited to the July 2006 ACRS meeting as invited experts. In his vote sheet, Commissioner Lyons endorsed the ACRS/ACNW work on Knowledge Management (pp. 13).

The Subcommittee commends the ACRS Executive Director on his recruitment of new members and facilitating the appointment process. Additionally, the Subcommittee commends the ACRS staff's work on Knowledge Management.

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director keep the committee informed of further developments on this matter.

4) Quadripartite Meeting Status

On March 31, 2006, all ACRS abstracts for the 2006 Quadripartite meeting were uploaded to the web site. During the April ACRS meeting, these abstracts were provided to the members for review. During the June meeting, the members will be provided with copies of the abstracts from Germany and Japan. The members are reminded that final papers and power point presentation slides are due by Friday, July 28, 2006.

In addition, several meeting attendees from some of the Member Countries are scheduled to visit the Calvert Cliffs Nuclear Plant on October 17, 2006. So far, Armijo, Maynard, Sieber, and Wallis plan to attend.

On July 5, 2006, letters will be sent to the Commissioners, EDO, and NRC Program Office Directors, inviting them to participate/attend the Quadripartite Meeting.

RECOMMENDATION

The Subcommittee recommends that the members make sure that the papers and presentation slides are completed by July 28, 2006. Also, other members who could participate in the visit to Calvert Cliffs Nuclear Plant should inform Mugeh and/or the ACRS Executive Director.

5) Streamlining the NRR Rulemaking Process

In a memorandum (COMEXM-06-0006) dated April 7, 2006 (pp. 14-15) Chairman Diaz and Commissioner McGaffigan sent a proposal to Commissioners Merrifield, Jaczko, and Lyons for streamlining the NRR Rulemaking Process. In that memo, it is stated that ". . . not withstanding 10 CFR 2.809 and the Memorandum of Understanding between the ACRS and the EDO, the staff may waive review by the ACRS at the proposed rule stage." Also, it is stated "comments from the ACRS may be submitted to the Commission either during the comment period for the proposed rule, or following the close of the public comment period, but prior to issuance of the final rule."

If implemented, this proposal will limit the number of opportunities that the ACRS has now to review a proposed rule. Also, this will contradict Commission direction in previous SRMs. For example, in the April 5, 2000 SRM, the Commission stated that the ACRS should work with the NRC staff to enhance efforts to risk-inform 10 CFR Part 50, including Appendices A and B.

Also, in the April 13, 2006 SRM, the Commission stated that the ACRS and the staff should continue to work together to ensure that staff and ACRS reviews of important technical issues are coordinated in a manner to ensure timely resolution of these issues.

Without involvement by the ACRS in the early stages of the development of a proposed rule, the Committee may not be able to contribute effectively to the development of a rule. During the survey of the NRC staff related to 2005 self-assessment of ACRS, some NRC staff members stated that "Early interaction by the ACRS with the EDO and the NRC staff on the regulatory significance of complex technical issues was very useful."

The ACRS Chairman, the Executive Director, and Deputy Executive Director contacted Commissioner McGaffigan on May 3, 2006 and provided comments on a draft version of the SRM. On May 31, 2006, a final SRM related to this matter has been issued (pp.15A-15C). The final SRM, which has been significantly changed compared to the previous version, incorporates the comments provided by the ACRS Chairman.

6) Visit to the Limerick Nuclear Plant and Meeting with the Region I Administrator

During the May ACRS meeting, the members decided to meet with the Region I Administrator on July 26 and visit the Limerick plant on July 27, 2006. The following members have agreed to participate:

Wallis	Armijo
Sieber	Maynard
Shack	
Powers	

A list of discussion topics proposed by Mr. Sieber is attached (pp. 16).

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the propose list of discussion topics.

7) ACRS Meeting with the NRC Commissioners

The ACRS is tentatively scheduled to meet with the NRC Commissioners on Thursday, September 7, 2006, between 9:30 and 11:30 a.m. The Committee needs to approve a list of topics during the June meeting. Topics proposed by the Planning and Procedures Subcommittee are as follows:

I. Overview

- License Renewal
- Power Uprate
- Risk-Informing 10 CFR 50.46
- Future Activities

II. PWR Sump Performance

III. Safety Research Program Report

IV. Safety Conscious Work Environment/Safety Culture

V. Future Plant Design Activities

- ESP
- 10 CFR Part 52
- ESBWR

RECOMMENDATION

The Subcommittee recommends that the Committee approve a list of topics and that the ACRS Executive Director send the topics proposed by the Committee to the Commissioners for feedback.

In view of Dr. Klein becoming the NRC Chairman in July 2006, the ACRS Executive Director should explore the feasibility of postponing the ACRS meeting with the Commission to October or November 2006.

8) ACRS letter Writing process and Related Matters

In view of the recent experience, the Committee should discuss whether changes, if any are needed to make its letter writing process more efficient. The Committee seems to spend more time on unrelated issues during the preparation of its reports than is necessary. This can be particularly distracting when the Committee is working on a plant specific application (e.g., license renewal, power uprate) and generic type issues unrelated to the application are introduced. The Committee should make sure that the contents of the letters, including additional comments are relevant to the plant specific issues discussed. Including comments not related to the subject matter of a particular letter diminishes the value of the recommendation on the main issue.

RECOMMENDATION

The Subcommittee recommends that the Committee discuss this matter and propose a course of action for enhancing the letter writing process.

9) Staff Requirements Memorandum on Regulatory and Resource Implications of a DOE Spent Nuclear Fuel Recycling Program

In a Staff Requirements Memorandum dated May 16, 2006 (pp. 17-19), the Commission directed the staff to focus on the development of a conceptual licensing process for the Administrations's Global Nuclear Energy Partnership (GNEP) - related facilities. The conceptual process should consider the most effective and efficient elements of the NRC's licensing processes for major facilities, including review of the one-step licensing provisions for enrichment facilities, and features of the nuclear power plant combined licensing under 10 CFR Part 52 (i.e., construction authorization and operating license hearing process, design certification process, and early site permit process). The development of a conceptual licensing process is an inter-office undertaking, likely with NMSS in the lead, but NRR, NSIR, RES, and OGC all having significant roles. The Commission stated that the ACRS and ACNW could also help in defining the issues most important to licensing, inspecting, and ultimate decommissioning of reprocessing facilities (and related fuel-cycle facilities).

RECOMMENDATION

The ACNW currently has lead responsibility on enrichment, fuel cycle and reprocessing facilities, however, the Subcommittee recommends that the Committee propose a course of action to support the ACNW in defining the issues noted above.

10) SRM related to an expanded work scope for the Center for Nuclear Waste Regulatory Analyses

In a SRM dated February 9, 2006 (pp 19-20), the Commission tasked the ACNW with providing recommendations to the Commission, with input from the ACRS, on how the Center for Nuclear Waste Regulatory Analyses (CNWRA) might broaden its assistance to the NRC. The CNWRA is part of the Southwest Research Institute and the ACNW has been providing advice to the Commission on the CNWRA's programs for a number of years. The work at CNWRA is being managed by NMSS as a Technical Assistance Program, and at present it is primarily focused on providing support for NRC activities associated with the proposed Yucca Mountain repository. NMSS, NRR, and RES are aware of the CNWRA's capabilities and are considering possible additional use of the CNWRA's expertise. Dana Powers, as the ACRS lead member on the review of the NRC's research programs, is being kept informed of the ACNW activities in developing a response to the Commission request. ACNW plans to have CNWRA representatives brief the ACNW on the status of CNWRA's current programs on July 19,2006 (for approximately 2 hours) and to invite NMSS, NRR, and RES to the ACNW's July 19 meeting to discuss these offices' views on broadening the CNWRA's assistance to the NRC. It is anticipated that this last discussion would take about 1½ hours. Since Dr. Powers will not be able to attend this meeting, other interested ACRS members are invited to attend.

RECOMMENDATION

The Subcommittee recommends that Dr. Savio keep Dr. Powers informed of the ACNW discussions with the staff at the July 19, 2006 ACNW meeting. Other ACRS members who wish to attend should inform Dr. Larkins or Dr. Savio.

11) Seminar on 9/11 Event

The staff plans to hold a seminar on September 11, 2006 during which representatives from NIST will present the results of their analysis of the 9/11 event, specifically the impact damage, the fire effects on structural steel, and the collapse of the twin towers (pp. 21). The staff invites interested ACRS members to attend this seminar. Also, the staff would like to know whether any ACRS members are interested in visiting the NIST Fire Research Facilities in Gaithersburg.

RECOMMENDATION

The Subcommittee recommends that video tape of this seminar be provided to the members. Any members who are interested in visiting the NIST Fire Research Facilities should inform the ACRS Executive Director or Mike Junge.

ANTICIPATED WORKLOAD
May 31 - June 2, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	—	Santos	Draft Final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems"	A	To support staff schedule	—
		Santos	Interim Staff Guidance on Aging Management Program for Inaccessible Areas of BWR Mark I Containment Drywell Shell **[INFORMATION BRIEFING]**	—	—	—
		Santos	**Subcommittee Report** - Interim Review of the Monticello License Renewal Application	—	—	—
Denning	—	Junge/Nourbakhsh	Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations"	A	To support staff schedule	—
Kress	—	Fisher/Snodderly	Overview of Advanced Reactor Activities **[INFORMATION BRIEFING]**	—	—	—
Wallis	—	Nourbakhsh/ Duraiswamy	Status Report on Quality Assessment of Selected NRC Research Projects	—	—	—

ANTICIPATED WORKLOAD
July 12-14, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	—	Thornsbury	Safeguards and Security Matters: State-of-the-Art Consequence Analysis **[CLOSED]**	—	—	—
		Thornsbury	**SUBCOMMITTEE REPORT** - Draft Regulatory Guide on Risk-Informed Digital System Reviews	—	—	—
Powers	—	Santos	Results of the Staff Study to support NRR Decisionmaking on the Need for Establishing Limits for Phosphate Ion Concentrations in Groundwater at the Sites of Plants Applying for License Renewal	B	To provide Committee's views	—
Shack	—	Nourbakhsh	Integrating Risk and Safety Margins	A	To provide Committee's views	—
Sieber	—	Junge/Santos	Final Review of the Nine Mile Point License Renewal Application and the Associated Final SER	A	To support staff schedule	—

ANTICIPATED WORKLOAD
July 12-14, 2006 (Cont'd)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Wallis	—	Caruso	Results of the Integrated Chemical Effects tests Related to the Resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance"	A	To provide Committee's views	—
	All Members	Larkins	Preparation for Meeting with the NRC Commissioners	—	—	—

ANTICIPATED WORKLOAD
September 7-9, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Snodderly/ Nourbakhsh	Draft Final NUREG on 10 CFR 50.46 LOCA Break Frequency Reevaluations	A	To support staff schedule	—
Bonaca	—	Santos	Final Review of the Monticello License Renewal Application and the Associated Final Safety Evaluation Report **[TENTATIVE]**	A	To support staff's accelerated schedule	—
Kress	—	Fischer	Draft Regulatory Guide, DG-1145, Guidance for COL Applications	A	To support staff schedule	—
Powers	—	Fischer	Lessons Learned from the Review of ESP Applications	B	To provide Committee's views	—
		Nourbakhsh	Draft Report on the Quality Assessment of the NRC Research Projects on Containment Capacity Study at SNL and Molten Core Concrete Interaction Study at ANL	—	To be completed in October	—
Shack	—	Thornsbury	Draft Reg. Guide, "Acceptance Criteria for ECCS for Light Water Nuclear Power Reactors," and Draft Final 10 CFR 50.46, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements"	A	To meet SRM schedule 9/29/06	—
Wallis	All Members	Larkins, et.al	Meeting with the NRC Commissioners	—	—	—

10

ACRS Items Requiring Committee Action

1 **<u>Screening Analysis for Generic Issue 197 "Iodine Spiking Phenomena"</u>**

Member: Richard Denning **Engineer:** Cayetano Santos

Estimated Time:

Purpose: Determine a Course of Action

Priority:

Requested by: NRR/RES D. Beaulieu x3243

The ACRS report dated May 21, 2004 described the Committee's review of the staff's progress in resolving steam generator tube integrity issues highlighted in NUREEG-1740 "Voltage-Based Alternative Repair Criteria." In this report the Committee noted that the staff treats iodine spiking conservatively and recommended "that the staff develop a mechanistic understanding of iodine spiking so that analyses reflect current plant operations and the capabilities of modern fuel rods." The EDO response dated August 25, 2004, stated that generic issue 197 "Iodine Spiking Phenomena" has been identified to address the Committee's recommendations. The staff made a commitment to send the ACRS a copy of the screening analysis of this generic issue. The results of the initial screening of Generic Issue 197 is documented in a May 8, 2006 memorandum from Jennifer Uhle to Carl Paperiello. The review panel concluded that this issue does not represent a new safety concern and recommended that the issue be dropped from further consideration.

The Planning and Procedures Subcommittee recommends that Dr Denning propose a course of action on this matter.

11.

2 **<u>Draft Final Regulatory Guide DG-8028, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants"</u>** (Open)

Member: John Sieber **Engineer:** Michael Junge

Estimated Time 2 hours

Purpose: Determine a Course of Action

Priority: Medium

Requested by: RES Harriet Karagiannis

In a memorandum dated July 20, 2005 (ADAMS Accession No. ML052100152), RES requested the ACRS to review and comment on draft Regulatory Guide DG-8028, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," (ADAMS Accession No. ML051670280). The Committee stated in a Memorandum dated September 23, 2005, that it plans to review the draft Regulatory Guide DG-8028 after reconciliation of public comments. The Committee had no objection to the staff's proposal to issue the proposed draft Regulatory Guide DG-8028 for public comment. The primary purpose of this proposed revision is to clarify the terminology related to the physical barriers that licensees could use to prevent unauthorized personnel access to high and very high radiation areas. This revision also includes two additional clarifications as well as several editorial and formatting changes. Changes as a result of the public comment period were not significant. None of these changes alter the regulatory positions established in the current version of the guide.

The Planning and Procedures Subcommittee recommends that Mr. Sieber recommend a course of action on this matter.

12

From: Andrew Bates
To: John Larkins
Date: 05/23/2006 1:51:27 PM
Subject: Membership

Commissioner Lyons' Comments on COMSECY-06-0027

Recommendation of the ACRS Member Candidate Screening Panel
for Appointment of New Members to the ACRS

I approve the recommendation to appoint Drs. Banerjee, Corridini, and Abdel-Khalik to the ACRS.

I understand that maintaining an appropriate balance of expertise among Committee members remains a guiding principle in the recommendation of new members. To meet the coming challenges of adding advanced reactor work to an existing operating plant workload, for which the Commission recently approved an expansion of the ACRS to its statutory limit of 15 members, I would expect consideration will be given to attaining an optimal balance of expertise based on what will best serve the needs of the Commission. In working to attract new members with the necessary specific areas of expertise, I further expect that the Committee and the Candidate Screening Panel will continue to value breadth of experience in each member, enabling each member to contribute effectively to a broad range of issues.

Indirectly related to this COMSECY, I would also like to make two additional comments. First, regarding the building and maintenance of a Committee knowledge-base, I was pleased to see the recent description of the ACRS's (and ACNW's) planned approach to knowledge management, aimed at retaining and making available historical Committee technical and non-technical information for current and future Committee members and staff. I was particularly pleased with the collaborative approach being taken with other Offices and stakeholders. As this system matures, it could potentially provide a very useful resource agency-wide.

Second, I commend the Committee for providing advice that I have found to be thoughtful, technically grounded, and well-articulated. I continue to highly value it and look forward to its continuation in meeting our future challenges.

/RA/ 5/18/06
Peter B. Lyons Date

13

COMNJD-06-0004/
COMEXM-06-0006

April 7, 2006

MEMORANDUM TO: Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons

FROM: Nils J. Diaz /RA/
Edward McGaffigan, Jr. /RA/

SUBJECT: STREAMLINING THE NRR RULEMAKING PROCESS

In light of increased rulemaking activities, which are only expected to grow in the near future, we believe it is of paramount importance to further enhance NRR rulemaking activities to improve efficiency and timeliness, while eliminating unnecessary burdens. Thus, we propose streamlining the rulemaking process by removing unnecessary constraints, while simultaneously enhancing transparency and public participation. There are several tools by which the agency can achieve these goals, including the following:

- At the discretion of the Director of NRR, and in consultation with the General Counsel, the staff may waive the development and submission of rulemaking plans;

- The staff may waive review by the Committee to Review Generic Requirements ("CRGR") at the proposed rule stage, and, notwithstanding 10 C.F.R. § 2.809 and the Memorandum of Understanding between the ACRS and the EDO, waive review by the Advisory Committee on Reactor Safeguards ("ACRS") at the proposed rule stage (as was done, for example, in the ongoing Part 52 rulemaking). Comments from CRGR should be limited to addressing, at the final rule stage, any public comments received relevant to backfit matters. Comments from the ACRS may be submitted to the Commission either during the comment period for the proposed rule, or following the close of the public comment period, but prior to issuance of the final rule.

- In addition, the staff may release proposed rule text for public review, and hold workshops, if necessary, prior to submission of the rule to the Commission. This has been successfully done in past rulemakings (*i.e.*, rulemakings associated with 10 CFR Parts 26, 35 and 70), and is done for most rulemakings by NMSS, at least with Agreement States. The early release of proposed rule text in concert with workshops should reduce or eliminate the need for extended public comment periods (i.e., those in excess of 75 days).

- An additional tool would be the widespread use of working groups and steering committees, designed to reduce the cumbersome concurrence process and eliminate duplicative management review.

We welcome additional mechanisms that the EDO, the General Counsel, or Director of NRR may develop for streamlining and increasing the transparency of the rulemaking process, thus

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allocating the appropriate level of resources for the most important rulemaking actions, and ensuring that the staff's hands are not tied by perceived or real procedural prerequisites that are unnecessary for a given rulemaking.

These mechanisms should be employed for any rulemaking actions where the Director of NRR sees a net benefit. For example, some of these mechanisms clearly would be appropriate for the pending 10 CFR § 50.68 direct final rule. These techniques will likely save resources, which, with the vastly expanded rulemaking agenda, are a significant concern for the agency. These actions are not intended to reduce any public involvement or eliminate processes mandated by the Administrative Procedure Act. Rather, we believe they will further empower all stakeholders.

The Director of NRR should examine all current and planned rulemakings to assess whether these techniques would be appropriate for current and anticipated rulemaking activities. Any additional mechanisms that would streamline the process further should be raised to the Commission for consideration.

Moreover, we are concerned with contractor dependence in completing our rulemaking activities. Contractors are heavily utilized in NRR rulemakings, including resolution of public comments and development of statements of consideration. With significant elements of the rulemaking process fundamentally outside of the agency's day-to-day control, both resources and schedules could be negatively impacted. The NRR staff, in consultation with OGC, should provide the Commission with a paper addressing the feasibility, as well as the advantages and disadvantages, of reducing contractor dependence in the rulemaking arena. In a related vein, the staff should address the option of OGC assisting in the allocation of resources prior to the proposed rule stage to help determine the most efficient use of resources. Furthermore, the staff should take necessary steps to ensure that, when contracting is needed, it is accomplished in a manner that best serves the needs of the agency; *i.e.*, in the most efficient and effective manner possible.

Finally, the staff should consider whether streamlining mechanisms can be usefully employed by other program offices that undertake rulemaking.

SECY, please track.

cc: A. Vietti-Cook, SECY
L. Reyes, EDO
G. Wallis, ACRS
K. Cyr, OGC
J. Dyer, NRR

15

May 31, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

Karen D. Cyr
General Counsel

FROM: Andrew L. Bates, Acting Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - COMNJD-06-0004/COMEXM-06-0006 - STREAMLINING THE NRR RULEMAKING PROCESS

The Commission has approved the following measures to improve the efficiency and timeliness of the rulemaking process. These measures should be implemented as soon as practicable.

1. The staff may waive the development and submission of rulemaking plans at the discretion of the Director of NRR, and in consultation with the General Counsel. When the staff determines that a rulemaking plan is necessary, the staff should consider options to develop additional efficiencies, such as making the rulemaking plan more concise (perhaps no more than a few pages), or providing a rulemaking plan through informal mechanisms such as Commission technical assistant briefings.

2. The staff may waive review by the Committee to Review Generic Requirements ("CRGR") at the proposed rule stage, and, notwithstanding 10 C.F.R. § 2.809 and the Memorandum of Understanding between the ACRS and the EDO, may waive review by the Advisory Committee on Reactor Safeguards ("ACRS") at the proposed rule stage (as was done, for example, in the ongoing Part 52 rulemaking). Comments from CRGR should be limited to addressing, at the final rule stage, any public comments received relevant to backfit matters. Comments from the ACRS may be submitted to the Commission either during the comment period for the proposed rule, or following the close of the public comment period, but prior to issuance of the final rule. While the Commission grants the staff permission to waive review by both committees at the proposed rule stage, due consideration should be given to the merits of earlier engagement with one or both committees, if the staff determines that such engagement will result in a more efficient and effective process for a particular rulemaking. When committee reviews are waived, the staffs of both committees should continue to be provided copies of the proposed rules and supporting documentation to keep them informed. The staff should work out suitable communication arrangements with ACRS to keep them informed of waivers of ACRS reviews at the proposed rule stage and to consider specific requests for earlier review opportunities. Nothing in this SRM should be construed as in any way discouraging open informal discussion of proposed rule documents with ACRS staff. The staff and ACRS should also work to coordinate schedules in order to enable timely and effective rulemaking.

15A.

3. The NRR staff may routinely release draft rule text, statements of consideration, and the technical basis for public review, and hold workshops, if necessary, prior to submission of a proposed rule to the Commission. Draft rule text has been released on a case-by-case basis for past rulemakings (*i.e.*, rulemakings associated with 10 CFR Parts 26, 35 and 70), and is done for most rulemakings by NMSS, at least with Agreement States. The early release of draft rule text and supporting documentation in concert with workshops should reduce or eliminate the need for extended public comment periods (i.e., those in excess of 75 days). The staff should notify the Commission of a planned release of draft rule text for public review prior to submission of the proposed rule to the Commission.

4. The Director of NRR, should examine all current and planned rulemakings to assess whether any techniques approved by the Commission via this COM, or that are already available via the staff's recently completed Rulemaking Process Improvement Implementation Plan would be appropriate to apply to ongoing rulemakings, or those that may begin in the current fiscal year. Any additional mechanisms identified by the staff that would help achieve the objectives noted above for the rulemaking process should be raised to the Commission for consideration.

5. The staff should continue using working groups as well as steering committees consisting of SES managers, as appropriate, to expedite the concurrence process and eliminate duplicative management review.

After taking the immediate steps described above, the staff should take the following action.

6. The Rulemaking Coordinating Committee should conduct an evaluation of the overall effectiveness of the just-completed interoffice Rulemaking Process Improvement Implementation Plan. All offices that are involved in the majority of the agency's rulemaking activities (i.e., NRR, NMSS, OGC, ADM, OIS) should participate in this assessment to determine if those improvements, as well as the additional improvements described above, have succeeded in streamlining agency rulemakings. The Committee and participating offices should further seek to identify any other potential options that could streamline the rulemaking process (not only for NRR, but for other affected program offices).

   Further, as part of the evaluation, the staff, in consultation with OGC, should address the feasibility, as well as the advantages and disadvantages, of reducing contractor dependence in the rulemaking arena. The staff should also examine ways to improve early collaboration with affected offices, particularly OGC and OIS, regarding the allocation of resources prior to the proposed rule stage, to determine the most efficient use of resources. Furthermore, the staff should address the necessary steps to ensure that, when contracting is needed, it is accomplished in a manner that best serves the needs of the agency, i.e., in the most efficient and effective manner possible.

   The staff should provide the results of this evaluation to the Commission within approximately one year.

15B.

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

15c.

Question for Limerick Site Visit

For the Licensee:

1. Describe the relationship between you, as a licensee, and the NRC as the safety regulator. Are the regulatory issues brought to your attention important safety matters? Is the Agency fair and impartial? Are there safety issues that the Agency fails to address?

2. What are your greatest challenges? How will you meet those challenges?

3. Describe the protocols and relationship with your system operator, as related to grip stability issues. What steps are you taking to improve grid stability protocols. What steps are the System Operator taking to improve grid stability. Has grid stability issues caused you to operate your plant in a manner different from normal operation at 100% power.

For Region I Management:

1. Do you have sufficient experienced staff resources to carry out your responsibilities? Describe your current number of vacancies, you plans to fill vacancies and your new employee training programs.

2. What are the most significant challenges the Region faces in the short term and long term?

3. Describe the functions where you have difficulty meeting Agency goals for timeliness and throughness. How do you plan to improve in these areas?

4. How would you improve the Reactor Oversight Process?

5. How would you improve the baseline inspection program?

6.; How can the ACRS contribute to your accomplishing your mission?

G:\ACRS LTRS\1 Question for Limerick Site VisitFor the.wpd

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May 16, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-06-0066 - REGULATORY AND RESOURCE IMPLICATIONS OF A DEPARTMENT OF ENERGY SPENT NUCLEAR FUEL RECYCLING PROGRAM

The Commission has approved staff's recommendations to initiate interactions with DOE and international entities through participation in workshops and meetings domestically and internationally, as appropriate and consistent with its further development, on the safety and safeguards aspects of DOE's spent fuel recycling program and reallocate one additional FTE (for at total of 2 FTE) and $100,000 for FY2006.

The Commission will consider estimates for resources in FY 2007 and FY 2008 as part of the FY 2008 budget process, and expressed concern about ramping up too quickly to support a Department of Energy (DOE) program that still contains major uncertainties, particularly in that the NRC is not currently authorized by statute to license a DOE reprocessing facility, although NRC is authorized by statute to license a demonstration Advanced Burner Reactor (ABR).

NRC resource commitments should be tied to DOE's program decisions. Also, the staff should work with DOE to establish a reimbursable agreement for NRC efforts. The staff should also consider requesting a non-fee based appropriation, as appropriate.

The staff should begin considering the specialized expertise that will be needed for these future reviews when hiring into current open positions.

The staff should focus on the development of a conceptual licensing process for the Administration's Global Nuclear Energy Partnership (GNEP) -related facilities to which the staff has committed in this paper

The staff, in conjunction with OGC, should prepare draft legislation for Commission approval that would give NRC licensing authority for demonstration scale DOE reprocessing, fuel fabrication, vitrification and interim waste storage facilities. In drafting the portion of the proposed legislation that addresses reprocessing facilities, the staff should ensure that they identify any impediments in existing law to NRC licensing of these facilities under a hearing process similar to Part 52 for advanced reactors. Other issues requiring statutory changes should also be identified.

The development of the conceptual licensing process should proceed at a pace commensurate with DOE's progress in identifying the technologies it plans to pursue under GNEP. The conceptual process should consider the most effective and efficient elements of the NRC's

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licensing processes for major facilities, including review of the one-step licensing provisions for enrichment facilities as described in Section 193 of the Atomic Energy Act, and features of nuclear power plant combined licensing under Part 52 (i.e., construction authorization and operating license hearing process, design certification process, and early site permitting process). The staff should also examine the process that EPA used to authorize the operation of the Waste Isolation Pilot Plant. The development of a conceptual licensing process is an inter-office undertaking, likely with NMSS in the lead, but NRR, NSIR, RES and OGC all having significant roles. The Advisory Committees on Reactor Safeguards and Nuclear Waste could also help in defining the issues most important to licensing, inspecting, and ultimate decommissioning of reprocessing facilities (and related fuel-cycle facilities). The staff should consider all aspects of the "full recycle" option of the GNEP in its conceptual process. The licensing process design should be comprehensive in scope, and should address, for example, reactor and other fuel cycle facility safety regulations, environmental reviews, domestic and IAEA safeguards, import and export controls, and waste management. The staff should keep the Commission informed of progress on this effort, and make recommendations as appropriate, on at least an annual basis.

(EDO) (SECY Suspense: 5/1/07)

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

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IN RESPONSE, PLEASE
REFER TO: M060111B

February 9, 2006

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

FROM: Andrew L. Bates, Acting Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ADVISORY COMMITTEE ON NUCLEAR WASTE (ACNW), 2:00 P.M., WEDNESDAY, JANUARY 11, 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Nuclear Waste (ACNW) to discuss its recent activities, especially in the areas of low-level radioactive waste, radiation protection, waste determination, decommissioning issues, and igneous activity in relation to the proposed high-level waste geologic repository.

The Committee should continue to work with both the Offices of Nuclear Material Safety and Safeguards (NMSS) and Nuclear Regulatory Research (RES) to identify opportunities to enhance the technical bases for waste-related activities through monitoring relevant research. The Committee should find, with input from the ACRS, an approach to provide the Commission with a coordinated set of recommendations on how the Center for Nuclear Waste Regulatory Analyses (CNWRA) might broaden its assistance to NRC, for example, to support Office of Nuclear Reactor Regulation (NRR) programs and/or other new and significant regulatory research activities.

The Committee should also work with staff to identify and assess methods of monitoring for compliance and to identify possible enhancements for increasing confidence in the validity of associated analytical models. The committee should specifically consider how these methods could strengthen the reliability and durability of institutional controls.

The Committee should provide the Commission with an analysis of the current state of knowledge regarding igneous activity which the Commission can use as a technical basis for its decision making.

The Committee should review and provide advice to the Commission on the March 2005 French Academy of Sciences report on radiation risks at low dose rates. This should be a comparative analysis of the French study and the findings in the June 2005 BEIR VII report. Among the items the Committee should specifically examine is whether the views and data developed by the Department of Energy's Low Dose Radiation Research Program may have been considered in the French Academy study, but not the BEIR VII study.

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The Committee should provide input on specific technical issues related to waste determinations, when requested by the staff, in areas where the Committee's independent technical expertise will be valuable for decision-making. The Committee should monitor research on technology regarding waste incidental to reprocessing (WIR) and review Department of Energy research reports on this subject and report to the Commission, as appropriate.

The Committee should review best practices in decommissioning to look for ways to improve the design and construction of reactor and materials facilities that would lead to less environmental impact and more efficient decommissioning.

Within the established ACNW Charter, the Committee should continue to provide recommendations to the Commission on significant generic waste issues of importance to policy-making.

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

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From: Mark Salley
To: Sam Duraiswamy
Date: 05/30/2006 2:19:15 PM
Subject: NIST 9/11 Analysis

Sam,

Regarding our conservation earlier today, we are arranging for the Scientists at NIST to provide a 1 ½ to 2 hour presentation to the NRC on September 11, 2006. This "seminar" will be along the lines of the "Chernobyl" presentation made earlier this year. We thought the ACRS may have an interest in this. We are hoping to have the presentation live by NIST Researchers in our auditorium & taped and telecast (video conference) to the Regions. This will be open to all NRC employees.

A bit of Background, the NIST Researchers were heavily involved in trying to understand & calculate (model) what happened on 9/11 to the WTC. The week before last week, we had Commissioner Jaczko at NIST Fire Research facilities for a VIP visit. As a part of the presentation, Dr. Kevin McGrattan provided an overview on "how" they attempted to analyze and understand what happened that day. Dr. McGrattan discussed how they used 4 different models to study the event: The Impact Damage, The Fire, The Fire Effects on the Structural Steel, and the Collapse. Another aspect is the study of WTC7 (the 47 story High Rise that collapsed later on that day) which was not directly impacted. We believe this will make an exciting and informative presentation to all NRC employees. Some (many? all?) of the ACRS members may be interested and are invited if it meets your schedule.

On a secondary note, we will be using the NIST fire models in the 10CFR50.48(c) (NFPA 805) implementation. Two of these models, CFAST and FDS are included in the suite of 5 models that make up the draft V&V document (Draft NUREG-1824) that we sent you earlier this year. If the Fire Protection Sub-Committee (or Full Committee) would like to visit NIST's Fire Research Facilities, I would be more that happy to approach NIST on the subject. NIST Fire Researchers are extremely knowledgeable and professional and a pleasure to work with ~ In my opinion, there are none better. Their location is also quite close in Gaithersburg.

Please let me know if you require any additional information.

MHS

Mark Henry Salley P.E.
Fire Research Team Leader
U.S. Nuclear Regulatory Commission
Office of Research, (RES)
Division of Risk Analysis & Application (DRAA)
Probabilistic Risk Analysis Branch (PRAB)
Washington, D.C. 20555-0001

Mail Stop T10-E52

Phone (301) 415-2840
FAX (301) 415-5062

Email: MXS3@NRC.gov

CC: INternet:Ahamins@nist.gov; INternet:Kmcgrattan@nist.gov; Jason Dreisbach; Michael Junge; Patrick Baranowsky

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
533rd	11	1

Title **RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS**

List of Documents Attached	
See attached list	**11**

Instructions to Preparer	Lead Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	SAM DURAISWAMY

SUBJECT	ANALYSIS	EDO LTR.	ACRS LTR.
Report on the Safety Aspects of the License Renewal Application for the Browns Ferry Nuclear Plant Units 1, 2 and 3 (MVB/CS)	5/09/06 (pp. 1-2)	4/27/06 (pp. 3-5)	3/23/06 (pp. 6-8)
Final Review of the Exelon Generation Company, LLC, Application for Early Site Permit and the Associated NRC Staff's Final Safety Evaluation Report (DAP/MVB/DCF)	5/09/06 (pp. 9-10)	5/02/06 (pp. 11-12)	3/24/06 (pp. 13-15)
Review of the 1994 Addenda to the ASME Code for Class 1, 2, and 3 Piping Systems and the Resolution of the Differences Between the NRC Staff and ASME (JSA/CS)	5/24/06 (p. 16)	5/18/06 (pp. 17-18)	4/14/06 (pp. 19-20)
Generic Safety Issue 191 - Assessment of Debris Accumulation on PWR Sump Performance (GBW/RC)	5/24/06 (pp. 21-22)	5/02/06 (pp. 23-24)	4/10/06 (pp. 25-31)
Anonymous Letter Concerning the TRACE Computer Code Development and Review Practices (GBW/RC)	5/24/06 (p. 32)	3/30/06 (pp. 33-34)	2/15/06 (pp. 35-48)
Draft Final Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (GEA/MAJ)	5/24/06 (pp. 49-50)	5/18/06 (pp. 51-52)	4/20/06 (pp. 53-54)
Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs" (RSD/RC)	5/30/06 (p. 55)	3/15/06 (p. 56)	2/22/06 (pp. 57-58)
Application of the TRACG Computer Code to Evaluate the Stability of the Economic Simplified Boiling Water Reactors (GBW/RC)	5/30/06 (p. 59)	5/22/06 (pp.60-61)	4/21/06 (pp. 62-64)
Grand Gulf Early Site Permit Application: Evaluation of Transportation Accidents on the Mississippi River (DAP/MVB/DCF)	5/31/06 (p. 65)	5/18/06 (p. 66)	5/31/06 (pp. 67-68)

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

May 9, 2006

MEMORANDUM TO: Mario Bonaca, Chairman
Plant License Renewal Subcommittee

FROM: Cayetano Santos Jr., Senior Staff Engineer
Technical Support Branch, ACRS

Cayetano Santos Jr.

SUBJECT: ANALYSIS OF EDO RESPONSE TO THE ACRS REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3

Attached is the EDO's April 27, 2006 response to the Committee's March 23, 2006 report on the safety aspects of the license renewal application (LRA) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3. A copy of the Committee's report is also attached.

COMMITTEE REPORT AND EDO RESPONSE

1. The Committee recommended that the LRA for BFN Units 1, 2, and 3 be approved under two conditions. The first condition is that the drywell refueling seals be included within the scope of license renewal and be subjected to periodic inspections. Alternatively, the drywell shells should be subjected to periodic volumetric inspections to detect external corrosion. The EDO response stated that the Tennessee Valley Authority (TVA) has chosen to perform periodic volumetric examinations of the drywell shells near the sand bed areas of each unit. The first inspection will be performed before entering the period of extended operation and subsequent inspections will be performed at intervals not to exceed 10 years. The staff also accepted TVA's evaluation that the inclusion identified in the drywell shell of Unit 1 was an original metal defect and did not represent a site for future corrosion.

2. The Committee's second condition for recommending approval of the LRA is that if the extended power uprate (EPU) is implemented before the period of extended operation, the staff should require that TVA evaluate the operating experience of Units 1, 2, and 3 at the uprated power level and incorporate lessons learned into their aging management programs prior to entering the period of extended operation. The EDO response stated that a commitment to perform such a review will be required as part of the EPU amendment.

3. The Committee report noted that a recent inspection found errors in aging management program implementation packages and recommended that the inspections performed before BFN enters the period of extended operation verify that the implemented corrective actions have been effective. The EDO response did not address this recommendation.

4. The Committee report noted that the final Safety Evaluation Report (SER) refers to some restart inspections as one-time inspections and recommended that the SER be revised to use these terms consistently. The EDO response stated that the final SER was corrected and published as NUREG-1843.

ANALYSIS

In general, the EDO response is satisfactory. The EDO response did not address the recommendation to verify the effectiveness of corrective actions to address errors in the implementation packages of aging management programs. The EDO response also noted that Supplement 1 to the final SER addresses some of the Committee's concerns. This supplement was provided to the Committee in a memorandum dated April 25, 2006.

Attachments: As stated

cc: w Attachments : ACRS Members
J. Larkins
A. Thadani
M. Snodderly
E. Thornsbury
S. Duraiswamy

NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

April 27, 2006

MAY - 8 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESPONSE TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3

Dear Dr. Wallis:

During the 530th meeting of the Advisory Committee on Reactor Safeguards (ACRS or the Committee), held on March 9-11, 2006, ACRS completed its review of the license renewal application (LRA) for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, and the associated safety evaluation report (SER), NUREG-1843 prepared by the U.S. Nuclear Regulatory Commission (NRC) staff.

In its final report, the Committee agreed to recommend the renewal of the operating licenses for BFN Units 1, 2, and 3 with the inclusion of certain conditions discussed in Recommendations and 3 of your letter dated March 23, 2006. We appreciate the Committee's objective and -depth review of the BFN application and SER. The staff has issued a supplemental SER (SSER), Supplement 1 to NUREG-1843 that addresses the Committee's concerns. The staff's resolution and disposition of the Committee's comments are also elaborated below.

ACRS Comment:

> "The drywell refueling seals should be included within the scope of license renewal and be subjected to periodic inspections. Alternatively, as proposed by the staff, the drywell shells should be subjected to periodic volumetric inspections to detect external corrosion."
>
> "As an alternative to the staff's proposal, the applicant committed to perform a one-time confirmatory inspection...One-time inspection of the shell does not provide assurance that leakage of the refueling seals after the one-time inspection is performed will not create an environment that could result in the future drywell degradation."

Response:

The staff pursued this issue further with the applicant. The applicant has chosen to adopt the alternative recommendation to perform periodic volumetric examination of the drywell shell near the sand bed areas of concern according to the American Society of Mechanical Engineers SME) Code Section-XI, Subsection IWE, with suitable enhancement to the program ments. The applicant will perform the first inspection on each unit prior to the period of extended operation. Subsequent periodic inspections will be performed on each unit at an interval not to exceed 10 years. The staff will review the results of these inspections to ensure that the acceptance criteria of ASME Section-XI, Subsection IWE-3000, were met including loss of any localized base metal degradation, that required repair consistent with IWE-3122 criteria. The staff evaluated and accepted these program enhancements which are shown in the SSER.

ACRS Comment:

> "Ultrasonic inspections performed in 1999, 2002, and 2004 identified a small inclusion in the drywell liner of Unit 1. The applicant will submit this information to the staff in writing. The staff plans to document its evaluation of this information in a supplemental SER. Based on our discussions with the applicant and staff, the resolution of this issue does not affect our recommendations regarding this LRA."

Response:

In a letter dated April 4, 2006, the applicant confirmed in writing that the inclusion was an original metal defect and subsequent inspections revealed no measurable difference in the depth and size of the inclusion. The applicant stated that the inspectors did not observe any change in the thickness of the liner in this area. The applicant hence concluded that the presence of the inclusion did not affect the strength of the drywell containment shell and it did not represent a site for future corrosion. The staff accepted the applicant's evaluation. The staff included details of these evaluations in the SSER.

ACRS Recommendation:

> "Section 3.7 of the final SER still refers to some restart inspections as one-time inspections. The final SER should be revised to be consistent with these definitions."

Response:

The staff regrets this oversight. The final SER published as NUREG-1843 was corrected and maintains the consistency that was of concern to the Committee.

ACRS Recommendation:

> "If the extended power uprate (EPU) is implemented, the staff should require that TVA evaluate Units 1, 2, and 3 operating experience at the uprated power level and incorporate lessons learned into their aging management programs prior to entering the period of extended operation."

Response:

By letter dated November 28, 2005, the staff responded to the Committee's concern by referencing a previous staff response to the ACRS dated October 26, 2004, regarding the safety aspects of the license renewal application for the Dresden and Quad Cities Nuclear Power Stations. In accordance with that letter, BFN will be required to include a regulatory commitment in its EPU amendment requiring operating experience and aging management programs review prior to entering the period of extended operation.

The staff recognizes the ACRS's commitment to safety and appreciates the Committee's continued support of the license renewal process.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

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

ACRSR-2180

March 23, 2006

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3

Dear Chairman Diaz:

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, we completed our review of the license renewal application (LRA) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 and the associated final Safety Evaluation Report (SER) prepared by the NRC staff. On August 23, 2005, we visited the Browns Ferry site and reviewed activities under way for license renewal, power uprate, and restart. Our Plant Operations and Plant License Renewal Subcommittees also reviewed these matters on September 21, 2005. Our Plant License Renewal Subcommittee reviewed the LRA and SER with Open Items on October 5, 2005. We issued an interim letter on the safety aspects of this application on October 19, 2005. During our reviews, we had the benefit of discussions with representatives of the NRC staff, including Region II personnel, and the Tennessee Valley Authority (TVA). We also had the benefit of the documents referenced. This report fulfills the requirements of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSIONS AND RECOMMENDATIONS

1. With the inclusion of the conditions in Recommendations 2 and 3, the application for license renewal for BFN Units 1, 2, and 3 should be approved.

2. The drywell refueling seals should be included within the scope of license renewal and be subjected to periodic inspections. Alternatively, as proposed by the staff, the drywell shells should be subjected to periodic volumetric inspections to detect external corrosion.

3. If the extended power uprate (EPU) is implemented before the period of extended operation, the staff should require that TVA evaluate the operating experience of Units 1, 2, and 3 at the uprated power level and then incorporate lessons learned into their aging management programs prior to entering the period of extended operation.

DISCUSSION

TVA requested renewal of the BFN Units 1, 2, and 3 operating licenses for 20 years beyond their current operating terms, which expire on December 10, 2013, June 28, 2014, and July 2, 2016, respectively.

and no unacceptable weld cracks exist. We concur with the staff's conclusion that this program will adequately manage the aging effects for which it is credited.

In the original BFN LRA, the applicant requested renewed licenses at EPU conditions for all three units. In a letter dated January 7, 2005, TVA requested that the EPU and the LRA be separated. Even though the staff reviewed the LRA based on current licensed power levels for each unit, the final SER has several references to EPU conditions. The steam dryers are included in the scope of license renewal, but their aging management review will be performed as part of the safety evaluation of the EPU application. The time-limited aging analyses (TLAAs) associated with neutron embrittlement, reactor vessel fatigue, radiation degradation of drywell expansion gap foam, and stress relaxation of the core plate hold-down bolts were performed assuming EPU conditions.

In the final SER, the staff documents its review of the license renewal application and other information submitted by TVA and obtained through the audits and inspections conducted at the plant site. The staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's aging management programs (AMPs); and the identification and assessment of TLAAs requiring review.

The BFN application either demonstrates consistency of aging management programs with the Generic Aging Lessons Learned (GALL) Report or documents deviations from the approaches specified in the GALL Report. The staff reviewed this application in accordance with NUREG-1800, the Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants.

The staff also performed inspections and an audit of AMPs and aging management reviews (AMRs). A recent inspection found that the applicant had made significant progress in developing the AMP implementation packages but identified errors in them. The applicant initiated a Problem Evaluation Report to identify the causes of the errors and determine corrective actions to prevent recurrence. Inspections performed before BFN enters the period of extended operation should verify that implemented corrective actions have been effective.

The audit of the AMPs and AMRs is documented in a report by the Brookhaven National Laboratory. The audit examined 28 AMPs and the associated AMRs and verified that the AMPs are consistent with the GALL Report or concluded that they would adequately manage aging during the period of extended operation. Several of the existing AMPs will be enhanced to include Unit 1 prior to the period of extended operation. Appendix F of the LRA describes TVA's plan to resolve the differences between the licensing bases of Unit 1 and Units 2 and 3 before Unit 1 restart. The staff's review of Appendix F did not identify any omissions or discrepancies.

The staff concluded that the scoping and screening processes implemented by the applicant have successfully identified SSCs within the scope of license renewal and subject to an AMR. With the inclusion in the scope of license renewal of those Unit 1 systems and components that were in layup and have not been replaced, we agree with this conclusion.

Open item 2.4-3 in the SER concerns aging management of drywell shell corrosion. The staff was concerned that leakage through refueling seals at the top of the drywell could lead to corrosion of the drywell shell in a location that cannot be inspected. This aging effect has been incorporate lessons learned into their aging management programs prior to entering the period of extended operation. The EDO response to our interim letter stated that the staff's SER for the EPU would include a commitment to perform such an evaluation.

With the inclusion of commitments to perform periodic inspections of BFN Units 1, 2, and 3 drywell refueling seals or drywell shells and to perform an evaluation of operating experience at the EPU level and incorporate lessons learned into their aging management programs prior to entering the period of extended operation, the application for license renewal of Browns Ferry Units 1, 2, and 3 should be approved.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3," January 2006.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3," August 2005.
3. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Application for Renewed Operating Licenses," December 31, 2003.
4. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - January 28, 2004 Meeting Follow-Up - Additional Information," February 19, 2004.
5. Brookhaven National Laboratory, "Audit and Review Report for Plant Aging Management Programs (AMPs) and Aging Management Reviews (AMRs), Browns Ferry Nuclear Plant Units 1, 2, and 3, Docket Nos.: 05000259, 05000260, 05000296," April 26, 2005.
6. U.S. Nuclear Regulatory Commission, "Browns Ferry Nuclear Plant - Inspection Report 05000259/2004012, 05000260/2004012, and 05000296/2004012," January 27, 2005.
7. U.S. Nuclear Regulatory Commission, "Browns Ferry Nuclear Plant - Inspection Report 05000259/2005013, 05000260/2005013, and 05000296/2005013," November 7, 2005.
8. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application (LRA) - Annual Update (TAC Nos. MC1704, MC1705, and MC1706)," January 31, 2006.
9. Letter from William J. Shack, Acting Chairman, ACRS, to Luis A. Reyes, Executive Director for Operations, NRC, "Interim Report on the Safety Aspects of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3," October 19, 2005.
10. Letter from Luis A. Reyes, Executive Director for Operations, NRC, to William J. Shack, Acting Chairman, ACRS, "Response to Advisory Committee on Reactor Safeguards - Interim Report on the Safety Aspects of the License Renewal Application for Browns Ferry Nuclear Plant, Units 1, 2, and 3," November 28, 2005.
11. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Summary of NRC Site Visit and Meeting Regarding Extended Power Uprate (EPU) and License Renewal Application (LRA)," January 7, 2005.
12. U.S. Nuclear Regulatory Commission,"10 CFR Parts 2, 50, 54, and 140, Nuclear Power Plant License Renewal," *Federal Register*, Vol. 54, No. 240, December 13, 1991, pp. 64943-64980.
13. U.S. Nuclear Regulatory Commission,"10 CFR Parts 2, 51, and 54, Nuclear Power Plant License Renewal; Revisions," *Federal Register*, Vol. 60, No. 88, May 8, 1995, pp. 22461-22495.

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

May 9, 2006

MEMORANDUM TO: Dana Powers, Chair
Early Site Permit Subcommittee

FROM: David C. Fischer, Senior Staff Engineer /RA/

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON THE FINAL REVIEW OF THE EXELON GENERATION COMPANY, LLC, APPLICATION FOR EARLY SITE PERMIT AND THE ASSOCIATED NRC STAFF'S FINAL SAFETY EVALUATION REPORT

Attached is a copy of the EDO's May 2, 2006, letter of response to the ACRS's March 24, 2006, report on the Committee's review of the early site permit application for the Clinton site and the associated final Safety Evaluation Report (SER) prepared by the NRC staff. A copy of the Committee's letter is also attached.

Committee Letter

In its letter, the Committee concluded that the early site permit application and the staff's final SER show that the proposed nuclear power plant site adjacent to the existing Clinton Nuclear Power Station is an acceptable site for nuclear power plants that meet the plant parameter envelope proposed by the applicant. The Committee also concluded that the staff had thoroughly reviewed a performance-based method proposed by the applicant for determining the safe shutdown earthquake (SSE) ground motion and stated that this method is an attractive alternative to methods endorsed in current regulatory guides. The Committee recommended that the staff consider development of a regulatory guide dealing with the alternative, performance-based, method for assessing the seismic hazard of a site.

EDO Response

In the EDO's response letter, the staff stated that it has been interacting with industry representatives since summer 2005 to further develop the generic technical bases for the performance-based approach for determining the SSE. The staff stated that, in parallel with these interactions, it has initiated the process for revising Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," to allow for the use of the performance-based approach, as well as to address other seismic related technical issues. The staff's current plan calls for a draft revision of Regulatory Guide 1.165 by the end of 2006. After the NRC staff has developed the draft regulatory guide, the staff plans to discuss the proposed revisions with the ACRS.

Analysis

The EDO's response is satisfactory. The Committee should plan to review guidance being developed to address the performance-based method for assessing the seismic hazard of a site. In addition, the ACRS staff will coordinate with the NRC staff to schedule appropriate

ACRS Subcommittee meetings to discuss ESP lessons learned and to review guidance for future combined license (COL) applicants.

cc: ACRS Members
SDuraiswamy
MSnodderly
EThornsbury

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

May 2, 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: FINAL REVIEW OF THE EXELON GENERATION COMPANY, LLC, APPLICATION FOR EARLY SITE PERMIT AND THE ASSOCIATED NRC STAFF'S FINAL SAFETY EVALUATION REPORT

Dear Chairman Wallis:

Thank you for your letter dated March 24, 2006, regarding the final safety evaluation report (FSER) for the Exelon Generation Company, LLC, (EGC) early site permit (ESP) application. The staff of the U.S. Nuclear Regulatory Commission (NRC) will reproduce your letter as Appendix E to the FSER for the EGC ESP, which will be issued as a final NRC technical report, a NUREG, on May 1, 2006. In your letter, the Advisory Committee on Reactor Safeguards (ACRS) agreed with the NRC staff's conclusions that the EGC ESP site is adequate for the proposed use when subject to the six proposed permit conditions, and the ACRS recognized that the NRC staff has thoroughly reviewed EGC's new performance-based method for determining the safe shutdown earthquake ground motion (SSE). On this basis, the NRC staff has updated the FSER to reflect the ACRS conclusions.

In your letter, you also stated that the NRC staff should consider development of a regulatory guide to address the performance-based method for assessing the seismic hazard of a site. As discussed in the March 9, 2006, ACRS meeting, the staff has been interacting with industry representatives since summer 2005 to further develop the generic technical bases for the performance-based approach for determining the SSE. In parallel with these interactions, the NRC staff has initiated the process for revising Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," to allow for the use of the performance-based approach, as well as to address other seismic related technical issues. The current plan calls for a draft revision of Regulatory Guide 1.165 by the end of 2006. After the NRC staff has developed the draft regulatory guide, we plan to discuss the proposed revisions with the ACRS.

The NRC staff appreciates the insights that the ACRS has provided concerning the sections on safety in the EGC ESP. These insights are a valuable contribution to the NRC staff's review and development of the FSER.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

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

March 24, 2006

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

SUBJECT: FINAL REVIEW OF THE EXELON GENERATION COMPANY, LLC, APPLICATION FOR EARLY SITE PERMIT AND THE ASSOCIATED NRC STAFF'S FINAL SAFETY EVALUATION REPORT

Dear Chairman Diaz:

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, we completed our review of the early site permit application for the Clinton site and the associated final Safety Evaluation Report (SER) prepared by the NRC staff. We reviewed the application and the final SER to fulfill the requirement of 10 CFR 52.23 that the ACRS report on those portions of an early site permit application that concern safety. We issued an interim letter on this application and the associated draft SER on September 22, 2005. This matter was also discussed during our Subcommittee meeting on March 8, 2006. During these reviews, we had the benefit of discussions with representatives of the NRC staff and Exelon Generation Company, LLC (Exelon). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

- The early site permit application and the staff's final SER show that the proposed nuclear power plant site adjacent to the existing Clinton Nuclear Power Station is an acceptable site for nuclear power plants that meet the plant parameter envelope proposed by the applicant.
- The staff has thoroughly reviewed a performance-based method proposed by the applicant for determining the safe shutdown earthquake (SSE) ground motion. This method is an attractive alternative to methods endorsed in current regulatory guides.
- The staff should consider development of a regulatory guide dealing with the alternative, performance-based, method for assessing the seismic hazard of a site.

DISCUSSION

Exelon has applied for an early site permit for locating nuclear power plants or modules having a total power generation rate of 2400 to 6800 MWt on a site adjacent to the currently operating Clinton plant, which is a BWR 6 within a Mark III containment. The early site permit application is based on the now familiar "plant parameter envelope" approach since the applicant has not identified the particular reactor technology that will be adopted. The plant parameter envelope is based on the characteristics of certified designs such as the AP1000 and Advanced Boiling

Water Reactor (ABWR) as well as other designs such as the International Reactor Innovative and Secure (IRIS), Economic Simplified Boiling Water Reactor (ESBWR), Gas-Turbine Modular Helium Reactor (GT-MHR), and Pebble Bed Modular Reactor (PBMR).

The staff's review of this application included a detailed review of the alternative, performance-based method proposed by the applicant for determining the SSE ground motion spectrum. The staff identified six permit conditions for the proposed site. The staff has used technically sound, objective criteria for identifying these permit conditions. The staff and the applicant have agreed to 32 combined license (COL) action items. The action items for the proposed Clinton site can be compared to 30 action items for the North Anna early site permit and 26 action items for the Grand Gulf early site permit.

Nature of the Site

The proposed site is located in a rural setting in central Illinois. The terrain is essentially flat with some rolling hills. Nearby populations centers with populations in excess of 25,000 include Springfield (74 km), Peoria (75 km), Champaign (49 km), Urbana (66 km), Decatur (36 km), and Bloomington (36 km). Near the site (<16 km) are the small towns Clinton (population 7,000), as well as DeWitt, Weldon, and Wapella each with a population of less than 1,000.

Population trends in the larger cities near the site have been estimated based on census data. Modest growth in population is anticipated in these cities over the next 60 years. Interestingly, data obtained from other sources led the applicant to anticipate that populations in the rural regions around the site will decline modestly over the next 60 years.

Weather

Weather at the proposed site is well characterized in recent years as would be expected for a site with an operating nuclear power plant. The weather is marked by rather warm summer periods and harsh winters. Weather extreme characteristics of the site have been based on historical data. Neither the applicant nor the staff has considered the potential for cycles in weather that may complicate the prediction of future weather extremes based on historical records. Nevertheless, we believe that the applicant has adequately characterized the site weather for the purposes of an early site permit.

Seismicity

The proposed site is affected by the New Madrid seismic zone and the Wabash Valley seismic zone. Since the nuclear power plant at the Clinton site was licensed, the estimated frequency of major earthquakes at the New Madrid seismic zone has been increased. The estimate of the maximum potential magnitude of earthquakes at the Wabash Valley seismic zone has also been increased. There is a background seismicity of the site represented by the Springfield earthquake estimated to have occurred at a location about 70 km from the site, approximately 6,000 years ago and to have had a magnitude of 6.2 to 6.8 on the Richter scale.

In other applications for early site permits, the applicants have adopted the methods recommended in Regulatory Guide 1.165 to estimate the SSE ground motion spectrum. Exelon has adopted an alternative method. This alternative is based on an industry standard (ASCE 43-05) that itself is based on work done by the Department of Energy for assessing the seismic safety of its nuclear facilities. The alternative is considered "performance based" because it uses a target probability for the maximum acceptable facility damage from an earthquake.

Exelon has selected the frequency of 10^{-5}/yr for the onset of significant inelastic deformation of systems, structures, and components. This target provides a rather substantial margin to core damage and containment failure.

The staff has reviewed thoroughly the proposed alternative method for estimating the seismic hazard at the proposed site. The staff's review included examination of the credibility of parametric quantities in the models and an independent assessment of the analysis results by direct integration of the seismic risk equation. Also, the staff has reviewed carefully the applicant's assessment of the local seismic hazard. We concur with the staff that the alternative approach adopted by Exelon for this application provides a high level of safety. The seismic core damage frequency that can be inferred from the proposed ground motion spectrum ($\sim 2 \times 10^{-6}$/yr) is significantly less than the median found in seismic probabilistic risk assessments for 29 existing nuclear power plants. The performance-based alternative method yields results that are in concert with the Commission's expectation that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions.

The alternative, performance-based, method uses a target frequency that does not change with time as new information on the seismicity of power plant sites changes. In this sense, the alternative method provides some additional regulatory stability. For this reason, if no other, we expect that the alternative method will be attractive to licensees and applicants for a variety of purposes. The staff may want to consider developing a regulatory guide on the use of the alternative methodology. Certainly, the detailed review of the method conducted by the staff for this early site permit would provide a substantial technical basis for the development of such a regulatory guide.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
1. Exelon Generation Company, LLC, Early Site Permit Application, September 23, 2003.
2. ACRS Interim Letter, Exelon Generation Company, LLC, Application for Early Site Permit and the Associated NRC Staff's Draft Safety Evaluation Report, dated September 22, 2005.
3. EDO response to ACRS Interim Letter, "Interim Letter: Exelon Generation Company, LLC, Application for Early Site Permit and the Associated NRC Staff's Draft Safety Evaluation Report on the Clinton Early Site Permit Site," dated October 26, 2005.
4. Final Safety Evaluation Report for Exelon Early Site Permit Application, dated February 17, 2006.
5. Exelon Generation Company, LLC, letter to the U.S. Nuclear Regulatory Commission Subject: "Seismic Risk (Performance Goal) Based Approach Primer Revision," dated January 14, 2005.
6. NRC Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," March 1997.
7. American Society of Civil Engineers, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, ASCE/SEI 43-05 (ASCE Standard 43-05), 2005.

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

May 24, 2006

MEMORANDUM TO: Joseph S. Armijo, Chairman
Materials and Metallurgy Subcommittee

FROM: Cayetano Santos Jr., Senior Staff Engineer
Technical Support Branch, ACRS

SUBJECT: ANALYSIS OF EDO RESPONSE TO THE ACRS LETTER ON THE REVIEW OF THE 1994 ADDENDA TO THE ASME CODE FOR CLASS 1, 2, AND 3 PIPING SYSTEMS AND THE RESOLUTION OF THE DIFFERENCES BETWEEN THE NRC STAFF AND ASME

Attached is a copy of the EDO's May 18, 2006 response to the Committee's April 14, 2006 letter on the Review of the 1994 Addenda to the ASME Code for Class 1, 2, and 3 Piping Systems and the Resolution of the Differences Between the NRC Staff and ASME. A copy of the Committee's letter is also attached.

COMMITTEE LETTER

The Committee letter stated that most of the differences between the staff and ASME are resolved and the staff proposes to address the one remaining issue related to dynamic strain aging of carbon steels at elevated temperatures by placing a restriction on the endorsement of the ASME Code in 10 CFR 50.55a. The Committee noted that this approach is practical, but encouraged the staff to work with ASME to resolve the one remaining issue.

EDO RESPONSE

The EDO response stated that the staff will continue to work with ASME and public stakeholders in an attempt to resolve technical differences. The response also noted that the staff has exhaustively reviewed test data and analyses regarding dynamic strain aging and believes a restriction in 10 CFR 50.55a is necessary.

ANALYSIS

The EDO response is satisfactory.

Attachments: As stated

cc: w Attachments : ACRS Members
J. Larkins
A. Thadani
M. Snodderly
E. Thornsbury
S. Duraiswamy

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

May 18, 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: RESPONSE TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS LETTER, DATED APRIL 14, 2006, CONCERNING THE REVIEW OF THE 1994 ADDENDA TO THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS CODE FOR CLASS 1, 2, AND 3 PIPING SYSTEMS AND THE RESOLUTION OF THE DIFFERENCES BETWEEN THE NUCLEAR REGULATORY COMMISSION STAFF AND THE ASME

Dear Dr. Wallis:

Thank you for your letter of April 14, 2006, concerning the resolution of differences between the Nuclear Regulatory Commission (NRC) staff and the American Society of Mechanical Engineers (ASME) involving the 1994 Addenda to Section III of the ASME Boiler and Pressure Vessel Code for Class 1, 2, and 3 piping systems.

As discussed in the staff's April 7, 2006, presentation to the ACRS, the NRC, in Title 10 Code of Federal Regulations (CFR) 50.55a, has not permitted the use of the ASME Code criteria for the seismic design of ASME Class 1, 2, and 3 piping systems since significant relaxations of the criteria were introduced in the 1994 Addenda. After several years of extensive discussions that took place as part of the staff's participation in special working groups established by the ASME, most of the differences between the staff and the ASME have been resolved through modifications to the 1994 criteria.

The one remaining issue involves the potential for a reduction in material strength due to dynamic strain aging of certain carbon steels at temperatures above 300°F. The staff proposes to address this issue by placing a restriction on the use of the ASME code piping criteria in a future 10 CFR 50.55a rule update. The staff notes that, even with the proposed restriction, the new piping rules would still be a relaxation of the ASME Code criteria as currently accepted in 10 CFR 50.55a.

In the subject letter, the Committee encouraged the staff to continue to work with the ASME to resolve this remaining issue. The staff will continue to work with ASME and public stakeholders in an attempt to resolve technical differences. The NRC staff has exhaustively reviewed the currently available test data and analyses regarding the issue of dynamic strain aging and believes a restriction in 10 CFR 50.55a is necessary to address this issue. If this issue is resolved in the future, and appropriate changes are made to the ASME Code, we could endorse that version without this proposed restriction. The ASME will have an opportunity to comment on the draft rule update.

We appreciate the Committee's attention to this topic.

Sincerely,

William Kane /RA/

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

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

April 14, 2006

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REVIEW OF THE 1994 ADDENDA TO THE ASME CODE FOR CLASS 1, 2, AND 3 PIPING SYSTEMS AND THE RESOLUTION OF THE DIFFERENCES BETWEEN THE NRC STAFF AND ASME

Dear Mr. Reyes:

During the 531st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2006, we reviewed the resolution of the differences between the NRC staff and the American Society of Mechanical Engineers (ASME) regarding the 1994 Addenda to Section III of the ASME Boiler and Pressure Vessel Code for Class 1, 2, and 3 piping systems. During our reviews, we had the benefit of discussions with representatives of the NRC staff and ASME. We also had the benefit of the documents referenced.

RECOMMENDATION

Most of the differences between the staff and ASME are resolved. The staff proposes to address the one remaining issue related to dynamic strain aging of certain carbon steels at temperatures greater than 300 °F by placing a restriction on the endorsement of the ASME Code in 10 CFR 50.55a. This approach is practical; however, we encourage the staff to work with ASME to resolve the one remaining issue.

DISCUSSION

The NRC staff initially did not endorse the revised seismic design criteria in the 1994 Addenda to the ASME Code because of concerns with the technical basis used to establish these criteria. Since that time, the ASME has initiated changes to the Code to address the staff's concerns. These changes include eliminating the application of the seismic rules to flow-transient loads, eliminating the NB-3200 strain criteria, modifying the Class 2 and 3 Level B limits to be consistent with the Level D limits, eliminating changes specifying the methods to generate seismic loads in the evaluation of reversing dynamic loads, and adding provisions to address potential strain concentrations. The staff agrees with these changes.

The remaining unresolved issue between ASME and the staff relates to the effects of dynamic strain aging on the ultimate tensile capacity of certain carbon steels at temperatures greater than 300 °F. The staff proposes to address this issue by placing a restriction in the 10 CFR 50.55a endorsement of the ASME Code. This approach is practical; however, we encourage the staff to work with ASME to resolve the one remaining issue.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Seismic Analysis of Piping," NUREG/CR-5361, June 1998.
2. Letter to G.M. Eisenberg, Director, Nuclear Codes and Standards, ASME, from Brian W. Sheron, NRR, "ASME Code Revisions to the Design Rules for Piping Systems," May 24, 1995.
3. Presentation by John R. Fair, NRR, to the ACRS Subcommittee on Materials and Metallurgy, "Piping Seismic Design Criteria," March 25, 1999.
4. Presentation by John R. Fair, NRR, to William J. Shack, ACRS, "Status of ASME Code Piping Seismic Design Criteria," October 3, 2003.

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

May 24, 2006

MEMORANDUM TO: ACRS Members

FROM: R. Caruso, Senior Staff Engineer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER CONCERNING GSI-191 - ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE

Attached for your information is a copy of the EDO's May 2, 2006 response to the ACRS's letter of April 10, 2006, concerning the Committee's review of staff progress in resolving GSI-191. A copy of the Committee's letter is also attached.

Committee Letter

In its letter, the Committee (1) concurred with the staff's intent to increase the size of their sump screens as quickly as feasible, but noted that this measurer will not be sufficient to resolve all long-term cooling issues, (2) recommended that further work be done to provide a technical basis for assessing the adequacy of planned modifications and develop staff review guidance, (3) recommended that improved predictive methods and guidance be developed for the screen pressure drop in the presence of particle/fiber mixtures and chemical reaction products, (4) recommended that improved predictive methods and guidance be developed related to the amount of debris that bypasses the screens, and its effects on downstream components, including the core itself, (4) recommended that work on the equilibrium chemistry model should continue and the model should be validated and additional guidance for its use developed, and (5) recommended that the work on coating debris formation and transport should continue, and predictive capability and guidance should be developed for the effects of coatings.

EDO Response

The EDO agreed that additional guidance is needed in several technical areas, to ensure a consistent and defensible review of licensee submittals. The EDO also agreed that developing predictive methods would be valuable. However, the EDO considers development of predictive models to be a challenging and long-term effort which may not achieve timely closure of GSI-191 issues and may not effectively address plant-specific design variations.

Instead, the EDO describes the current staff approach as one that strives to ensure that licensees make modifications that "reasonably bound technical uncertainties and extrapolations from test to plant conditions." The staff is "... committed to finishing planned NRC sponsored research activities in spring 2006."

The EDO then states that the staff will address other concerns raised by the ACRS "... as appropriate information becomes available..." and he noted that the staff plan for review of licensee efforts "... will consist of observations of testing and audits of vendor/licensee sump evaluations." He closes with the statement that "... the staff needs sufficient technical bases to evaluate the sump modifications, and we are developing integrated plans to acquire these bases where they are currently lacking."

Analysis

The EDO's response is unsatisfactory. It is internally inconsistent - it opens with an agreement for the need for review guidance, and the value of predictive models, but then it states that such model development would interfere with achieving the schedule goals. It says in one sentence that the staff is committed to completing research activities in spring 2006, while the next sentence says that additional work by industry and the staff may be needed to address remaining issues. The last paragraph says that they will develop integrated plans to acquire technical bases, even though they plan to shutdown the ongoing efforts in the spring of 2006.

This document says so many conflicting things that it is useless, except as a tool a debate, where each side might selectively use isolated phrases that support their diametrically-opposed positions. The illogic is so twisted that it effectively ends up saying <u>nothing</u>.

The one part of this response that is truly troublesome is the description of the review process that is planned. This confirms what I have been told by the staff - they plan to only look at a few tests at the vendor sites, and audit a few (~10) plants, and then declare that the issue is resolved. They believe that by carefully selecting the plants to be audited, they can effectively cover the entire population. This might be a valid approach if the methodology was well defined and the phenomena were well understood, but that is not the case here. There are too many degrees of freedom in the analysis methodology, in the testing protocols, and in the geometry of the 59 different plants, not to mention chemical effects, the state of coatings, and core configurations.

This whole effort is supposed to demonstrate that the plants comply with one of the five criteria in 10 CFR 50.46. The methodologies that are used to demonstrate compliance with the other four criteria are well controlled and understood, and generally have large margins. This part of the 50.46 methodology does not meet those criteria, and the staff seems to be saying that it does not have to be as rigorous, because the initiating event is unlikely. I am very uncomfortable with simply defining away risk like this.

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

May 2, 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: GENERIC SAFETY ISSUE 191 - ASSESSMENT OF (EFFECT OF) DEBRIS ACCUMULATION ON PRESSURIZED-WATER REACTOR SUMP PERFORMANCE

Dear Dr. Wallis:

In your letter dated March 24, 2006, the Advisory Committee on Reactor Safeguards (ACRS) provided six conclusions and recommendations regarding the staff's efforts to resolve Generic Safety Issue (GSI) 191, "Assessment of (effect of) Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance." The staff offers the following responses to your letter.

The staff acknowledges the Committee's agreement with the plan to make near-term sump modifications to reduce vulnerability to sump clogging, and we agree with the Committee that additional actions may be needed to resolve the sump clogging issues. Whether such measures are needed should become clear through the ongoing testing activities and licensee actions intended to address GSI-191.

The Committee's letter recommends that the staff develop review guidance in several technical areas related to GSI-191. The staff agrees that such guidance is needed to ensure a consistent and defensible review of licensees' submittals to address sump clogging issues. Review guidance is already available for many aspects of the issue in the staff's December 6, 2004, safety evaluation of Nuclear Energy Institute Document 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," dated May 28, 2004. We recognize that additional guidance needed in some areas such as chemical effects and water management strategies, and we plan to develop and update guidance as the state of knowledge evolves. It may be necessary to augment this guidance based on reviews of licensees' supplemental responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors."

There are various approaches for ensuring that licensee sump modifications sufficiently address GSI-191 technical issues. The Committee's letter recommends developing improved predictive methods to provide the technical basis for resolving several stated concerns. The Committee indicated that models could be used for estimating the sump screen pressure drop due to particle/fiber mixtures, chemical reaction products, and coating debris; scaling industry proof testing results to post-LOCA plant conditions; and assessing the quantity and properties of debris that bypasses the sump screen. The staff agrees that developing such predictive methods would be valuable. However, we consider development of predictive models to be a challenging and long-term effort which may not achieve timely closure of GSI-191 issues and may not effectively address plant-specific design variations.

Our current approach is to ensure that licensees make the sump modifications necessary to reasonably bound technical uncertainties and extrapolations from test to plant conditions. We are using results from the Nuclear Regulatory Commission (NRC) sponsored research, industry testing activities, and conservative licensee evaluations to ensure that sufficient technical bases exist to adequately resolve many GSI-191 technical issues. The staff is committed to finishing planned NRC sponsored research activities in spring 2006. Additional work by industry and the staff may be needed to address some remaining issues such as chemical effects and downstream effects. We will continue to participate in industry efforts to address sump performance issues and will incorporate information obtained into its issue resolution strategy as appropriate.

The staff and the industry plan additional analyses of the effects of debris that may enter the reactor vessel. Other concerns raised by the Committee will be addressed as appropriate information becomes available. For example, we plan to review approaches used by each of the five vendors selected by licensees to support them in addressing GSI-191. The reviews will consist of observations of testing and audits of vendor/licensee sump evaluations. These activities will continue over the next 2 years.

In summary, the staff recognizes, as does the Committee, that more work remains to be done on GSI-191. We agree with the Committee's conclusion that the staff needs sufficient technical bases to evaluate the sump modifications, and we are developing integrated plans to acquire these bases where they are currently lacking. We appreciate the Committee's review and recommendations and look forward to resolving this important issue with you.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

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

REVISED

April 10, 2006

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

SUBJECT: GENERIC SAFETY ISSUE 191 - ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE

Dear Chairman Diaz:

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, we considered several reports by the NRC staff regarding their efforts to resolve Generic Safety Issue 191(GSI-191), "Assessment of Debris Accumulation on PWR Sump Performance." The staff discussed licensee responses to Generic Letter 2004-02 (GL 2004-02), "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," and presented the results of efforts by the Office of Nuclear Regulatory Research (RES) to understand several phenomenological issues that have arisen as part of the GSI-191 effort, including chemical effects, downstream effects, and head loss correlations through debris beds. The results were presented to our Thermal-Hydraulics Phenomena Subcommittee on February 14-16, 2006. We had the benefit of presentations by and discussion with representatives of the NRC staff and members of the public. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. In response to GL 2004-02, many licensees plan to increase the size of their sump screens as quickly as feasible. Based on the current state of knowledge, we concur with this intent. However, it is not evident that this measure will be sufficient to resolve all long-term core cooling issues.

2. Results of prototypical experiments planned by industry to validate screen effectiveness will be difficult to extrapolate to plant conditions. Further work is required to provide the technical basis by which the staff can assess the adequacy of the planned modifications to the plants. Guidance should be developed to support the staff's review.

3. Recent research has revealed significant influences of particle/fiber mixtures and chemical reaction products on screen pressure drop for which improved predictive methods and guidance should be developed.

4. Increasing screen size to reduce the pressure drop may increase the amount of fine debris and chemical products that passes through the screen. Methods for predicting the quantity and properties of this bypassed debris should be developed. Potential adverse effects on downstream components, including pumps, valves, the core entrance regions, and the core itself, should be evaluated.

5. There has been some success at using adjustable parameters in an equilibrium chemistry model to match the chemical species that form in sumps. The methods should be validated further and guidance should be developed for their use.

6. The results of tests of coating debris formation and transport should be included in the assessment of core coolability as they become available. Future work should include the development of adequate predictive capability for the effects of coating debris on screen pressure drop and bypass.

OVERVIEW

At our meeting with the Commission on December 8, 2005, several Commissioners expressed the view that the sump screen issue should receive high priority. This was formally stated in the Commission's staff requirements memorandum of December 20, 2005: "... The ACRS shall make among its highest priorities its role in the resolution of GSI-191. ..." At the Commission meeting we indicated that we were waiting to hear status reports from the staff. We have now received several reports, some of them preliminary, and this has enabled us to form an opinion on progress towards resolving GSI-191.

We have written previous letters on the sump screen issue. In particular we raised the matter of chemical effects and questioned some aspects of the NEI guidance which the staff had endorsed.

The staff issued GL 2004-02 on September 13, 2004, and has received responses from all licensees. Though all licensees responded to the generic letter, the staff has concluded that none of the responses was complete. Gaps were evident in all important areas, particularly chemical and downstream effects. The staff has issued requests for additional information (RAIs) relating to several significant effects. Many licensees are finalizing plans to replace the screens before these RAIs are resolved.

While progress has been made in all areas of research, much remains to be done. These programs have produced significant results and are making important contributions to understanding the issues related to PWR sump performance. Many relevant physical and chemical phenomena are being explored. Assessments of other important effects may need to be added to the program.

This research has yet to lead to an ability to develop and validate predictive methods. Much of the work is exploratory in nature, in response to indications that existing analytical capabilities were incomplete and inadequate. The results from some programs are not yet available or are awaiting staff review.

The GL 2004-02 responses and recent research have raised new questions. Present plans by licensees to make hardware changes in their plants are driven by the need to reduce the potential for excessive head loss across sump screens during recirculation. Increasing the screen size will reduce this head loss, but the staff's ability to assess the adequacy of the reduction may be limited by uncertainties in the available knowledge base. In addition, downstream effects may be exacerbated by some screen designs and configurations. The staff needs effective means to evaluate these downstream effects and their influence on core coolability.

DISCUSSION

Industry Response to Generic Letter 2004-02

In general, licensees intend to address the sump screen issue by making a significant increase in the flow areas of the screens. Some designs may also have smaller openings and/or active debris removal mechanisms. Physical changes have already been made in some plants. Modifications to almost all plants are planned to be completed by the end of calendar year 2007. Some licensees have requested extensions until the spring outage of 2008. Each of the five vendors of the new sump screens plans to undertake integrated-effect "proof tests" with screens or segments of screens to demonstrate the ability of the screens to accommodate the anticipated loading of debris with an acceptable pressure drop.

The prediction of debris formation, transport, and impact on core coolability is a very complex technical problem. A number of phenomenological issues must be addressed, either by the development of a predictive capability or by the implementation of engineering solutions that circumvent the more difficult issues. The industry is focusing on engineering approaches that maximize screen area to the extent practical, control of materials that affect the quantity and character of debris generation, and the control of sump chemistry to minimize chemical effects.

Regulatory Approach

The staff intends to undertake eight to ten audits of plant modifications. The scope of the audits will be expanded if the staff encounters problems with the technical adequacy of the planned resolutions.

Because of the "proof test" nature of the planned industrial testing program, it is essential that the staff have a level of understanding and a modeling capability for the underlying phenomena adequate to support their technical review of the licensee results. It is doubtful that the current understanding of these phenomena will be adequate to support such a review. The results of recent research have served to call into question some previous guidelines and assumptions without replacing them with validated, improved methods.

Research Efforts

Research is being performed to address the following phenomena:

- Chemical effects – experiments (Los Alamos National Laboratory (LANL) and Argonne National Laboratory (ANL)) and model development for speciation (Center for Nuclear Waste Research Activities (CNWRA))

- Head loss from debris buildup on screens – experiments (Pacific Northwest National Laboratory (PNNL)) and model development (RES)

- Downstream effects – experiments (LANL)

- Coating debris formation and transport – experiments (Electric Power Research Institute (EPRI), Naval Surface Warfare Center (NSWC))

We have seen only the preliminary results from some of these research efforts. It is premature for us to perform a comprehensive evaluation until all the work is complete. However, several research projects have developed important new quantitative information which reveals the significance of certain phenomena. Understanding of those phenomena has not yet been established to the point where validated predictive tools are available. RES has set a target of the spring of 2006 to bring these activities to a conclusion. This schedule is unrealistic in view of the many unresolved issues.

Chemical effects

Exploratory integrated chemical effects tests (ICET) revealed that some species, particularly aluminum oxyhydroxide and calcium phosphate, can be produced under certain conditions. It was concluded that plant-specific evaluations would be required.

ANL is investigating the interaction between calcium silicate insulation (CalSil) and trisodiumphosphate (TSP), which forms calcium phosphate. A qualitative understanding of the chemical processes has been achieved. Studies of head loss on screens using debris quantities that duplicated earlier LANL tests with no chemical additives showed some variability. When calcium phosphate was produced by adding TSP to CalSil, or calcium chloride to TSP, the pressure drop increased substantially. For example, in one test (ICET3-9) the pressure drop through a fiberglass bed was 0.14 psi at a flow velocity of 0.1 ft/s. When calcium chloride was added in stages to the solution of TSP, the pressure drop eventually rose to 5.2 psi at a flow velocity below 0.02 ft/s. Since the flow regime was probably laminar, for which pressure loss is proportional to flow velocity, this corresponds to an increase in bed resistance by a factor of about 200, amounting essentially to blockage of the screen. Similar results were obtained in Tests 1 and 2.

The results of chemical speciation prediction by codes using chemical equilibrium models and measured corrosion rates are encouraging over the range of species that have been studied. CNWRA found that some ICET results could be matched by adjusting the speciation parameters.

Head Loss Tests

PNNL has been conducting head loss tests with mixtures of fiberglass and CalSil in amounts corresponding to those used in earlier LANL tests. The results in some cases differ significantly from the results obtained by LANL. No distinct pattern is evident though some trends might be inferred. In an extreme case, when the constituents were introduced in a particular way, the head loss was roughly 100 times more than the head loss with a well-mixed debris bed of the same overall composition. These results indicate that the structure of the debris bed and the way in which it is formed can have a huge influence on the head loss. Unless the assumption of a homogeneous bed can be justified, it will be necessary to develop an adequate model for these effects (for plants that intend to retain CalSil) or to find a way to scale them in the proof tests now planned by industry. The alternative of developing theoretical models for the way in which the bed builds up in different parts of the screen over time during a variety of accidents is probably unrealistic and may be beyond the capabilities of present state-of-the-art.

RES has begun development of a theoretical model to predict the head loss in a nonhomogeneous debris bed. Substantiation and validation of such a model would be a major undertaking.

Downstream Effects

Tests conducted by LANL revealed that fine debris, of a size characteristic of the debris expected during energetic loss-of-coolant accidents (LOCAs), would pass through a typical sump screen under some conditions. Unless a debris bed has been established, most particles of CalSil and fine fiberglass pass through the screen. Significant quantities of reflective metallic insulation were observed to pass through under some conditions. In the absence of a detailed model for the history of debris bed development on a screen and the arrival of various constituents as functions of location and time, there are considerable uncertainties about how to apply such results to an actual plant. An order of magnitude calculation, with 5000 ft^3 of debris produced, indicates that about 6% of the debris would fill the typical lower plenum of a reactor vessel, if it settled there and was not transported to the core or filtered by debris catchers below the fuel. The larger the screen, the more open area there is likely to be through which fine debris can pass. Chemical reaction products are also likely to pass through open areas of the screen.

In reply to our subcommittee's questions about the effects of such debris on core coolability, the staff and representatives of the Westinghouse Owners Group (WOG) stated that they thought the core would be adequately cooled in a number of scenarios. However, they presented no physical models or analytical predictions to show a validated, quantitative basis for such conclusions.

Tests by LANL of debris transported to throttle valves have revealed a significant effect on pressure drop. Adequate predictive methods are therefore needed for the amount of this debris which actually reaches these valves, and for the resulting consequences.

Coatings

EPRI is conducting experiments on the formation of debris from qualified and unqualified coatings. The results were not presented at our meetings.

NSWC is conducting some basic tests of terminal velocity and transport of paint chips of various shapes, sizes, and composition. Guidance for use of these data remains to be developed.

What Is Missing

We are not aware of research efforts in several important areas.

The most significant omission appears to be an adequate understanding of the effects of the various debris species which enter the reactor vessel and reach the core. These effects are likely to depend on the LOCA scenario, particularly the location and size of the break, and on the screen design. Although guidance developed by the WOG describes several of the phenomena to be modeled to represent these effects, the WOG apparently leaves the evaluation to engineering judgment and ad hoc model development. Unless these effects can somehow be avoided, there is a need for a comprehensive set of validated tools for representing them. Developing the tools would involve significant experimental and model development efforts.

The proof tests being developed by industry to evaluate new screen designs involve the phenomena described earlier in this letter, as well as others. Synthesizing these evaluations into a defensible method for scaling test results to the actual LOCA scenario is no trivial matter. We have yet to see scaling laws, methods of extrapolation, or theoretical representations (e.g. computational codes) which can make a convincing case that the test results can be applied to the actual plant. For example, one issue is how to use tests on a single module to predict the performance of an array of modules. The Office of Nuclear Reactor Regulation (NRR) may need to draw on further research results in order to evaluate submissions based on these proof tests.

Formation and transport of coating debris are being studied. We have not seen results of work on the effects of this debris on screen head loss. In view of the difficulty of predicting head loss with the existing mix of ingredients, and the surprises that have been encountered, it is necessary to establish a knowledge base for the effects of coatings on head loss by means of an adequate set of experiments and predictive methods.

Research has already revealed that the structure of a debris bed influences head loss and the bypass of fine material. As screens become larger and perhaps have more complex geometry, the variability of bed structure over the surface of the screen is likely to increase. Some areas, such as the base of vertical screens or the outer layers of multiple screens, may be covered by a pile of coarse debris, other areas may support "thin beds" that are blocked by chemical products or fine debris, while some areas may be clear of debris, providing paths through which fine material can pass. There is a need to reduce uncertainty in predicting the performance of these screens under a wide variety of scenarios. Since modeling everything theoretically is impractical, the emphasis should be placed on designing for predictability, supported by data.

THE PATH FORWARD

In response to GL 2004-02, licensees have undertaken the task of showing that they satisfy the requirements of recirculation core cooling. In most cases, the response has been to plan the replacement of sump screens by those with significantly larger area. The hole size and other characteristics of these screens may also be changed.

These changes are in the right direction to alleviate the potential for excessive head loss. However, in view of uncertainties introduced by new research results, the incomplete response by industry to the generic letter, the difficulties of validating the "proof tests" planned by industrial consortia, and downstream effects, NRR will need to develop assurance that it has the capability to evaluate the effects of these changes. The staff anticipates that, if sufficient uncertainty is encountered, supplemental actions may be required. These may include the following measures:

- Removal from containment of constituents that are known to cause problems with head loss and lack of predictability.

- Development of screen designs that are insensitive to the plethora of uncertainties associated with many existing designs. These designs may include active screens or similar devices that can handle many forms of debris without the need for knowing the details of the debris characteristics.

- Design of screens for minimum bypass of fine debris. Emphasis is currently being placed on reducing head loss, but downstream effects should also be considered.

- Identification of other solutions to core cooling that get around the manifold uncertainties associated with the present range of screen designs and can more confidently demonstrate success in meeting specifications.

- Use of probabilistic analysis to show that the most undesirable debris bed configurations are highly unlikely. Evaluation would be based on realistic analysis rather than on a conservative approach.

We endorse the immediate plans to increase the size of sump screens because this will alleviate the potential for excessive head loss. This action by itself may not be sufficient to resolve all long-term core cooling issues.

We anticipate working further with the staff on these important matters.

Dr. William Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission Generic Letter 2004-02: "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors", September 13, 2004.
2. U.S. Nuclear Regulatory Commission Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors", June 9, 2003.
3. Letter from Mario V. Bonaca, Advisory Committee on Reactor Safeguards, "Proposed Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at PWRs," July 19, 2004.
4. "Draft NRC Staff Review Guidance for Evaluation of Downstream Effects of Debris Ingress into the PWR RCS On Long Term Core Cooling Following a LOCA", undated.
5. "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," WCAP-16406-P, Westinghouse Owners Group, June 2005.
6. Information Notice 2005-26: "Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment," September 16, 2005.
7. NRC Information Notice 2005-26, Supplement 1: "Additional Results of Chemical Effects Tests in a Simulated PWR Sump Pool Environment," January 20, 2006.
8. "Integrated Chemical Effects Test Project: Test #1 Data Report," LA-UR-05-124, June 2005.
9. "Integrated Chemical Effects Test Project: Test #2 Data Report," LA-UR-05-6146, September 2005.
10. "Integrated Chemical Effects Test Project: Test #3 Data Report," LA-UR-05-6996, October 2005.
11. "Integrated Chemical Effects Test Project: Test #4 Data Report," LA-UR-05-8735, November 2005.
12. "Integrated Chemical Effects Test Project: Test #5 Data Report," LA-UR-05-9177, January 2006.
13. Memorandum from Michele G. Evans to John N. Hannon, "Final Transmittal of Information Summarizing Integrated Chemical Effects Results and Implications", October 25, 2005.
14. "Corrosion Rate Measurements and Chemical Speciation of Corrosion Products Using Thermodynamic Modeling of Debris Components to Support GSI-191," NUREG/CR-6873, April 2005.
15. "Screen Penetration Test Report," NUREG/CR-6885, LA-UR-04-5416, October 2005.
16. Memorandum from Ralph Architzel to James Lyons, "Report on Results of Staff Pilot Plant Audit– Crystal River Analyses Required for the Response to Generic Letter 2004-02 and GSI-191 Resolution," June 29, 2005.
17. Memorandum from Ralph Architzel to Thomas Martin, "Report on Results of Staff Pilot Plant Audit– Fort Calhoun Station Analyses Required for the Response To Generic Letter 2004-02 and GSI-191 Resolution," January 26, 2006.

May 24, 2006

MEMORANDUM TO: ACRS Members

FROM: R. Caruso, Senior Staff Engineer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS MEMORANDUM CONCERNING AN ANONYMOUS LETTER RELATED TO THE TRACE COMPUTER CODE DEVELOPMENT AND REVIEW PRACTICES

Attached for your information is a copy of the EDO's March 30, 2006 response to the ACRS's memorandum of February 15, 2006, concerning an anonymous letter received by Dr. Wallis and Dr. Ransom. A copy of the Committee's memorandum is also attached.

Committee Memorandum

In its memorandum, the Committee forwarded the anonymous letter to the EDO for possible action and requested that the Committee be kept informed of the staff's disposition. This is the third in a series of anonymous communications received by the ACRS regarding the TRACE code.

EDO Response

The EDO responded that it plans to address these comments in the context of a meeting with the Thermal-Hydraulics Phenomena Subcommittee later this summer. The staff has evaluated the arguments and derivations in this latest letter, and it wishes to discuss those evaluations with the Committee

Analysis

The EDO's response is satisfactory. The meeting of the T-H Subcommittee will be a good opportunity to review the substance of these anonymous comments, and provide new members of the Committee with an introduction to the TRACE code.

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

March 30, 2006

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

FROM: Luis A. Reyes
Executive Director for Operations

SUBJECT: RESPONSE TO ANONYMOUS LETTER CONCERNING THE TRACE COMPUTER CODE DEVELOPMENT AND REVIEW PRACTICES

This is in response to your memorandum, dated February 15, 2006, concerning an anonymous letter received by Dr. Graham B. Wallis (Chairman) and Dr. Victor H. Ransom (Member) of the Advisory Committee on Reactor Safeguards (ACRS). Specifically, that letter raises concern about the numerical solution method used in the TRAC/RELAP5 Advanced Computational Engine (TRACE) computer code (formerly known as TRAC-M), which the U.S. Nuclear Regulatory Commission (NRC) uses to model and analyze two-phase, two-fluid thermal-hydraulic phenomena that occur in nuclear power reactors under accident conditions. This is the third in a series of anonymous letters concerning TRACE that ACRS members have received over the past several years. In response to each of these anonymous letters, the NRC's Office of Nuclear Regulatory Research (RES) evaluated the author's concerns and addressed them through public interaction with the ACRS Thermal-Hydraulics Subcommittee.

Having evaluated the author's arguments and derivations in the latest letter, the staff has again determined that the issues can best be addressed through public interaction with the ACRS. Toward that end, the staff is planning a series of meetings this summer with the Thermal-Hydraulics Subcommittee to discuss the development, assessment, and quality assurance of the TRACE code, with particular emphasis on the concerns expressed in the latest anonymous letter.

You should also note that the TRACE code is currently being qualified by various institutions worldwide, and the staff plans to subject the code to peer review near the end of Fiscal Year (FY) 2007. This peer review should further confirm the staff's view regarding concerns expressed in the latest anonymous letter.

CONTACT: Christopher Murray, RES
301-415-6745

We look forward to working with the ACRS Thermal-Hydraulics Subcommittee to address the arguments and derivations in the latest letter, in an effort to further assess and validate the TRACE code.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY
OGC
OCA
OPA
CFO

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

ACRSR-2176A

February 15, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

/RA/

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: ANONYMOUS LETTER CONCERNING THE TRACE COMPUTER CODE DEVELOPMENT AND REVIEW PRACTICES

On January 10, 2006, Dr. Graham Wallis and Dr. Victor Ransom, both members of the Advisory Committee on Reactor Safeguards (ACRS), received an anonymous letter (enclosed), which describes several issues related to the development, validation, and verification of the TRACE reactor systems analysis code. On March 8, 2004, we provided you with a copy of an anonymous e-mail on the same subject that was sent to Dr. Wallis. On October 14, 2004, we provided you with a copy of an anonymous letter that was sent to Dr. Wallis and Dr. Ransom. The ACRS intends to consider the comments in the letter as it continues its review of the TRACE code.

We are forwarding the letter to you for any possible action. The Committee would like to be informed of your disposition of this matter.

Enclosure: As stated

cc: ACRS Members
C. Paperiello, RES
J. Dyer, NRR
P. Baranowsky, RES
F. Eltawila, RES
W. Burton, RES
G. Mulley, OIG

Professor G. B. Wallis
Professor V. Ransom
ACRS on Thermal Hydraulics
US NRC
11545 Rockville Pike
Rockville, MD 20852

SUBJECT: Additional Information About the Incorrect EOS Solution in the TRAC/TRACE Codes

Dear Professors Wallis and Ransom:

Additional information regarding the incorrect equation of state solution (EOS) approach used in the TRAC/TRACE computer codes is enclosed. The material includes a demonstration that the approach in the codes does not in any way represent a 'solution' of any kind for the EOS. The approach is not a mathematically correct linearization of the EOS and does not represent any correct or reasonable engineering approximation.

The demonstration is based on an EOS that admits to analytical solutions. The results show that (1) the approach in TRAC/TRACE is not correct and cannot lead to the correct analytical solution, (2) the correct linearization by use of implicit function theory leads to the correct solution, and (3) implicit function theory leads to the exact derivatives as obtained directly from the analytical EOS. While a single demonstration does not prove a theorem, reasonable minds should be able to agree on the fundamental issues involved in such a straightforward application of mathematical theory to practical engineering applications. The method to obtain a correct solution of the EOS is a no-brainer.

Given that variations of the TRAC codes (PWR and BWR) have been the basis of calculations submitted for review and evaluation by the NRC, I think this issue requires significant attention by both the NRC and the ACRS. It may be that the code variations have been based on numerical solution methods prior to the SETS methods. At one time in the past, the TRAC code was based on a simple and straightforward numerical solution method.

Thank you for your attention to this matter.

Exact EOS Solution for Perfect Gas vs. TRACE Method

0.1 Objectives

The objectives of these notes include providing (1) additional information about the application of implicit function theory to the solution of the equation of state (EOS), (2) a demonstration that the solution approach to the EOS as used in TRAC/TRACE is not correct. Other important concepts will be introduced as the developments are presented in the following discussions. The discussions are based around a simple example, but nonetheless a case that appears in the TRAC/TRACE applications. We look at the EOS solution for single-phase flow and additionally, that situation is further specialized to the case of a perfect gas.

0.2 The Single-Phase Flow Case

The TRAC/TRACE codes solve for the average "partial" density and density-energy product in the mass and energy cells. And as discussed in previous notes and buried in the TRAC/TRACE manuals the following system of equations are set up to determine the pressure and temperature. The equations are

$$F_1(P, T) = \bar{\rho} - \rho(P, T) = 0 \tag{0-1}$$

for the mass-conservation solution, and

$$F_2(P, T) = \overline{\rho e} - \rho(P, T)e(P, T) = 0 \tag{0-2}$$

for energy conservation. These equations might not be used in the codes, but they provide a straightforward case that is exactly analogous to what is in the codes. Additionally, they should be used for the correct solution for the EOS for single-phase flows.

The objective is to obtain the value of the pressure, P, and temperature, T, at the new-time level. The method used in the codes along with the correct methods are developed in these notes.

Several derivatives of the fluid-state properties are needed throughout the developments. All such derivatives, which lead to thermophysical properties of the fluid, can be expressed in terms of the following basic properties.

The *specific heat at constant pressure*

$$C_P = \left(\frac{\partial h}{\partial T}\right)_P \tag{0-3}$$

the *specific heat at constant volume*

$$C_v = \left(\frac{\partial e}{\partial T}\right)_v \tag{0-4}$$

the *coefficient of isothermal compressibility*

$$\kappa = \frac{1}{\rho}\left(\frac{\partial \rho}{\partial P}\right)_T \tag{0-5}$$

and the *coefficient of volume expansion*

$$\beta = -\frac{1}{\rho}\left(\frac{\partial \rho}{\partial T}\right)_P \tag{0-6}$$

The specific heats are related by

$$C_P = C_v + \frac{Tv\beta^2}{\kappa} \tag{0-7}$$

which is especially useful in the form

$$\rho C_v \kappa = \rho C_P \kappa - T\beta^2 \tag{0-8}$$

For a perfect gas Equ (0-7) gives

$$C_P = C_v + R \tag{0-9}$$

Additionally, the specific internal energy and the enthalpy are related by

$$h = e + P/\rho \tag{0-10}$$

and the ratio of the specific heats will be denoted by

$$\gamma = C_P/C_v \tag{0-11}$$

The derivatives that are needed are summarized in the following table

EOS Derivative	General Case	Perfect Gas
$\left(\frac{\partial\rho}{\partial P}\right)_T$	$\rho\kappa$	$\frac{\rho}{P}$
$\left(\frac{\partial\rho}{\partial T}\right)_P$	$-\rho\beta$	$-\frac{\rho}{T}$
$\rho\left(\frac{\partial\rho}{\partial P}\right)_T + e\left(\frac{\partial\rho}{\partial P}\right)_T$	$\rho h\kappa - T\beta$	$\frac{1}{\gamma - 1}$
$\rho\left(\frac{\partial\rho}{\partial T}\right)_P + e\left(\frac{\partial\rho}{\partial T}\right)_P$	$\rho(C_P - h\beta)$	0

The TRAC/TRACE Approach

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At the end of a time step the EOS is used as follows in the codes. Equations (0-1) and (0-2) are written as a system of two equations for two unknowns as

$$F_1^{n+1} = F_1^n + \left(\frac{\partial F_1}{\partial P}\right)^n (\Delta P)^{n+1} + \left(\frac{\partial F_1}{\partial T}\right)^n (\Delta T)^{n+1} = 0 \tag{0-12}$$

and

$$F_2^{n+1} = F_2^n + \left(\frac{\partial F_2}{\partial P}\right)^n (\Delta P)^{n+1} + \left(\frac{\partial F_2}{\partial T}\right)^n (\Delta T)^{n+1} = 0 \tag{0-13}$$

And as outlined in some previous notes, a single pass through the standard and correct solution process is used to get a numerical value for one of the solution variables. In the case of the TRAC/TRACE codes the equation system is written to get a value for the void fraction. While the two-fluid case could be outlined here, the single-phase case is much easier to follow and demonstrates exactly what is wrong with the approach used in the codes.

Using the information given in the table above, the derivatives needed for Equs. (0-12) and (0-13) are obtained easily. The two-by-two system for the change in the pressure and temperature over the time step are solved analytically to get

$$(\Delta P)^{n+1} = -\frac{1}{D}\begin{vmatrix} (\bar{\rho} - \rho(P,T)) & \rho\beta \\ (\overline{\rho e} - \rho(P,T)e(P,T)) & \rho(h\beta - C_p) \end{vmatrix} \tag{0-14}$$

for the pressure, and

$$(\Delta T)^{n+1} = -\frac{1}{D}\begin{vmatrix} -\rho\kappa & (\bar{\rho} - \rho(P,T)) \\ (T\beta - \rho h\kappa) & (\overline{\rho e} - \rho(P,T)e(P,T)) \end{vmatrix} \tag{0-15}$$

where the determinant of the coefficient matrix is

$$D = \begin{vmatrix} \frac{\partial F_1}{\partial P} & \frac{\partial F_1}{\partial T} \\ \frac{\partial F_2}{\partial P} & \frac{\partial F_2}{\partial T} \end{vmatrix} = \frac{\partial F_1}{\partial P}\frac{\partial F_2}{\partial T} - \frac{\partial F_1}{\partial T}\frac{\partial F_2}{\partial P} \tag{0-16}$$

For the general case, using the information in the table above, the determinant evaluates to

$$D = \rho^2 C_v \kappa \tag{0-17}$$

and for the perfect gas case to

$$D = \frac{\rho}{T(\gamma - 1)} \tag{0-18}$$

In order to save on the typing, the superscript has been omitted from all the results in Equs (0-14) through (0-18). Additionally, what is actually done in the codes to get numerical values for all these quantities is not known.

In the codes, as mentioned above, only a single pass through the EOS solver is used. This exactly corresponds to the equations written above in the sense that we have not indicated that any iterative method is used to solve the non-liner equations. note that almost all the quantities in the equations are functions of the state that is being calculated and thus iterative methods are called for if the correct solution is sought.

Given the information in Equs (0-14) through (0-18) the change in the pressure and temperature over the time step are easily obtained. To make it explicitly clear what is happening in the codes these solutions are written here

$$(\Delta P)^{n+1} = -\frac{1}{D}\{[\bar{\rho} - \rho(P,T)]\rho(h\beta - C_p) - \rho\beta[\overline{\rho e} - \rho(P,T)e(P,T)]\} \tag{0-19}$$

for the pressure, and

$$(\Delta T)^{n+1} = \frac{1}{D}\{\rho\kappa[\overline{\rho e} - \rho(P, T)e(P, T)] + (T\beta - \rho h\kappa)[\bar{\rho} - \rho(P, T)]\} \tag{0-20}$$

where a sign has been factored through the latter equation.

These results correspond exactly to those that are used in the codes as 'solution' for the EOS at the new-time level. At this point in the codes one of the results is taken as the new-time value and the other is discarded. Here are the important aspects of these results. Note that the functions for which the zeros are sought, the terms inside the square brackets in Equs (0-19) and (0-20), are in the results. The quantities $\bar{\rho}$ and $\overline{\rho e}$ have been obtained by solutions of the finite-difference equations for mass and energy. The numerical values for the density and internal energy are most likely the values from the previous time step. The same is true for all the other state properties on the right-hand side of the equations above. Thus, as noted in previous notes, the new-time 'solution' in the codes will depend on whatever is stored in these locations in the codes. No attempt is made to update these values even once.

The fact that the approach used in the codes does not begin to have any basis whatsoever in mathematics or engineering can be easily demonstrated as follows. It is not a linearization of the non-linear EOS. The very purpose of the Newton-Raphson method for solution of non-linear equations is to drive the functions to zero; see Equs (0-1) and (0-2). Thus as the solution is approached the functions in the square brackets on the right-hand side of Equs (0-19) and (0-20) approach zero. One of the convergence checks of the procedure is to check the magnitude of $(\Delta X)^{n+1}$ $X = P, T$, so it must decrease as the iterations increase and in the end is not a part of the solutions. The stark contrast with the 'solution' in the codes is self evident.

While the above discussion should be very clear and leave no possibility for hand-waving work-arounds and dismissal, additional demonstrations is given in the following discussions. The case of a perfect gas, for which analytical solutions can be obtained, is discussed. The correct analytical solutions are compared with the results given by implicit function theory, which was recommended in previous notes as the mathematically sound basis for solution of the EOS in TRAC/TRACE. A brief summary of implicit function theory starts the discussions.

0.3 Implicit Function Theory

The correct linearization of the EOS solution is based on the same functions that are used in TRAC/TRACE; Equs (0-1) and (0-2). These equations implicitly define the thermodynamic state properties as functions of the code solution variables $\bar{\rho}$ and $\overline{\rho e}$. The thermodynamic state properties are thus written

$$P = P(\bar{\rho}, \overline{\rho e}) \tag{0-21}$$

for the pressure, P, and

$$T = T(\bar{\rho}, \overline{\rho e}) \tag{0-22}$$

for the temperature, T. Linearization of the dependent properties over a time step then gives

$$(\Delta P)^{n+1} = \left(\frac{\partial P}{\partial \bar{\rho}}\right)^{n}_{\overline{\rho e}}(\Delta\bar{\rho})^{n+1} + \left(\frac{\partial P}{\partial \overline{\rho e}}\right)^{n}_{\bar{\rho}}(\Delta\overline{\rho e})^{n+1} \tag{0-23}$$

for the pressure, and

$$(\Delta T)^{n+1} = \left(\frac{\partial T}{\partial \bar{\rho}}\right)^{n}_{\overline{\rho e}}(\Delta\bar{\rho})^{n+1} + \left(\frac{\partial T}{\partial \overline{\rho e}}\right)^{n}_{\bar{\rho}}(\Delta\overline{\rho e})^{n+1} \tag{0-24}$$

for the temperature. Equations (0-23) and (0-24) are the mathematically exact linearizations of the thermodynamic equation of state. Note the difference between these equations and those used in the TRAC/TRACE approach given by Equs (0-19) and (0-20). The correct linearization which is given by Equs (0-23) and (0-24) here does not involve the functions.

The derivatives on the right-hand sides of Equs (0-23) and (0-24)are not available from equations of state for materials. The derivatives are obtained by application of implicit function theory as given in the following paragraphs.

The Jacobian for the functions is

$$J = \frac{\partial(F_1, F_2)}{\partial(P, T)} = \begin{vmatrix} -\rho\kappa & \rho\beta \\ (T\beta - \rho h\kappa) & \rho(h\beta - C_p) \end{vmatrix} \tag{0-25}$$

Note that this is the same determinant and value as Eq (0-15) used in the solution of the system by Cramer's method above. Its value has been given for the general case and the perfect-gas case by Equs (0-17) and (0-18), respectively.

Implicit function theory needs the derivatives of the functions with respect to the code solution variables $\bar{\rho}$ and $\overline{\rho e}$. These are

$$\begin{aligned} \frac{\partial F_1}{\partial \bar{\rho}} &= 1 \qquad \frac{\partial F_1}{\partial \overline{\rho e}} = 0 \\ \frac{\partial F_2}{\partial \bar{\rho}} &= 0 \qquad \frac{\partial F_2}{\partial \overline{\rho e}} = 1 \end{aligned} \tag{0-26}$$

The derivatives of the thermodynamic state properties with respect to the code-solution variables are given by implicit function theory as follows

$$\left(\frac{\partial P}{\partial \bar{\rho}}\right)_{\overline{\rho e}} = -\frac{1}{D}\begin{vmatrix} 1 & \rho\beta \\ 0 & \rho(h\beta - C_p) \end{vmatrix} = \frac{\rho}{J}(C_p - h\beta) \tag{0-27}$$

$$\left(\frac{\partial P}{\partial \overline{\rho e}}\right)_{\bar{\rho}} = -\frac{1}{D}\begin{vmatrix} 0 & \rho\beta \\ 1 & \rho(h\beta - C_p) \end{vmatrix} = \frac{\rho\beta}{J} \tag{0-28}$$

for the pressure, and

$$\left(\frac{\partial T}{\partial \bar{\rho}}\right)_{\overline{\rho e}} = -\frac{1}{D}\begin{vmatrix} -\rho\kappa & 1 \\ (T\beta - \rho h\kappa) & 0 \end{vmatrix} = \frac{T\beta - \rho h\kappa}{J} \tag{0-29}$$

$$\left(\frac{\partial T}{\partial \overline{\rho e}}\right)_{\bar{\rho}} = -\frac{1}{D}\begin{vmatrix} -\rho\kappa & 0 \\ (T\beta - \rho h\kappa) & 1 \end{vmatrix} = \frac{\rho\kappa}{J} \tag{0-30}$$

for the temperature.

These can be evaluated for the general case by putting Equ (0-17) for J. For the perfect-gas case, J is given by Equ (0-18) and the last (lower right) entry in the matrix for Equs (0-27) and (0-28) is zero; see the table above in these notes. Carrying out the perfect-gas case gives the four derivatives of the thermodynamic state properties with respect to the code solution variables to be

$$\begin{aligned} \left(\frac{\partial P}{\partial \bar{\rho}}\right)_{\overline{\rho e}} &= 0 \\ \left(\frac{\partial P}{\partial \overline{\rho e}}\right)_{\bar{\rho}} &= (\gamma - 1) \\ \left(\frac{\partial T}{\partial \bar{\rho}}\right)_{\overline{\rho e}} &= -\frac{T}{\rho} \\ \left(\frac{\partial T}{\partial \overline{\rho e}}\right)_{\bar{\rho}} &= \frac{T}{P}(\gamma - 1) \end{aligned} \tag{0-31}$$

The correct linearized estimate for the new-time values for the pressure and temperature are obtained by the results from Equs (0-27) through (0-30) into Equs (0-23) and (0-24) and using the code-calculated values for the change in the solution variables over the time step. The correct linearized EOS solution for the general case is then

$$(\Delta P)^{n+1} = \frac{1}{(\rho^2 C_v \kappa)^n}\left[(C_p - h\beta)^n (\Delta\bar{\rho})^{n-1} + (\rho\beta)^n (\Delta\overline{\rho e})^{n+1}\right] \tag{0-32}$$

for the pressure, and

$$\Delta T)^{n+1} = \frac{1}{(\rho^2 C_v \kappa)^n}[(T\beta - \rho h\kappa)^n (\Delta\bar{\rho})^{n+1} + (\rho\kappa)^n (\Delta\overline{\rho e})^{n+1}. \tag{0-33}$$

for the temperature. The perfect-gas results of Equ (0-31) are put into Equs (0-23) and (0-24) to get the corresponding correctly linearized EOS for that case.

We next show that having carried out the application of implicit function theory to the EOS the same results are obtained directly from the perfect gas EOS. The perfect gas is described by

$$P = \bar{\rho}RT \tag{0-34}$$

and for the energy

$$e = C_v T \tag{0-35}$$

so that the density-internal energy product is

$$\overline{\rho e} = \frac{P}{(\gamma - 1)} \tag{0-36}$$

And writing these explicitly in the form of the dependent variables as functions of the independent variables gives

$$P = (\gamma - 1)\overline{\rho e} \tag{0-37}$$

for the pressure, and

$$T = \frac{(\gamma - 1)\overline{\rho e}}{R\bar{\rho}} \tag{0-38}$$

for the temperature. The derivatives of the thermodynamic state properties with respect to the code-calculated variables obtained directly from Equs (0-37) and (0-38) are thus exactly those given in Equ (0-31) above. These latter being obtained through application of implicit function theory.

0.4 Summary and Conclusions

In these notes we have shown the following:

(1) The approach to the 'solution' of the equation of state in the TRAC/TRACE codes does not have any correct basis whatsoever in either mathematics or engineering.

(2) The correct solution is obtained by use of implicit function theory.

(3) The validity of (2) has been demonstrated by application of implicit function theory to a simple EOS model for a perfect gas.

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

May 25, 2006

MEMORANDUM TO: George Apostolakis, Chairman
Reliability and PRA Subcommittee

FROM: Michael A. Junge, Senior Staff Engineer
Advisory Committee on Reactor Safeguards

SUBJECT: ANALYSIS OF EDO RESPONSE TO THE ACRS LETTER, DATED APRIL 20, 2006, CONCERNING THE DRAFT FINAL REGULATORY GUIDE 1.205, "RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION FOR EXISTING LIGHT-WATER NUCLEAR POWER PLANTS"

Attachment 1 contains a copy of the Executive Director for Operations (EDO) May 18, 2006 response to the Advisory Committee on Reactor Safeguards (ACRS) April 20, 2006 letter regarding draft final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants." Attachment 2 contains a copy of the Committee letter.

Recommendation 1

RG 1.205 should be issued after the peer-review guidance is clarified.

EDO Response

The Committee comments were appreciated and the EDO acknowledged that Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants", be issued conditioned upon Recommendations 2 and 3.

Analysis

The EDO agrees with the ACRS recommendation.

Recommendation 2

RG 1.205 should be revised to make clear that in cases where licensees elect to rely on information contained in an internal-event Probabilistic Safety Assessment (PSA) or other analyses such as Individual Plant Examinations of External Events (IPEEE) to quantify risk associated with fires, these analyses should be peer reviewed.

EDO Response

The Office of Nuclear Reactor Regulation enhanced Sections B ("Discussion, Fire PSA") and 4.3 ("Fire PSA/PRA") to indicate that the need for peer review applies across the entire spectrum of risk analysis methods that may be loosely phrased as "fire PSA," including analyses based on an internal-event PSA or fire IPEEE.

Analysis

The EDO agrees with the ACRS recommendation.

Recommendation 3

The staff should develop models for human performance that focus on the probability distribution of the time to complete a recovery action under specified conditions.

EDO Response

As the ACRS notes, the USNRC Human Reliability Analyses (HRA) models (ATHENA and SPAR-H) do not focus on the probability distribution of the availability time required to complete an action under specified conditions for evaluating the reliability of operator recovery actions under specified conditions appropriate to fire. The Office of Nuclear Regulatory Research (RES) will consider incorporating this issue in the "HRA Research Program Plan" as part of their ongoing effort to improve HRA techniques for Human Performance in all areas of nuclear power plant operation, including fires. RES plans to meet with the ACRS to discuss this topic in order to develop a better understanding of this specific recommendation and plan accordingly.

Analysis

The EDO agrees with the ACRS recommendation.

RECOMMENDATION

The EDO response is satisfactory for Recommendations 1 through 3.

cc: ACRS Members
J. Larkins
A. Thadani
E. Thornsberry
M. Snodderly
S. Duraiswamy

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

May 18, 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: DRAFT FINAL REGULATORY GUIDE 1.205, "RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION FOR EXISTING LIGHT-WATER NUCLEAR POWER PLANTS"

Dear Dr. Wallis:

I am writing in response to your letter of April 20, 2006, which summarized the results of the Advisory Committee on Reactor Safeguards' (ACRS') review of the subject draft final regulatory guide. We appreciate the Committee's recommendation that Regulatory Guide 1.205 (RG 1.205) be issued, conditioned on the following: (1) clarification of the peer-review guidance, namely that "in cases where licensees elect to rely on information contained in an internal-event PSA [probabilistic safety assessment] or other analyses such as IPEEEs [individual plant examinations of external events] to quantify risk associated with fires, these analyses should be peer reviewed"; and (2) "staff [development of] models for human performance that focus on the probability distribution of the time to complete a recovery action under specified conditions." Both of these recommendations have been addressed, as discussed below.

In response to your first recommendation, the Office of Nuclear Reactor Regulation enhanced Sections B ("Discussion, Fire PSA") and 4.3 ("Fire PSA/PRA") to indicate that the need for peer review applies across the entire spectrum of risk analysis methods that may be loosely phrased as "fire PSA," including analyses based on an internal-event PSA or fire IPEEE. In response to your second recommendation, as the ACRS notes, the U. S. Nuclear Regulatory Commission Human Reliability Analysis (HRA) models (ATHEANA and SPAR-H) do not focus on the probability distribution of the available time required to complete an action under specified conditions for evaluating the reliability of operator recovery actions under specified conditions appropriate to fire. The Office of Nuclear Regulatory Research (RES) will consider incorporating this issue in the "HRA Research Program Plan" as part of their ongoing effort to improve HRA techniques for human performance in all areas of nuclear power plant operation, including fires. RES plans to meet with the ACRS to discuss this topic in order to develop a better understanding of this specific recommendation and plan accordingly.

We look forward to issuing RG 1.205 and thank the Advisory Committee for Reactor Safeguards for its endorsement.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

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

April 20, 2006

Mr. Luis A. Reyes
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL REGULATORY GUIDE 1.205, "RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION FOR EXISTING LIGHT-WATER NUCLEAR POWER PLANTS"

Dear Mr. Reyes:

During the 531st meeting of the Advisory Committee on Reactor Safeguards, April 5 - 7, 2006, we reviewed draft final Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants." We Issued a letter on a previous version of this Regulatory Guide on June 14, 2005, and discussed the staff's proposed response to this letter during the 526th meeting on October 6-8, 2005. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. RG 1.205 should be issued after the peer-review guidance is clarified.

2. RG 1.205 should be revised to make clear that in cases where licensees elect to rely on information contained in an internal-event Probabilistic Safety Assessment (PSA)[1] or other analyses such as Individual Plant Examinations of External Events (IPEEE) to quantify risk associated with fires, these analyses should be peer reviewed.

3. The staff should develop models for human performance that focus on the probability distribution of the time to complete a recovery action under specified conditions.

BACKGROUND AND DISCUSSION

The National Fire Protection Association (NFPA) issued a performance-based standard for fire protection for light-water reactors in 2001 (NFPA 805). 10 CFR 50.48 (c) allows licensees to voluntarily adopt and maintain a fire protection program that meets the requirements of NFPA 805 as an alternative to meeting the requirements of 10 CFR 50.48 (b). NEI has worked with representatives of the industry and the NRC staff to develop implementing guidance for the specific provisions of NFPA 805 and 10 CFR 50.48 (c). In April 2005, NEI published this guidance as NEI 04-02, Revision 0. By memorandum dated May 3, 2005, the staff sent to us the draft final Regulatory Guide for our review.

In our June 14, 2005 letter, we recommended that the draft final Regulatory Guide not be issued. The main reason for this recommendation was that the proposed methods in NEI 04-02, Revision 0 for risk-informed decisionmaking were not based on a fire PSA. In a letter dated August 2, 2005, the staff agreed with the principal argument of our letter and stated that it would work with NEI to ensure that the parts of NEI 04-02, Revision 0 that the staff endorses use correct methodology and language.

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[1]The terms "Probabilistic Safety Assessment" and "Probabilistic Risk Assessment" (PRA) are treated as synonymous in the regulatory guide.

NEI issued Revision 1 to NEI 04-02, in September 2005. The March 2006 version of the draft final RG 1.205 endorses the revised NEI report with the exception of Section 6. These documents have satisfactorily addressed the principal concerns that we expressed in our June 14, 2005 letter.

Plant-specific fire PSAs have shown that fires can be among the major contributors to risk. We believe that any changes to the fire protection program that claim to be risk informed should be based on a rigorous peer-reviewed, plant-specific fire PSA.

In the Background Section of RG 1.205, the staff states that it anticipates that licensees will develop a fire PSA and that, without it, licensees "will not realize the full safety and cost benefits of transitioning to NFPA." In Section 3.2.3, the staff states that, "for PSA-based methodologies," license amendment requests should include an explanation of why the fire PSA is considered technically adequate, as well as a description of the associated peer review. However, 10 CFR 50.48 (c) permits license amendment requests that are not based on a fire PSA. Such requests will have to be based on information in an internal-event PSA or an IPEEE to quantify risk associated with fires. RG 1.205 now appears to indicate that the staff would accept such alternative analyses without a peer review. The staff has agreed to clarify the RG to make clear that a peer review should be conducted for these alternative analyses. After clarifying the guidance for peer review, RG 1.205 should be issued.

RG 1.205 also addresses operator manual actions. If such actions are credited in lieu of Appendix R requirements and have not been approved by the NRC, then they must be treated as plant changes. Section B.2.2.4 of NEI 04-02, Revision 1 states: "The reliability of the recovery action should be commensurate with its risk-significance." The NEI document specifies that, in evaluating this reliability, "the amount of time available to the licensee to complete the recovery action versus the time to actually complete the action should be considered and evaluated." The evaluation should also consider the uncertainties associated with "(i) human performance, (ii) the difference between field verification conditions and actual environmental and fire conditions, and (iii) design basis (e.g., thermal hydraulic analysis) versus actual time constraints."

We agree with these statements. However, we note that their implementation would be facilitated by human reliability models that focus on the probability distribution of the time required to complete a certain action under specified conditions. Neither of the NRC models for human performance (ATHEANA and SPAR-H) focuses on this distribution. They instead treat the available time as just one of many performance shaping factors. The staff should work with the human reliability analysis experts in the Office of Nuclear Regulatory Research to develop appropriate models for evaluating the reliability of operator recovery actions.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," March 2006 (ADAMS Accession No. ML060600183).
2. NEI 04-02, Revision 1, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)," September 2005 (ADAMS Accession No. ML052590476).
3. Letter from the EDO to Dr. Wallis, dated August 2, 2005, Subject: Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (ADAMS Accession No. ML051940255).
4. Letter from Dr. Wallis to the EDO, dated June 14, 2005, Subject: Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (ADAMS Accession No. ML051650432).
5. Memo from M. Salley, RES, to S. Weerakkody, NRR, "Transmittal of Fire Risk Analysis Review Guidance in Support of NFPA 805 Based Changes to the Fire Protection Program" dated January 12, 2006 (ADAMS Accession No. ML060120449).

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

May 30, 2006

MEMORANDUM TO: ACRS Members

FROM: R. Caruso, Senior Staff Engineer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER CONCERNING STANDARD REVIEW PLAN SECTION 14.2.1, "GENERIC GUIDELINES FOR EXTENDED POWER UPRATE TESTING PROGRAMS"

Attached for your information is a copy of the EDO's May 15, 2006 response to the ACRS's letter of April 19, 2006, concerning the Committee's response to another EDO letter of February 22, 2006, concerning the proposed revised SRP Section 14.2.1. A copy of the Committee's letters and the EDO responses are also attached.

Committee Letter

In its letter, the Committee responded to a staff statement that plant-specific issues can influence a decision for large transient testing, and it is not practical or even feasible to improve the SRP decision logic. The Committee examined the proposed logic, and noted that although the factors presented in the SRP are appropriate, there is little guidance provided to the reviewer as to standards of acceptance. The Committee suggested a more structured approach with specific steps to arrive at a determination whether a particular test should be performed, and it requested that the staff meet with the Committee to discuss some approaches to improving the SRP.

EDO Response

The EDO replied that the staff believes that the existing SRP guidance is adequate and sufficient, but the staff would welcome the opportunity to meet with the Committee to further discuss the respective points of view and reach a common understanding of this issue.

Analysis

The EDO's response is troubling. The staff has agreed to meet with the Committee, but the terse statement that the existing guidance is "adequate and sufficient" is provocative, to say the least. The staff will arrange for a future meeting to discuss this issue, but I am not sure that they will ever change their thinking on this.

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

May 15, 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: STANDARD REVIEW PLAN, SECTION 14.2.1, "GENERIC GUIDELINES FOR EXTENDED POWER UPRATE TESTING PROGRAMS"

Dear Dr. Wallis:

In your letter dated April 19, 2006, you stated that the seven factors in Section III.C of the Standard Review Plan (SRP) provide little guidance to U.S. Nuclear Regulatory Commission (NRC) reviewers for the standards of acceptance. You also provided an example of a structured decision process and suggested a meeting with the NRC staff to discuss approaches to improve the SRP.

The staff believes that the existing SRP guidance is adequate and sufficient. However, we would welcome the opportunity to meet with the Advisory Committee on Reactor Safeguards to further discuss our respective points of view and to reach a common understanding of this issue. The staff will schedule the meeting, as you suggested, in the near future.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

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

April 19, 2006

Mr. Luis A. Reyes
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESPONSE TO YOUR MARCH 29, 2006 LETTER REGARDING STANDARD REVIEW PLAN, SECTION 14.2.1, "GENERIC GUIDELINES FOR EXTENDED POWER UPRATE TESTING PROGRAMS"

Dear Mr. Reyes:

In our letter dated February 22, 2006, we provided the following recommendation on Standard Review Plan (SRP) Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs:"

> Paragraph III.C of SRP Section 14.2.1 should be rewritten to provide more structured and explicit guidance defining those conditions under which large transient tests would be exempted or required.

In your March 29, 2006 response, you stated that plant-specific issues can influence a decision for large transient testing. As a result, the staff concluded that it is not practical or even feasible, to improve the SRP decision logic.

Large transient tests have special objectives. They test not only the performance of individual components and structures but also the integrated response of the system, including control functions. Because large transient tests impose substantial hydrodynamic and thermal loads on the plant, they have impacts on the plant risks. Although these risk impacts are not substantial, it is appropriate to exempt the licensee from performing the tests if they provide little benefit. Conversely, transient tests can identify the unexpected. It would be preferred to uncover issues within the context and precautions of a controlled test, rather than during an unplanned transient.

Section 14.2.1 of the SRP identifies the following seven factors to consider in determining whether a licensee should be exempted from performing a test:

- Power uprate operating experience
- Introduction of new thermal-hydraulic phenomena or identified system interactions
- Facility conformance to limitations associated with computer modeling and analytical methods
- Plant operator familiarization with facility operation and trial use of operating and emergency operating procedures
- Minimal reductions in the margin of safety
- Guidance contained in vendor topical reports
- Risk implications

Although it is appropriate to consider these factors, there is little guidance provided to the reviewer as to standards of acceptance.

We understand that plant-specific considerations could impact the decision process. However, a structured decision process does not have to be rigid. The process does not make the decision; it is an aid to the decision. It is practical and feasible to develop such a logical structure without constraining the ability of the staff to include plant specific considerations. An example of such a structure follows:

- Identify each large transient test and associated objectives from the initial startup program.
- Determine which systems, operations, system interactions, and procedures are changed by the uprate.
- Assess whether the plant modifications or changes affect the conclusions of the initial start-up tests. If not, these tests would not have to be performed.
- Identify any new tests that would be required to verify the proper operation of any modified or new equipment.
- Determine whether other tests will be performed that will ensure that each modified component will perform as intended. If not, a transient test would be expected.
- Assess whether there are multiple modified components, such that the system is effectively new. If so, a transient test would be expected.
- Assess whether analytic modeling capability encompasses the changed range of parameters. If not, a transient test would be expected.
- Assess whether physical phenomena or system interactions could be substantially affected by the change (e.g., potential lifting of relief valves or water level rising to steamline). If so, a transient test would be expected.
- Determine whether the range of system conditions falls within the history of previous power uprates. If not, a transient test would be expected.

We would appreciate the opportunity to meet with the staff to discuss approaches to improving SRP Section 14.2.1.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Letter from L. Reyes, EDO, to G. Wallis, ACRS, Subject: Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," dated March 29, 2006 (ADAMS Accession No. ML060680235).

2. Letter from G. Wallis, ACRS, to L. Reyes, EDO, Subject: Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," dated February 22, 2006 (ADAMS Accession No. ML060530320).

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

May 30, 2006

MEMORANDUM TO: ACRS Members

FROM: R. Caruso, Senior Staff Engineer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER CONCERNING THE APPLICATION OF THE TRACG COMPUTER CODE TO EVALUATE THE STABILITY OF THE ESBWR

Attached for your information is a copy of the EDO's May 22, 2006 response to the ACRS's letter of April 21, 2006, concerning the Committee's review of the staff's draft safety evaluation report related to the use of the TRACG computer code to evaluate the stability of the ESBWR. A copy of the Committee's letter is also attached.

Committee Letter

In its letter, the Committee recommended that the staff should approve the use of TRACG to analyze the stability of the ESBWR during normal operation, anticipated operational occurrences, and the low-power phase of reactor startup. The Committee also noted that it is looking forward to considering the application of this computer code during the review of the ESBWR itself, and commented that it was especially interested in seeing how various modeling parameters would be applied

EDO Response

The EDO accepted the recommendation of the Committee, and it committed to discuss the results of the application of this code during the design certification review during a future meeting.

Analysis

The EDO's response is satisfactory.

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

May 22, 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: APPLICATION OF THE TRACG COMPUTER CODE TO EVALUATE THE STABILITY OF THE ECONOMIC SIMPLIFIED BOILING WATER REACTOR

Dear Dr. Wallis:

Thank you for your April 21, 2006, letter presenting the recommendations and conclusions of the Advisory Committee on Reactor Safeguards (ACRS or the Committee) with regard to the staff's draft safety evaluation report (SER) related to the use of the TRACG computer code to evaluate the stability of the economic simplified boiling water reactor (ESBWR). The ACRS Thermal-Hydraulic Phenomena Subcommittee reviewed TRACG for the stated purpose on January 19 and March 14, 2006; during which time, General Electric (GE) and the Nuclear Regulatory Commission (NRC) staff presented and discussed their studies and conclusions.

In your letter, you agreed with the NRC staff's decision to approve the use of TRACG to analyze the stability of the ESBWR during normal operation, anticipated operational occurrences (AOOs), and the low-power phase of reactor startup. You stated that GE and the NRC staff presented detailed calculations and addressed your questions on the modeling capability of TRACG for stability purposes, modeling of two-phase flow in the chimney of an ESBWR reactor, modeling of natural circulation oscillations, and other topics as presented in your letter. You also stated that the staff performed several useful confirmatory analyses. These included runs of the LAPUR and RELAP5 codes, the use of a drift-flux void propagation model, and sensitivity studies to confirm TRACG's robustness. You expressed the expectation that as part of the ESBWR design certification, GE and the NRC staff will continue to evaluate the artificial attenuation of void waves if the Courant number is not close to one, as the use of a low Courant number may lead to numerical diffusion. The NRC staff will discuss the results of the design certification review during a future meeting.

We appreciate the Committee's timely review of this matter. The Committee's comments throughout the ESBWR design certification application have been useful in the staff's review efforts and greatly benefitted the staff in the finalization of the SER. As you recommended, we intend to proceed and approve the use of the TRACG computer code to evaluate stability of the ESBWR design.

Sincerely,

Luis Reyes
Executive Director
for Operations

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

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

April 21, 2006

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: APPLICATION OF THE TRACG COMPUTER CODE TO EVALUATE THE STABILITY OF THE ECONOMIC SIMPLIFIED BOILING WATER REACTOR

Dear Mr. Reyes:

During the 531st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2006, we reviewed the staff's draft Safety Evaluation Report related to the use of the TRACG computer code to evaluate the stability of the Economic Simplified Boiling Water Reactor (ESBWR). This issue was reviewed by our Thermal-Hydraulic Phenomena Subcommittee on January 19 and March 14, 2006. During our reviews, we had the benefit of presentations by and discussions with representatives of the NRC staff and General Electric (GE). We also had the benefit of the documents referenced.

RECOMMENDATIONS

The staff should approve the use of TRACG to analyze the stability of the ESBWR during normal operation, anticipated operational occurrences, and the low-power phase of reactor startup.

DISCUSSION

TRACG has been validated for the analysis of anticipated operation occurrences in boiling water reactors (BWRs) and for loss-of-coolant accident analyses of the ESBWR. It has also been used as a basic computational tool for predicting the performance of BWRs in commercial service. It is currently under review for use in addressing stability-related issues for operating BWRs.

The question we addressed was whether TRACG can adequately model those ESBWR features that affect stability. The main difference between the ESBWR and current operating BWRs is the use of natural circulation, rather than forced circulation, to provide flow to the core during full-power operation. This leads to a number of design changes, including the use of a subdivided "chimney" section above the core.

Our evaluation was limited to the capabilities of TRACG to represent the major physical phenomena and was not a detailed assessment of the performance of an ESBWR.

Our review focused on several questions:

- How well does TRACG model the phenomena that have an important influence on ESBWR stability?

- Do the data from operating reactors and the test facilities accurately represent the phenomena that will exist in the ESBWR?
- Does TRACG adequately model two-phase flow in the chimney?
- Are the nodalization of the chimney and the associated computational scheme adequate to represent unsteady flow in the chimney?
- Does TRACG adequately model natural circulation oscillations?
- Are the predictions of pressure drop fluctuations in the core, the chimney, and other parts of the natural circulation loop reasonable?
- Is the interaction between criticality conditions and the void fraction, flow rate, and heat transfer fluctuations reasonably represented?
- Are the predicted transient responses and decay ratios credible?

In response, GE and the staff presented detailed calculations. There are several sources of data from operating BWRs that have experienced oscillatory behavior. Limited experimental data relevant to the ESBWR are also available. These data include void fraction measurements by Ontario Hydro in large-diameter pipes and transient tests at SIRIUS/CRIEPI which were specifically designed to model some features of the ESBWR.

GE presented several comparisons between TRACG predictions and data recorded at operating BWRs (Peach Bottom, La Salle, Leibstadt, and Dodewaard). These comparisons included scenarios during which the plants were operating at or close to natural circulation conditions. The comparisons indicated that the code has the ability to model the phenomena that are relevant to the ESBWR, and that it represents these oscillations with reasonable accuracy.

Based on comparisons with the Ontario Hydro tests, TRACG appears to provide a reasonable representation of the average void fraction in a large duct, such as the ESBWR chimney, as a function of flow rate and steam quality. At a meeting with our Thermal-Hydraulic Phenomena Subcommittee, GE also presented predictions for the ESBWR response to random void fraction fluctuations that were observed in some tests.

GE explored various nodalizations of the chimney. GE demonstrated that the computational scheme can describe void propagation without significant distortion, numerical diffusion, or artificial mixing. However, they presented other results which indicated that there could be significant numerical diffusion, leading to artificial attenuation of void waves, if the Courant number was not close to 1. GE was able to argue that the effects of this distortion were not significant for the particular case of the ESBWR response that they presented. However, GE and the staff will need to evaluate these effects carefully when more complete analyses are performed in support of the ESBWR design certification.

GE showed that TRACG modeled the main features of low-pressure (startup) oscillations observed in the CRIEPI/SIRIUS tests. These results were consistent with qualitative descriptions of the governing physical processes. High-pressure oscillations in CRIEPI/SIRIUS were also successfully modeled by TRACG. Natural circulation instability in the FRIGG tests, which used electrical heating and lacked the damping introduced by neutronic feedback in the ESBWR, was also successfully modeled by TRACG.

TRACG simulations of ESBWR transients displayed the usual density-wave oscillations that are familiar from BWR experience, but did not reveal significant natural circulation oscillations. GE and the staff provided detailed calculations and physical arguments to explain the absence of these oscillations to our satisfaction. This included presentation of the interaction between components of pressure drop and buoyancy fluctuations in components of the system. They also explained why criticality feedback tended to induce density-wave oscillations but suppress natural circulation oscillations.

The staff performed several useful confirmatory analyses. These included runs of the LAPUR and RELAP5 codes, and the use of a drift-flux void propagation model. In addition, the staff performed several sensitivity studies using TRACG to confirm the code's robustness. They also confirmed that the use of a low Courant number could lead to numerical diffusion.

On the basis of these detailed explanations we found the predicted transient responses to be credible and concluded that TRACG was able to model them adequately. We expect to consider them further during our review of the ESBWR design certification application.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from David B. Matthews, Director, Division of New Reactor Licensing, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "Draft Safety Evaluation for the Application of TRACG for ESBWR Stability," January 12, 2006.
2. Memorandum from Frank M. Akstulewicz to Laura A. Dudes, Safety Evaluation by the Office of Nuclear Reactor Regulation, "Application of the TRACG Computer Code to Stability Analysis for the ESBWR Design — NEDE-33083P, Supplement 1," March 28, 2006.
3. NEDE-33083P, Supplement 1, "TRACG Application for ESBWR Stability Analysis," General Electric Nuclear Energy, December 2004.
4. NEDE-32176P, Rev. 2, "TRACG Model Description," December 1999.
5. NEDE-33083P-A, "TRACG Application for ESBWR," March 2005.

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

May 31, 2006

MEMORANDUM TO: Mario Bonaca for Dana Powers, Chair
Early Site Permit Subcommittee

FROM: David C. Fischer, Senior Staff Engineer

David C Fischer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON THE GRAND GULF EARLY SITE PERMIT APPLICATION: EVALUATION OF TRANSPORTATION ACCIDENTS ON THE MISSISSIPPI RIVER

Attached is a copy of the EDO's May 18, 2006, letter of response to the ACRS's April 14, 2006, letter on the Committee's review of the evaluation of transportation accidents on the Mississippi River related to the early site permit (ESP) application for the Grand Gulf site and the associated changes to the NRC staff's final Safety Evaluation Report (FSER). A copy of the Committee's letter is also attached.

Committee Letter

In its letter, the Committee found the staff's analyses of river transportation accidents on the Mississippi River near the proposed Grand Gulf early site permit site to be acceptable and supported the staff's proposed changes to the Safety Evaluation Report to describe these analyses.

EDO Response

In the EDO's response letter, the staff stated that it "appreciates the insights that the ACRS provided concerning the safety sections for the Grand Gulf ESP. These insights made a valuable contribution to our review and development of the FSER."

Analysis

The EDO's response is satisfactory. The Committee could use this as an example of how ACRS review of ESP applications adds value to the staff's review to help ensure the protection of public health and safety.

cc: ACRS Members
JTLarkins
SDuraiswamy
MSnodderly
EThornsbury

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

May 18, 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: GRAND GULF EARLY SITE PERMIT APPLICATION: EVALUATION OF TRANSPORTATION ACCIDENTS ON THE MISSISSIPPI RIVER

Dear Chairman Wallis:

Thank you for your supplemental letter dated April 14, 2006, regarding the final safety evaluation report (FSER) of the System Energy Resources, Inc. application for the Grand Gulf early site permit (ESP).

In a letter to the staff dated December 23, 2005, the Advisory Committee on Reactor Safeguards (ACRS) communicated its concern regarding the staff's analyses on hazards posed to the proposed site by transportation accidents on the Mississippi River. The staff addressed this concern in a memorandum dated March 27, 2006, and later presented its conclusions at the 531st meeting of the ACRS on April 6, 2006. In your letter following the 531st meeting of the ACRS, you stated that both the staff's analyses of river transportation accidents and the proposed changes to the staff's safety evaluation report are acceptable.

The U.S. Nuclear Regulatory Commission staff appreciates the insights that the ACRS provided concerning the safety sections for the Grand Gulf ESP. These insights made a valuable contribution to our review and development of the FSER.

Sincerely,

William F. Kane /for

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

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

April 14, 2006

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: GRAND GULF EARLY SITE PERMIT APPLICATION: EVALUATION OF TRANSPORTATION ACCIDENTS ON THE MISSISSIPPI RIVER

Dear Mr. Reyes:

During the 531st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2006, we met with representatives of the NRC staff and System Energy Resources, Inc. (SERI), the applicant for an early site permit (ESP) for the Grand Gulf site, and discussed the evaluations performed by the applicant and the NRC staff of the hazards posed to the proposed site by transportation accidents on the Mississippi River as well as the proposed changes to the NRC staff's final Safety Evaluation Report. We provided an interim letter on the Grand Gulf ESP application and the draft Safety Evaluation Report on June 14, 2005, and a final letter on December 23, 2005. The Committee reviewed this application to fulfill the requirement of 10 CFR 52.23 that the ACRS report on those portions of an ESP application that concern safety. We also had the benefit of the documents referenced.

In our December 23, 2005, letter concerning the staff's final Safety Evaluation Report for the Grand Gulf early site permit application, we asked for clarification on risks associated with transportation accidents and possible explosions on the Mississippi River, which is approximately 1.8 kilometers from the proposed site. We asked particularly for a more complete explanation of the attenuation of shock waves that was attributed to the location and elevation of the site relative to the river. The staff asked the applicant to provide this clarification.

In response, the applicant adopted an alternative approach to the analysis of accidental explosions during transportation accidents on the river. This approach is centered on the low probability of an explosion that could produce a pressure pulse that exceeded about 7 kPa at the proposed site. To do this, the applicant examined three types of explosions that might occur should there be an accident involving barge traffic on the river:

- explosions contained within a barge
- explosions near a barge that had spilled volatile, combustible cargo so that a vapor cloud developed
- explosions of vapor clouds that drifted toward the proposed site

The staff independently evaluated the probabilities of these three classes of explosions. The staff was careful to use shipment frequencies, accident frequencies, spill frequencies, and the like that could be justified based on data applicable to barge traffic on the Mississippi River. The staff adopted conservative probabilities in those instances where sufficient data were not available to justify lower probabilities used in some cases by the applicant. Nevertheless, the staff concluded that the probability of an explosion producing a pressure pulse in excess of 7 kPa at the proposed power plant site was on the order of 10^{-6}/yr. The staff concluded that explosions of such low probability posed negligible risk to power plant facilities that might be located on the proposed site.

We found the staff's analyses of river transportation accidents acceptable and support the staff's proposed changes to the Safety Evaluation Report to describe these analyses.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum dated March 27, 2006, from David A. Matthews, NRR/ADRA/DNRL to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: ACRS Review of the Grand Gulf Early Site Permit Application - Final Safety Evaluation Report Changed Pages.
2. U.S. Nuclear Regulatory Commission, Final Safety Evaluation Report, "Safety Evaluation of Early Site Permit Application in the Matter of System Energy Resources, Inc., a Subsidiary of Entergy Corporation, for the Grand Gulf Early Site Permit Site," October 21, 2005.
3. Letter dated June 14, 2005, from G.B. Wallis, Chairman, ACRS, to L.A. Reyes, Executive Director for Operations, NRC, Subject: Interim Letter: Draft Safety Evaluation Report on Grand Gulf Early Site Permit Application.
4. Letter dated December 23, 2005, from Graham B. Wallis, ACRS, to L.A. Reyes, Executive Director for Operations, NRC, Subject: Early Site Permit Application for the Grand Gulf Site and the Associated Final Safety Evaluation Report.
5. Letter dated February 1, 2006, from L.A. Reyes, Executive Director for Operations, NRC, to Graham B. Wallis, ACRS, Subject: Early Site Permit Application for the Grand Gulf Site and the Associated Final Safety Evaluation Report.
6. System Energy Resources, Inc. (SERI), letter dated February 22, 2006, from George A. Zinke, SERI, to NRC Document Control Desk, Subject: Response to Request for Additional Information Regarding the Grand Gulf Early Site Permit Final Safety Evaluation Report.
7. System Energy Resources, Inc. (SERI), letter dated March 7, 2006, from George A. Zinke, SERI, to NRC Document Control Desk, Subject: Supplemental Information, Response to Request for Additional Information Regarding the Grand Gulf Early Site Permit Final Safety Evaluation Report.
8. Regulatory Guide 1.91, Revision 1, "Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants," dated February 1978.

Proposed Schedule for
Quality Assessment of Selected NRC Research Projects

1	Status Report By the Panels	June ACRS Meeting
2	Transmittal of Panel's Findings to Dana Powers and Hossein Nourbakhsh	August 21, 2006
3	Committee Discussion of Draft Report	September ACRS Meeting
4	Discussion/Approval of Final Report to RES Director	October ACRS Meeting

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