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

March 14, 2006

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

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 529th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, February 9-10, 2006, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 529th meeting, February 9-10, 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 NUREG Report, "Evaluation of Human Reliability Analysis Methods Against Good Practices," dated February 22, 2006.
- Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," dated February 22, 2006.

MEMORANDA:

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

- Draft Final Revision 2 to Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," dated February 14, 2006.
- Proposed Revisions to Regulatory Guides Regarding ASME Code Cases, dated February 14, 2006.
- Anonymous Letter Concerning the TRACE Computer Code Development and Review Practices, dated February 15, 2006.

OTHER:

- Letter to Mr. Paul B. Blanch from ACRS Chairman, Graham B. Wallis, dated February 10, 2006, Subject: Questions About The Role of the ACRS.

HIGHLIGHTS OF KEY ISSUES

1. Evaluation of Human Reliability Analysis (HRA) Methods Against Good Practices

The Committee met with representatives of the NRC staff to discuss the draft NUREG report on the evaluation of HRA methods against the good practices specified in NUREG-1792. The Committee reviewed the good practices in May 2004, while a joint meeting of the Human Factors and Reliability & Probabilistic Risk Assessment subcommittees reviewed the draft NUREG on the evaluation of methods in December 2005. The purpose of this report is to aid reviewers of HRAs in assessing the quality of analyses submitted to the NRC. It also provides the technical basis for developing review questions. Since this report highlights the strengths, limitations, and bases of various commonly applied HRA methods, it should also be useful to analysts preparing HRAs and other submittals requiring human performance considerations.

Committee Action:

The Committee issued a report to the NRC EDO, dated February 22, 2006, recommending that the draft report be issued for public comment.

2. Proposed Revisions to SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs"

The Committee met with representatives of the NRC staff to discuss the Standard Review Plan (SRP) 14.2.1, "Generic Guidelines for Extended Power Uprate [EPU] Testing Programs." The staff provided recent changes to the SRP, staff evaluations using the SRP, and a discussion of SRP Paragraph III.C, "Justification for Elimination of EPU Power Ascension Tests." Most of the SRP changes were editorial. The Committee focused on Paragraph III.C. The Committee said Paragraph III.C properly identifies the factors that would support a decision to eliminate EPU power ascension tests, but Paragraph III.C does not provide explicit guidance on how the decision should be made.

Committee Action:

The Committee issued a report to the NRC EDO, dated February 22, 2006, recommending that Paragraph III.C of SRP Section 14.2.1 be rewritten to provide more structured and explicit guidance defining those conditions under which large transient tests would be exempted or required.

3. FERRET Reactor Vessel Fluence Methodology

The Committee met with representatives of NRR to discuss the FERRET least squares adjustment methodology for reactor vessel dosimetry. The staff's presentation described the applicable General Design Criteria, the discrepancy between calculated and measured fluence values, and the history associated with the FERRET methodology. The general design criteria state that the reactor coolant pressure boundary should behave in a non-brittle manner and fluence is a major source of embrittlement in these materials. Fluence is also needed to calculate pressure-temperature limits for reactor pressure vessels. The FERRET methodology combines measured dosimetry foil activations and calculated neutron spectrums to determine best estimate fluence. The staff requested that this methodology be submitted for review after reviewing vessel dosimetry reports that showed large discrepancies in the ratios of calculated to measured values. In 2004 Westinghouse submitted a topical report regarding the FERRET methodology for the staff's review. This report was later revised based on staff comments. The revised report includes a database of 104 surveillance capsules with uncertainties of about 10%. The staff approved the FERRET methodology under the condition that the uncertainties are within the bounds of the database.

Committee Action

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

4. Draft ACRS Report on the NRC Safety Research Program

The ACRS provides the Commission a biennial report, presenting the Committee's observations and recommendations concerning the overall NRC Safety Research Program. During the February meeting, the Committee discussed its draft 2006 report to the Commission on the NRC Safety Research Program.

Committee Action

The Committee plans to continue its discussion of the draft report on the NRC safety research program during its March 2006 meeting.

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 February 8, 2006 meeting with the NRC staff and representatives of Progress energy Carolinas, Inc. (PEC) to review the draft safety evaluation report (SER) related to the license renewal application for the Brunswick Steam Electric Plant, Units 1 and 2. The current operating licenses for Units 1 and 2 expire on September 8, 2016, and December 27, 2014, respectively. During the meeting, PEC described the plant, its operating history, the license renewal review methodology, and its commitment tracking system. The primary containments are of the BWR Mark I design but are constructed of reinforced concrete with a carbon steel liner. The staff's draft SER was issued on December 20, 2005 and contains no open or confirmatory items.

Subcommittee Report on NRC Safety Culture Initiative

The Chairman of the joint Subcommittees on Human Factors and Reliability and Probability Risk Assessment provided a report to the Committee summarizing the results of the January 25, 2006 Subcommittee meeting with the NRC staff regarding the status of NRC's safety management/culture initiatives and associated approaches to address safety culture in the regulatory oversight process (ROP). The Subcommittee gathered information in three areas (1) description of safety culture components and how they would be used in a regulatory process, (2) status of NRC safety culture initiative and proposed approach, and (3) international experience related to safety culture. The Subcommittee Chairman proposed that a letter to Commission be written on NRC's safety culture initiative.

Subcommittee Report on Thermal Hydraulic Phenomena

The Chairman of the Thermal Hydraulics Subcommittee provided a report to the Committee summarizing the results of the January 19, 2006 meeting with the NRC staff regarding a revision to Regulatory Guide 1.82 to reflect lessons learned from the Vermont Yankee Power Uprate review. The revised Regulatory Guide should be available for ACRS consideration in mid-2006. The Subcommittee Chairman also reported that the staff safety evaluation related to the ESBWR stability analysis methodology was considered and an additional meeting with GE and the staff will be needed in March to resolve outstanding issues.

Subcommittee Report on Regulatory Policies and Practices and Thermal Hydraulic Phenomena

The Chairman of the joint Subcommittees on Regulatory Polices and Practices and Thermal Hydraulic Phenomena provided a report to the Committee summarizing the results of the January 25, 2006 meeting to discuss a preliminary version of the draft proposed regulatory guide in support of a voluntary alternative rule that would allow licensees to implement a redefined large break LOCA and associated risk-informed ECCS requirements.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of December 21, 2005, to comments and recommendations included in the ACRS' November 18, 2005 report on the safety aspects of the license renewal application for the Point Beach Nuclear Plant (PBNP) Units 1 and 2. The Committee decided that it was satisfied with the EDO's response.

  The EDO response stated that Region III staff will perform at least two biennial Problem Identification and Resolution (PI&R) inspections at PBNP before Unit 1 enters the period of extended operation and additional PI&R inspections before Unit 2 enters the period of extended operation. Region III staff will also spend at least 100 hours of inspection on special reviews of the licensee's Corrective Action Program after the original red findings have been closed out.

- The Committee considered the EDO's response of December 23, 2005, to comments and recommendations included in the ACRS' November 18, 2005 report on the staff recommendation to withdraw the proposed rule on post-fire operator manual actions. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of December 23, 2005, to comments and recommendations included in the ACRS' November 18, 2005 report on the draft final Generic Letter 2005-xx, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power." The Committee decided that it was satisfied with the EDO's response.

  The EDO's response stated the staff will consider exploring the grid reliability issues stated in the generic letter with the licensees after the electric reliability standards are approved and in effect, the staff will continue to work with FERC and NERC on grid reliability matters as suggested in your letter to ensure a

reliable offsite power system for the nuclear power plants, and we will brief the ACRS after the staff has evaluated the information submitted by the licensees in response to the subject generic letter.

- The Committee considered the EDO's response of December 23, 2005, to comments and recommendations included in the ACRS' November 21, 2005 report on the Committee's review of the Draft NRC Digital System Research Plan for FY 2005 - FY 2009. The Committee decided that it was satisfied with the EDO's response.

  In the EDO's response letter, the staff agrees with all of the Committee's recommendations. The staff plans to expand the research project in Section 3.3.1 of the plan to include development of an inventory and classification system as recommended. The staff plans to better identify regulatory needs and anticipated benefits across all research areas. The staff believes the research gives equal weight to the two aspects of software safety, and plans to ensure that the system-centric approach is more apparent in the plan. Finally, the staff plans to conduct research related to advanced nuclear power plant digital systems with a high priority once the design information becomes available.

- The Committee considered the EDO's response of January 19, 2006, to comments and recommendations included in the ACRS' December 21, 2005 report on the safety aspects of the draft final Generic Letter 2005-xx, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of February 1, 2006 to the ACRS' December 23, 2005 letter on the final Safety Evaluation Report of the System Energy Resources, Inc., application for the Grand Gulf early site permit. The Committee decided that it was satisfied with the EDO's response.

  The EDO response noted the Committee's concern with transportation accidents on the Mississippi River and has asked the applicant to provide additional information to demonstrate how it meets Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants." The NRC staff's evaluation of this information will be documented in an upcoming NUREG. Prior to issuance of the NUREG, the staff plans to inform the ACRS of the proposed changes. The Committee plans to review the staff's evaluation of this information.

- The Committee considered RES' response of December 7, 2005, to the findings included in the ACRS' November 4, 2005 letter on the ACRS' assessment of the quality of selected research projects. The Committee decided that it was satisfied with RES' response.

  The RES response stated that staff intends to re-examine the data and the data reduction from the Rod Bundle Heat Transfer tests at the Pennsylvania State University (PSU) before they are used for model and correlation development. The RES response stated that questionable assumptions involving the treatment

of fluid properties, flow patterns, and magnitude of the bundle pressure drop will be revised if those assumptions made by PSU are found to be inadequate. The RES response also stated that the grid effect on low void and low flow rates will receive additional consideration in future evaluations of these data. RES will soon propose a list of candidate projects for ACRS review in FY 2006.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from December 8, 2005 through February 8, 2006, the following Subcommittee meetings were held:

- Reliability and Probabilistic Risk Assessment and Human Factors — December 15-16, 2005

The joint Subcommittees examined the status of human reliability analysis including ATHEANA, SPAR-H, and industry approaches.

- Thermal-Hydraulic Phenomena — January 19, 2006

The Subcommittee reviewed the analytical methods to be used to evaluate stability scenarios for the ESBWR and discussed the staff's plans to revise Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."

- Human Factors and Reliability and Probability Risk Assessment — January 25, 2006

The Subcommittees examined the status of NRC's safety management/culture initiatives, and associated approaches to address safety culture in the regulatory oversight process.

- Regulatory Policies and Practices and Thermal-Hydraulic Phenomena — January 25, 2006

The Subcommittees reviewed the staff's draft proposed Regulatory Guide in support of risk-informed changes to loss-of-coolant accident technical requirements.

- Planning and Procedures — January 26-27, 2006

The Subcommittee discussed ACRS business processes, anticipated workload, future technical expertise needed on the Committee, strategy for handling anticipated heavy workload, proactive initiatives, knowledge management, ACRS subcommittee structure, stakeholders' comments received during the ACRS self-assessment survey, technical challenges in the areas of advanced reactor designs, early site permits, extended power uprates, and risk-informing 10 CFR Part 50.

- Planning and Procedures — February 8, 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.

- Plant License Renewal — February 8, 2006

The Subcommittee reviewed the License Renewal Application for the Brunswick Steam Electric Plant, Units 1 and 2 and the associated Safety Evaluation Report with Open Items.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee plans to review the draft final NUREG report, "Evaluation of Human Reliability Analysis Methods Against Good Practices," during a future meeting.
- The Committee would like to be kept informed of changes to Standard Review Plan Section III.c.
- The Committee would like to be kept informed of the disposition of issues related to the development, validation, and verification of the TRACE Code.
- The Committee plans to review the final changes to the ROP manual chapters and inspection procedures to address safety culture and the staff's safety culture initiative during its April 2006 meeting.
- The Committee plans to review the application of the TRACG Code for analyzing the Economic Simplified Boiling Water Reactor stability during its April 2006 meeting.
- The Committee plans to review proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," during a future meeting.
- The Committee plans to review the draft final rule and associated Regulatory Guide in support of a risk-informed alternative to ECCS requirements during a future meeting.
- The Committee plans to review the final Safety Evaluation Report related to the license renewal of the Brunswick Steam Electric Plant, Units 1 and 2 during its May 2006 meeting.

PROPOSED SCHEDULE FOR THE 530th ACRS MEETING

The Committee agreed to consider the following topics during the 530th ACRS meeting, to be held on March 9-11, 2006:

- Final Review of the Clinton Early Site Permit Application
- Staff's Evaluation of the Licensees' Responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors"
- Results of the Chemical Effects Tests Associated with PWR Sump Performance
- Final Review of the License Renewal Application for Browns Ferry Units 1, 2, and 3
- Draft Final Revision 4 (DG-1128) to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"
- Evaluation of Precursor Data to Identify Significant Operating Events
- Draft final ACRS Report on the NRC Safety Research Program

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

CERTIFIED

Date Issued: 03/31/2006
Date Certified: 04/10/2006

TABLE OF CONTENTS
MINUTES OF THE 529th ACRS MEETING

FEBRUARY 9-11, 2006

LETTERS:

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

- Draft NUREG Report, "Evaluation of Human Reliability Analysis Methods Against Good Practices," dated February 22, 2006.
- Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," dated February 22, 2006.

MEMORANDA:

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

- Draft Final Revision 2 to Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," dated February 14, 2006.
- Proposed Revisions to Regulatory Guides Regarding ASME Code Cases, dated February 14, 2006.
- Anonymous Letter Concerning the TRACE Computer Code Development and Review Practices, dated February 15, 2006.

OTHER:

- Letter to Mr. Paul B. Blanch from ACRS Chairman, Graham B. Wallis, dated February 10, 2006, Subject: Questions About The Role of the ACRS.

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

529th ACRS Meeting
February 9-11, 2006

CERTIFIED

MINUTES OF THE 529th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS FEBRUARY 9-11, 2006 ROCKVILLE, MARYLAND

The 529th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on February 9-11, 2006. Notice of this meeting was published in the *Federal Register* on January 25, 2006 (65 FR 4177) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

A transcript of selected portions of the meeting is available in the NRC's Public Document Room at One White Flint North, Room 1F-19, 11555 Rockville Pike, Rockville, Maryland. Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc. 1323 Rhode Island Avenue, NW, Washington, DC 20005. Transcripts are also available at no cost to download from, or review on, the Internet at http://www.nrc.gov/ACRS/ACNW.

ATTENDEES

ACRS Members: ACRS Members: Dr. Graham B. Wallis (Chairman), Dr. William J. Shack (Vice Chairman), Mr. John D. Sieber, (Member-at-Large), Dr. George E. Apostolakis, Dr. Mario V. Bonaca, Dr. Richard S. Denning, Dr. Thomas S. Kress, 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. In addition, Mr. Otto L. Maynard was introduced as the newest member on the Committee.

II. Evaluation of Human Reliability analysis (HRA) Methods Against Good Practices (Open)

[Note: Mr. Eric A. Thornsbury was the Designated Federal Official for this portion of the meeting.]

Dr. George Apostolakis, the cognizant Committee member for this issue, introduced the topic of the meeting. Dr. Apostolakis provided an overview of the subcommittee's previous activities related to human reliability. In addition to the report on the evaluation of human reliability analysis methods, the subcommittee reviewed related research on the SPAR-H model, the use of data for HRA, and the Halden experimental program. Dr. Apostolakis also recapped a meeting he had with the NRC Chairman, and described a renewed interest in HRA which focused on the time for human actions.

Dr. Apostolakis reminded the Members of their previous review of the HRA best practices report on which this evaluation was based. He commended the staff for arranging for outside contractors to review the staff-developed methods in order to obtain a more objective evaluation. Dr. Apostolakis commented that this evaluation is an excellent first step toward resolving the issue of model uncertainty in HRA.

During the introduction, Dr. Denning asked if any benchmark experiments existed that compare HRA models. In response, Dr. Apostolakis described an experiment performed by the European Community's Ispra Laboratory 25 years ago, where the results showed orders of magnitude variation among HRA methods and analysts. Dr. Wallis asked about a comparison of the methods not against each other, but against reality. Dr. Apostolakis explained that such hard experimental evidence will not materialize, since HRA is a "soft science."

Dr. Apostolakis introduced Mr. Jimi Yerokun, Chief, Human Factors and Human Reliability Analysis Section, Office of Nuclear Regulatory Research (RES) to begin the staff presentations.

NRC Staff Presentation

Mr. Yerokun introduced Dr. Erasmia Lois to provide the presentation. Dr. Lois provided a outline of the planned presentation, then explained the reasons for the work. The quality of PRA is an important aspect for regulatory decisionmaking, and HRA is identified as one of the areas that needs to be addressed. She reviewed the steps in the development of guidance for reviewing HRA: (1) the development of the good practices report and (2) the evaluation of methods with respect to those good practices. The draft report on this evaluation is undergoing review by internal staff as well as the ACRS. Dr. Lois asked the Committee for a letter endorsing the release of the draft report for public comment.

Dr. Lois described the process used for the evaluations, which included an expert meeting where the results were presented and debated. That expert meeting recommended a deeper look into the underlying technical basis of each model and recommended a discussion of the use of each method as intended versus actual use.

Dr. Lois continued by listing the methods evaluated in the report, which only addressed domestic methods likely to be used in licensee applications. She commented that most of the tools are really just quantification approaches, and not really HRA methods. She ascribed much of the variability seen in HRA to this fact that the use of some of these tools does not necessarily result in a human reliability analysis. An HRA must follow a consistent process and methods such as ATHEANA, THERP, and the EPRI methods do provide this guidance.

Dr. Lois noted that different methods exist because of different needs, such as detailed analysis versus scoping analysis. She described some of the strengths seen in the methods including a strong technical basis and step-by-step guidance on how to use a tool. Weaknesses usually included a weak technical basis in some of the methods.

Following a discussion by the staff and members on many of the issues described below, Dr. Lois turned the presentation over to Dr. Forester, Sandia National Labs. and Dr. Kolaczkowski, Science Applications Internal Corp. Dr. Apostolakis suggested that Dr. Forester proceed and describe the results of the evaluation of only a few of the methods. Dr. Forester then described THERP, the first HRA method and the most used method. It provides guidance for identifying human failure events, how to model them, and how to quantify them in PRA. THERP has less emphasis on diagnosis errors but instead performs a detailed task analysis of the human actions.

Dr. Forester then described the evaluation of the EPRI CBDT method, which addresses cases beyond those handled by a previous EPRI time reliability approach. He also described the evaluation of SPAR-H, and addressed questions from the members concerning its adequacy for use in regulatory applications.

Dr. Kolaczkowski completed the presentation by discussing the conclusion and the changes to the draft report due to comments received from the subcommittee and internal staff review. He discussed the overall insights on the different concepts used for quantification. These methods use either a base probability modified by performance shaping factors, or use direct expert elicitation. Mr. Kolaczkowski also discussed how the different methods treat uncertainty. Many of the methods provide standard uncertainty bounds, while others provide qualitative guidance. In conclusion, the research so far does not give a hard and fast rule for when a particular method should be used. The opinion of the staff is that one or more methods that are applicable for the selected application should be identified, rather than starting with a preferred method.

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

- Dr. Apostolakis asked if the ACRS will have an opportunity to review the final version of the report. Dr. Lois committed to submit the revised version back to the Committee.

- Dr. Wallis commented that the evaluation of methods is really a comparison against the good practices, and not an evaluation of how people really behave.

- Dr. Wallis asked about the use of HRA methods by other industries. Dr. Apostolakis answered that the nuclear industry is ahead of everybody else, noting in particular that other industries do not attempt to produce probabilities. Mr. Sieber pointed out that some of the data underlying the early models was non-nuclear. Dr. Apostolakis confirmed that using the work of Swain and Guttman as the primary example.

- Dr. Powers asked if EPRI endorses any particular method. Dr. Lois pointed to the EPRI HRA Calculator which employs several models. Dr. Apostolakis pointed out that a natural vetting process appears to be occurring in the use of the calculator, where one method in particular is becoming unused.

- Dr. Apostolakis encouraged more explicit statements with regard to weaker methods. Dr. Lois answered that they are planning to do so.

- Dr. Powers commented that HRA has many models that only solve half of the problem, since we are now asking more detailed and refined questions than the methods were designed to address. He suggested that we need to drive toward something that solves the whole problem to the level of comprehension that can now ask the questions. He noted that the good practices document is a good first step in that process.

- Mr. Sieber commented that many of the methods really depend on the skill of the analyst which he considers a strong weakness. Dr. Kolaczkowski agreed, and described the two conflicting issues in HRA. On one hand, structure would provide repeatability, but flexibility is desirable to allow analysts to better address special situations. The work is driving toward resolution of this conflict.

- Dr. Powers commented that it is unclear how good HRA needs to be. Dr. Kolaczkowski answered that we know that it depends on the application. Dr. Wallis added that it would help to talk about the problem being addressed. Dr. Shack added that importance measures might provide insight to how good it must be. Dr. Forester agreed that a screening analysis is helpful in determining the level of detail needed in an analysis.

- Dr. Denning noted that PRA is used quantitatively, with little consideration of uncertainty. His concern is that agreement on a method will narrow the perception of uncertainty without actually reducing it. He suggested that we look at the uncertainty as well as the probability and force the regulatory process to consider those uncertainties.

- Dr. Wallis asked for help understanding HRA numbers. Dr. Forester explained that the apparent precision of some numbers is often an artifact of calculations used in the methods. Dr. Apostolakis described how high failure probabilities can depend on the context of the actions. He also noted that the work is helping us get there, but is not yet answering the questions yet. Dr. Kolaczkowski added that a better understanding of the methods begins to help us grade the results.

Committee Action:

The Committee issued a letter to the NRC Executive Director for Operations (EDO), dated February 22, 2006, recommending that the staff issue the draft report for public comment.

III. Proposed Revisions to SRP Section 14.2.1, "Generic guidelines for Extended Power Uprate Testing Programs" (Open)

[Note: Mr. John G. Lamb was the Designated Federal Official for this portion of the meeting.]

The Chairman of the Power Uprate Subcommittee provided an introduction to the staff. The Committee had the benefit of presentations and discussions with representatives of the staff regarding Standard Review Plan (SRP) Section14.2.1, "Generic Guidelines for Extended Power Uprate [EPU] Testing Programs."

The staff provided recent changes to the SRP, staff evaluations using the SRP, and a discussion of SRP Paragraph III.C, "Justification for Elimination of EPU Power Ascension Tests." The staff stated that most changes to the SRP were editorial. The staff said the most significant change was the addition of a paragraph to clarify Paragraph III.C.c., "Facility Conformance to Limitations Associated with Computer Modeling and Analytical Methods." The staff stated the new paragraph is to ensure setpoint and parameter changes, and modifications do not invalidate analytical methods. The staff said if analytical methods are inadequate, the secondary review branches may need transient testing performed to make its final safety conclusion. The staff stated that some of the factors that the staff considers in Paragraph III.C are operating experience, thermal-hydraulic phenomena or system interactions, computer modeling, and plant operations and use of procedures. The Committee commented that Paragraph III.C properly identifies the factors that would support a decision to eliminate EPU power ascension tests, but Paragraph III.C does not provide explicit guidance on how the decision should be made.

Committee Action:

The Committee issued a report to the NRC EDO, dated February 22, 2006, recommending that 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.

IV. FERRET Reactor Vessel Fluence Methodology (Open)

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

The Committee met with representatives of the Office of Nuclear Reactor Regulation (NRR) to discuss the FERRET least squares adjustment methodology for reactor vessel dosimetry. The staff's presentation described the applicable general design criteria, the discrepancy between calculated and measured fluence values, and the history associated with the FERRET methodology. The general design criteria state that the reactor coolant pressure boundary should behave in a non-brittle manner and fluence is a major source of embrittlement in these materials. Fluence is also used in assessments of irradiation assisted stress corrosion cracking, material weldability, and pressure-temperature limits. The FERRET methodology combines measured dosimetry foil activations and calculated neutron spectrums to determine best estimate fluence. The staff requested that this methodology be submitted for review after reviewing vessel dosimetry reports that showed large discrepancies in ratios of calculated-to-measured values. In 2004 Westinghouse submitted a topical report regarding the FERRET methodology for the staff's review. This report was later revised based on staff comments. The revised report includes a database of 104 surveillance capsules with uncertainties of about 10%. The staff approved the FERRET methodology under the condition that the uncertainties were within the bounds of the database.

Committee Action

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

V. Draft ACRS Report on the NRC Safety Research Program (Open)

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

The ACRS provides the Commission a biennial report, presenting the Committee's observations and recommendations concerning the overall NRC Safety Research Program. During the February meeting, the Committee discussed its draft 2006 report to the Commission on the NRC Safety Research Program.

Committee Action

The Committee plans to continue its discussion of the draft report on the NRC safety research program during its March 2006 meeting.

VI. Subcommittee Reports

a. Subcommittee on Plant License Renewal

The Chairman of the Plant License Renewal Subcommittee provided a report to the Committee summarizing the results of the February 8, 2006 meeting with the NRC staff and representatives of Progress energy Carolinas, Inc. (PEC) to review the draft safety evaluation report (SER) related to the license renewal application for the Brunswick Steam Electric Plant, Units 1 and 2. The current operating licenses for Units 1 and 2 expire on September 8, 2016, and December 27, 2014, respectively. During the meeting, PEC described the plant, its operating history, the license renewal review methodology, and its commitment tracking system. The primary containments are of the BWR Mark I design but are constructed of reinforced concrete with a carbon steel liner. The staff's draft safety evaluation report was issued on December 20, 2005 and contains no open or confirmatory items.

b. Subcommittee Report on NRC Safety Culture Initiative

The Chairman of the joint Subcommittees on Human Factors and Reliability and Probability Risk Assessment provided a report to the Committee summarizing the results of the January 25, 2006 Subcommittee meeting with the NRC staff regarding the status of NRC's safety management/culture initiatives and associated approaches to address safety culture in the regulatory oversight process (ROP). The Subcommittee gathered information in three areas (1) description of safety culture components and how they would be used in a regulatory process, (2) status of NRC safety culture initiative and proposed approach, and (3) international experience related to safety culture. The Subcommittee Chairman proposed that a letter to Commission be written on NRC's safety culture initiative.

c. Subcommittee Report on Thermal Hydraulic Phenomena

The Chairman of the Thermal Hydraulics Subcommittee provided a report to the Committee summarizing the results of the January 19, 2006 meeting with the NRC staff regarding a revision to Regulatory Guide 1.82 to reflect lessons learned from the Vermont Yankee Power Uprate review. The revised Regulatory Guide should be available for ACRS consideration in mid-2006. The Subcommittee Chairman also reported that the staff safety evaluation related to the ESBWR stability analysis methodology was considered and an additional meeting with GE and the staff will be needed in March to resolve outstanding issues.

d. Subcommittee Report on Regulatory Policies and Practices and Thermal Hydraulic Phenomena

The Chairman of the joint Subcommittees on Regulatory Polices and Practices and Thermal Hydraulic Phenomena provided a report to the Committee summarizing the results of the January 25, 2006 meeting to discuss a preliminary version of the draft proposed regulatory guide in support of a voluntary alternative rule that would allow licensees to implement a redefined large break LOCA and associated risk-informed ECCS requirements.

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

- The Committee considered the EDO's response of December 21, 2005, to comments and recommendations included in the ACRS' November 18, 2005 report on the safety aspects of the license renewal application for the Point Beach Nuclear Plant (PBNP) Units 1 and 2. The Committee decided that it was satisfied with the EDO's response.

The EDO response stated that Region III staff will perform at least two biennial Problem Identification and Resolution (PI&R) inspections at PBNP before Unit 1 enters the period of extended operation and additional PI&R inspections before Unit 2 enters the period of extended operation. Region III staff will also spend at least 100 hours of inspection on special reviews of the licensee's Corrective Action Program after the original red findings have been closed out.

- The Committee considered the EDO's response of December 23, 2005, to comments and recommendations included in the ACRS' November 18, 2005 report on the staff recommendation to withdraw the proposed rule on post-fire operator manual actions. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of December 23, 2005, to comments and recommendations included in the ACRS' November 18, 2005 report on the draft final Generic Letter 2005-xx, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power." The Committee decided that it was satisfied with the EDO's response.

  The EDO's response stated the staff will consider exploring the grid reliability issues stated in the generic letter with the licensees after the electric reliability standards are approved and in effect, the staff will continue to work with FERC and NERC on grid reliability matters as suggested in your letter to ensure a reliable offsite power system for the nuclear power plants, and we will brief the ACRS after the staff has evaluated the information submitted by the licensees in response to the subject generic letter.

- The Committee considered the EDO's response of December 23, 2005, to comments and recommendations included in the ACRS' November 21, 2005 report on the Committee's review of the Draft NRC Digital System Research Plan for FY 2005 - FY 2009. The Committee decided that it was satisfied with the EDO's response.

  In the EDO's response letter, the staff agrees with all of the Committee's recommendations. The staff plans to expand the research project in Section 3.3.1 of the plan to include development of an inventory and classification system as recommended. The staff plans to better identify regulatory needs and anticipated benefits across all research areas. The staff believes the research gives equal weight to the two aspects of software safety, and plans to ensure that the system-centric approach is more apparent in the plan. Finally, the staff plans to conduct research related to advanced nuclear power plant digital systems with a high priority once the design information becomes available.

- The Committee considered the EDO's response of January 19, 2006, to comments and recommendations included in the ACRS' December 21, 2005 report on the safety aspects of the draft final Generic Letter 2005-xx, "Impact of Potentially Degraded

Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of February 1, 2006 to the ACRS' December 23, 2005 letter on the final Safety Evaluation Report of the System Energy Resources, Inc., application for the Grand Gulf early site permit. The Committee decided that it was satisfied with the EDO's response.

  The EDO response noted the Committee's concern with transportation accidents on the Mississippi River and has asked the applicant to provide additional information to demonstrate how it meets Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants." The NRC staff's evaluation of this information will be documented in an upcoming NUREG. Prior to issuance of the NUREG, the staff plans to inform the ACRS of the proposed changes. The Committee plans to review the staff's evaluation of this information.

- The Committee considered RES' response of December 7, 2005, to the findings included in the ACRS' November 4, 2005 letter on the ACRS' assessment of the quality of selected research projects. The Committee decided that it was satisfied with RES' response.

  The RES response stated that staff intends to re-examine the data and the data reduction from the Rod Bundle Heat Transfer tests at the Pennsylvania State University (PSU) before they are used for model and correlation development. The RES response stated that questionable assumptions involving the treatment of fluid properties, flow patterns, and magnitude of the bundle pressure drop will be revised if those assumptions made by PSU are found to be inadequate. The RES response also stated that the grid effect on low void and low flow rates will receive additional consideration in future evaluations of these data. RES will soon propose a list of candidate projects for ACRS review in FY 2006.

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

- The Committee plans to review the draft final NUREG report, "Evaluation of Human Reliability Analysis Methods Against Good Practices," during a future meeting.
- The Committee would like to be kept informed of changes to Standard Review Plan Section III.c.
- The Committee would like to be kept informed of the disposition of issues related to the development, validation, and verification of the TRACE Code.
- The Committee plans to review the final changes to the ROP manual chapters and inspection procedures to address safety culture and the staff's safety culture initiative during its April 2006 meeting.
- The Committee plans to review the application of the TRACG Code for analyzing the Economic Simplified Boiling Water Reactor stability during its April 2006 meeting.

- The Committee plans to review proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," during a future meeting.
- The Committee plans to review the draft final rule and associated Regulatory Guide in support of a risk-informed alternative to ECCS requirements during a future meeting.
- The Committee plans to review the final Safety Evaluation Report related to the license renewal of the Brunswick Steam Electric Plant, Units 1 and 2 during its May 2006 meeting.

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 February 8, 2006. The following items were discussed:

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through April 2006 was discussed. 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.

Response to the Staff Requirements Memorandum (SRM)

In a December 20, 2005 SRM resulting from the ACRS meeting with the NRC Commissioners on December 8, 2005, the Commission requested the following:

a) Following its retreat in January 2006, the ACRS should inform the Commission how the Committee plans to manage the increased workload resulting from the anticipated receipt of new reactor designs and combined license (COL) applications.

b) The ACRS shall make among its highest priorities its role in the resolution of GSI-191 [The staff shall expedite efforts to provide the ACRS with information necessary to make its assessments and recommendations]

Regarding Item (a), during its January 26-27, 2006 Planning and Procedures Subcommittee meeting, the members discussed a plan proposed by the ACRS staff for handling anticipated heavy workload in the areas of advanced reactors and COLs. A draft response to the Commission SRM was provided to the Committee for discussion and endorsement. The due date for responding to the Commission SRM is March 31, 2006.

Regarding Item (b), the ACRS Subcommittee on Thermal-Hydraulic Phenomena is scheduled to hold a meeting on February 14-16, 2006 to discuss interim results of the chemical effects tests and industry responses to the Generic Letter on PWR sumps. This matter is scheduled for full Committee discussion during the March 2006 ACRS meeting. The Subcommittee and the full Committee will continue to discuss issues related to PWR sump performance as further progress has been made by the staff.

Letter from Mr. Paul Blanch Regarding Vermont Yankee Extended Power Uprate

Mr. Paul Blanch, Energy Consultant, sent a letter to the ACRS Chairman dated January 20, 2006, documenting his views about the ACRS review of the Vermont Yankee extended power uprate. He expressed concern about whether Vermont Yankee will meet all applicable regulatory requirements at the extended power uprate conditions. He requested that the ACRS provide a statement, supported by objective evidence, that Vermont Yankee will be operated in compliance with applicable regulatory requirements thus assuring public safety. Dr. Denning, Chairman of the ACRS Subcommittee on Power Uprates, and Dr. Wallis prepared a draft response to Mr. Paul Blanch, for consideration by the Committee.

ACNW Meeting on Radiation Protection Program

The ACNW is scheduled to hear presentations by RES regarding the Radiation Protection Program during the April 18-19, 2006, ACNW meeting. ACNW invited interested ACRS members to participate in this session.

Actions, Agreements, Commitments, and Follow-up Items Resulting from the ACRS Retreat

A Planning and Procedures Subcommittee was held on January 26-27, 2006 to discuss various issues. A summary of the actions, agreements, commitments, and follow-up items resulting from this meeting were discussed.

ACRS Conference Room Upgrade

Arrangements are being made to upgrade the ACRS conference room audiovisual equipment. The upgrade will begin on March 13, 2006, and is expected to be completed on or before April 24, 2006. Arrangements are being made to hold future subcommittee meetings and the April 6-8, 2006 full Committee meeting in different locations.

Interview of Candidates to Fill the Vacancy on the Committee

During the March ACRS meeting, members and the ACRS Member Candidate Screening Panel interviewed best-qualified candidates for membership on the ACRS. A draft Federal Register Notice and Press Release seeking candidates with expertise in various disciplines was sent to the Commission for approval for publication.

Member Issue

Dr. Kress prepared a draft report to the Commission on Risk-informed Criteria for Acceptability of Power Uprates from a Site Suitability Perspective for consideration by the full Committee. Since this item was not announced in the Federal Register notice for the February meeting, the Committee could not discuss this matter at the February meeting. Comments were provided to Dr. Kress.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 530th ACRS Meeting, March 9-11, 2006.

The 529th ACRS meeting was adjourned at 5:40 pm on February 10, 2006.

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

April 10, 2006

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

FROM: Graham B. Wallis
ACRS Chairman

SUBJECT: CERTIFIED MINUTES OF THE 529th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), FEBRUARY 9-11, 2006

I certify that based on my review of the minutes from the 529TH 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.

available at the NRC worldwide Web site *http://www.nrc.gov/public-involve/doc-comment/omb/index.html.* The document will be available on the NRC home page site for 60 days after the signature date of this notice.

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

Dated at Rockville, Maryland, this 19th day of January 2006.

For the Nuclear Regulatory Commission.

Brenda J. Shelton,

NRC Clearance Officer, Office of Information Services.

[FR Doc. E6–887 Filed 1–24–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–255]

Nuclear Management Company, LLC; Notice of Withdrawal of Application for Amendment to Facility Operating License

The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of Nuclear Management Company, LLC (NMC) to withdraw its application of April 1, as supplemented May 26, August 25, and November 22, 2005, for proposed amendment to Facility Operating License No. DPR–20 for the Palisades Nuclear Plant located in VanBuren County, Michigan.

The proposed amendment would have changed Technical Specification (TS) 3.7.8, "Service Water System," to provide a one-time extension to the Completion Time for restoring a service water train to operable status.

The Commission had previously issued a Notice of Consideration of Issuance of Amendment published in the **Federal Register** on May 10, 2005 (70 FR 24654). However, NMC's letter of January 5, 2006, withdrew the proposed change.

For further details with respect to this action, see the application for amendment dated April 1, as supplemented May 26, August 25, and November 22, 2005, and NMC's letter of January 5, 2006, which withdrew the application for license amendment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area 01 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams/html.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209, or 301–415–4737 or by email to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 17th day of January 2006.

For the Nuclear Regulatory Commission.

L. Mark Padovan,

Project Manager, Plant Licensing Branch III–1, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

[FR Doc. E6–886 Filed 1–24–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

The ACRS Subcommittee on Plant License Renewal will hold a meeting on February 8, 2006, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, February 8, 2006—1:30 p.m. until 5 p.m.

The purpose of this meeting is to discuss the License Renewal Application for Brunswick Units 1 and 2 and associated Safety Evaluation Report (SER) related to the License Renewal. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, Carolina Power & Light Company now doing business as Progress Energy Carolinas Incorporated, and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. John G. Lamb (telephone 301/415–6855) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 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 to the agenda.

Dated: January 12, 2006.

Michael L. Scott,

Branch Chief, ACRS/ACNW.

[FR Doc. E6–850 Filed 1–24–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on February 9–11, 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).

Thursday, February 9, 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.–10:30 a.m.: Application of TRACG Code for Analyzing ESBWR Stability (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and General Electric (GE) Nuclear Energy regarding application of the TRACG Code for analyzing Economic Simplified Boiling Water Reactor (ESBWR) stability.

Note: A portion of the session may be closed to discuss the GE Nuclear Energy proprietary information.

10:45 a.m.–12:15 p.m.: Evaluation of Human Reliability Analysis (HRA) Methods Against Good Practices (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft NUREG report on the Evaluation of HRA Methods Against Good Practices specified in NUREG–

1792, "Good Practices for Implementing Human Reliability Analysis."

1:15 p.m.–2:45 p.m.: Proposed Revisions to SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed revisions to the Standard Review Plan (SRP) Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," and related matters.

3 p.m.–5 p.m.: Draft ACRS Report on the NRC Safety Research Program (Open)—The Committee will discuss the draft ACRS report to the Commission on the NRC Safety Research Program.

5:15 p.m.–6:45 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting. Also, the Committee will discuss a draft ACRS response to the Commission request in the December 20, 2005 Staff Requirements Memorandum regarding ACRS plans to manage the anticipated increased workload in the areas of advanced reactor designs and combined license applications.

Friday, February 10, 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.–10 a.m.: FERRET Reactor Vessel Fluence Methodology (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the FERRET methodology which is used to predict the fluence on the reactor vessel wall due to neutron leakage from the core.

10:15 a.m.–11:15 a.m.: Subcommittee Reports (Open)—The Committee will hear reports by and hold discussions with cognizant Chairmen of the ACRS Subcommittees regarding: interim review of the Brunswick Nuclear Plant license renewal application and the associated NRC staff's draft Safety Evaluation Report; safety conscious work environment and safety culture; proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident;" and the draft Regulatory Guide, "An Approach for Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements."

11:15 a.m.–12:15 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

1:15 p.m.–1:30 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.

1:30 p.m.–3:30 p.m.: Draft ACRS Report on the NRC Safety Research Program (Open)—The Committee will discuss the draft ACRS report to the Commission on the NRC Safety Research Program.

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

Saturday, February 11, 2006, Conference Room T-2b3, Two White Flint North, Rockville, Maryland

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

12:30 p.m.–1 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on 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.

In accordance with Subsection 10(d) Public Law 92–463, I have determined that it may be necessary to close a portion of this meeting noted above to discuss and protect information classified as GE Nuclear Energy proprietary information pursuant to 5 U.S.C. 552b(c) (4).

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., e.t., at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: January 19, 2006.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E6–889 Filed 1–24–06; 8:45 am]

BILLING CODE 7590–01–P

APPENDIX II

January 25, 2006

REVISED
SCHEDULE AND OUTLINE FOR DISCUSSION
529th ACRS MEETING
FEBRUARY 9-11, 2006

THURSDAY, FEBRUARY 9, 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. Evaluation of Human Reliability Analysis (HRA) Methods Against Good Practices (Open) (GEA/EAT)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff regarding the draft NUREG report on the Evaluation of HRA Methods Against Good Practices specified in NUREG-1792, "Good Practices for Implementing Human Reliability Analysis."

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

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

3) 10:15 - ~~11:45~~ A.M. **11:20 AM** Proposed Revisions to SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs" (Open) RSD/JGL)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding proposed revisions to the Standard Review Plan (SRP) Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," and related matters.

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

~~11:45~~ - 12:45 P.M. ***LUNCH***
11:20 AM

4) 12:45 - 2:15 P.M. FERRET Reactor Vessel Fluence Methodology (Open) (RSD/CS)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff regarding the FERRET methodology which is used to predict the fluence on the reactor vessel wall due to neutron leakage from the core.

   Representatives of the Westinghouse Electric Corporation may participate, as appropriate.

	2:15 - 2:30 P.M.	*****BREAK*****
5)	2:30 - 5:00 P.M.	Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN/SD) Discussion of the draft ACRS report to the Commission on the NRC Safety Research Program.
	5:00 - 5:15 P.M.	*****BREAK*****
6)	5:15 - ~~6:45~~ P.M. **6:20 PM**	Preparation of ACRS Reports (Open) Discussion of proposed ACRS reports on: 6.1) Evaluation of Human Reliability Analysis Methods Against Good Practices (GEA/EAT) 6.2) Proposed Revisions to SRP Section 14.2.1 (RSD/JGL) 6.3) Response to the Commission SRM dated December 20, 2005 regarding ACRS plans to manage the anticipated increased workload in the areas of advanced reactor designs and combined license applications (TSK/JHF)

FRIDAY, FEBRUARY 10, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7)	8:30 - ~~8:35~~ A.M. **9:50 AM**	Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)
8)	8:35 - 9:30 A.M.	Subcommittee Reports (Open) 8.1) Report by and discussions with the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim review of the Brunswick Nuclear Plant license renewal application and the associated NRC staff's draft Safety Evaluation Report (JDS/JGL). 8.2) Report by and discussions with the Chairman of the ACRS Subcommittee on Human Factors regarding the Safety Conscious Work Environment and Safety Culture (MVB/JHF). 8.3) Report by and discussions with the Chairman of the ACRS Subcommittee on Thermal-Hydraulic Phenomena regarding proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (GBW/RC). 8.4) Report by and discussions with the Chairman of the ACRS Subcommittee on Regulatory Policies and Practices regarding the draft Regulatory Guide, "An Approach for Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" (WJS/MRS).

	~~9:30 - 9:45~~ A.M. 9:50 - 10:15 AM	*****BREAK*****
9)	9:45 - 12:15 P.M. **(11:00-11:15 A.M. BREAK)**	Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN/SD) Discussion of the draft ACRS report to the Commission on the NRC Safety Research Program.
	12:15 - 1:15 P.M.	*****LUNCH*****
10)	1:15 - ~~2:15~~ P.M. **2:50 PM**	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)	2:15 - 2:30 P.M.	Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.) Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.
	2:30 - 2:45 P.M.	*****BREAK*****
12)	2:45 - ~~7:00~~ P.M. **5:40 PM**	Preparation of ACRS Reports (Open) Discussion of proposed ACRS reports on: 12.1) Evaluation of Human Reliability Analysis Methods Against Good Practices (GEA/EAT) 12.2) Proposed Revisions to SRP Section 14.2.1 (RSD/JGL) 12.3) Response to the Commission SRM dated December 20, 2005 regarding ACRS plans to manage the anticipated increased workload in the areas of advanced reactor designs and combined license applications (TSK/JHF)

~~SATURDAY, FEBRUARY 11, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND~~

~~13) 8:30 - 12:30 P.M.~~ ~~Preparation of ACRS Reports (Open)~~
~~(10:30-10:45 A.M. BREAK)~~ ~~Continue discussion of the proposed ACRS reports listed under Item 12, and the draft ACRS report on the NRC Safety Research Program as needed.~~

~~14) 12:30 - 1:00 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

529th ACRS MEETING
FEBRUARY 9-11, 2006

NRC STAFF (February 9, 2006)
T. Herrity, NRR
G. Parry, NRR
D. Marksberry, RES
G. DeMoss, RES
R. Barrett, RES
J. Yerokun, RES
D. Lewis, RES
M. Simmons, RES
R. Jenkins, RES
R. Pettis, Jr., NRR
G. Cranston, NRR
D. Thatcher, NRR
P. Prescott, NRR
M. Chernoff, NRR
C. Ader, RES
G. Galletti, NRR
B. Rogers, NRR
T. Alexion, NRR
L. Lois, NRR
S. Jones, NRR
M. Mitchell, NRR
R. Assa, RES
J. Tobin, OPA

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
J. Forester, Sandia Labs.
A. Kolaczowski, SAIC
D. Vojnorou, SNSA
D. Langley, TVA
R. Simon, TVA
P. S. Lowvorn, TVA
D. Raleigh, LIS, Scientech
K. Feintuch, DORL

February 16, 2006

SCHEDULE AND OUTLINE FOR DISCUSSION
530th ACRS MEETING
MARCH 9-11, 2006

THURSDAY, MARCH 9, 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 - 10:30 A.M. Final Review of the Clinton Early Site Permit Application (Open) (DAP/MRS/DCF)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff and Exelon Generation Company, LLC, regarding the early site permit application for the Clinton site and the associated NRC staff's Final Safety Evaluation Report.

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

3) 10:45 - 11:45 A.M. Staff's Evaluation of the Licensees' Responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors" (Open) (GBW/RC)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's evaluation of the licensees' responses to Generic Letter 2004-02 on PWR sumps.

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

11:45 - 1:00 P.M. ***LUNCH***

4) 1:00 - 3:00 P.M. Results of the Chemical Effects Tests Associated with PWR Sump Performance (Open) (GBW/RC)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the NRC staff and its contractor regarding results of the chemical effects tests related to PWR sump performance.

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

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

5) 3:15 - 5:15 P.M. Final Review of the License Renewal Application for Browns Ferry Units 1, 2, and 3 (Open) (MVB/CS)
- 5.1) Remarks by the Subcommittee Chairman
- 5.2) Briefing by and discussions with representatives of the NRC staff and the Tennessee Valley Authority regarding the license renewal application for Browns Ferry Units 1, 2, and 3 and the associated NRC staff's Final Safety Evaluation Report.

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

6) 5:30 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
- 6.1) Final Review of the Clinton Early Site Permit Application (DAP/MRS/DCF)
- 6.2) Chemical Effects Test Results/Industry Responses to the Generic Letter on PWR Sumps (GBW/RC)
- 6.3) Final Review of the License Renewal Application for Browns Ferry Units 1, 2, and 3 (MVB/CS)

FRIDAY, MARCH 10, 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 - 10:00 A.M. Draft Final Revision 4 (DG-1128) to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (Open) (JDS/JGL)
- 8.1) Remarks by the Subcommittee Chairman
- 8.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final revision 4 to Regulatory Guide 1.97.

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

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

9) 10:15 - 11:45 A.M. Evaluation of Precursor Data to Identify Significant Operating Events (Open) (JDS/JGL)
- 9.1) Remarks by the Subcommittee Chairman
- 9.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's evaluation of precursor data to identify significant operating events.

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

11:45 - 1:00 P.M. ***LUNCH***

10)	1:00 - 2:00 P.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.	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 - 4:30 P.M.	Draft final ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN) Discussion of the draft final ACRS report on the NRC Safety Research Program.
	4:30 - 4:45 P.M.	*****BREAK*****
13)	4:45 - 7:00 P.M.	Preparation of ACRS Reports (Open) Discussion of proposed ACRS reports on: 13.1) Final Review of the Clinton Early Site Permit Application (DAP/MRS/DCF) 13.2) Chemical Effects Test Results/Industry Responses to the Generic Letter on PWR Sumps (GBW/RC) 13.3) Final Review of the License Renewal Application for Browns Ferry Units 1, 2, and 3 (MVB/CS) 13.4) Draft Final Revision 4 (DG-1128) to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (JDS/JGL)

<u>SATURDAY, MARCH 11, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

14) 8:30 - 1:00 P.M. **(10:30-10:45 A.M. BREAK)**

<u>Preparation of ACRS Reports</u> (Open)
Continue discussion of the proposed ACRS reports listed under Item 13, and the draft final ACRS report on the NRC Safety Research Program, as needed.

15) 1:00 - 1:30 P.M.

<u>Miscellaneous</u> (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.

<u>NOTE:</u>

- **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
529th ACRS MEETING
FEBRUARY 9-11, 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 February 9-11, 2006
2	Evaluation of Human Reliability Analysis (HRA) Methods Against Good Practices
	2. Evaluation of Human Reliability Analysis (HRA) Methods Against HRA Good Practices (NUREG-1792) presentation of NRC, SNL, SAIC [Viewgraphs]
3	Proposed Revisions to SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs"
	3. Technical Discussion of SRP 14.2.1 — "Generic Guidelines for Extended Power Uprate Testing Programs" presentation by NRR [Viewgraphs]
4	FERRET Reactor Vessel Fuence Methodology
	4. FERRET, A Least Squares Best Estimate Evaluation for Reactor Vessel Dosimetry presentation by NRR [Viewgraphs]
10	Future ACRS Activities/Report of the Planning and Procedures Subcommittee
	5. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - February 8, 2006 [Handout #10.1]
11	Reconciliation of ACRS Comments and Recommendations
	6. Reconciliation of ACRS Comments and Recommendations [Handout #11.1]

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Evaluation of HRA Methods Against Good Practices
	1. Table of Contents
	2. Proposed Schedule
	3. Project Status Report
	4. "Evaluation of Human Reliability Analysis Methods Against Good Practices," draft NUREG-XXXX, November 9, 2005, transmittal to ACRS by memorandum dated January 12, 2006
3	Proposed Revisions to SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs"
	5. Proposed Agenda
	6. Status Report
	7. Letter from M. Bonaca, ACRS, to N. Diaz, Chairman, dated September 24, 2003, Subject: Draft Final Review Standard for Extended Power Uprates, RS-001
	8. Memorandum from J. Larkins, ACRS, to L. Reyes, EDO, dated November 9, 2005 Subject: Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs"
	9. Standard Review Plan 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs."
4	Review of the FERRET Reactor Vessel Fluence Methodology
	10. Table of Contents
	11. Meeting Schedule
	12. Status Report
	13. U.S. Nuclear Regulatory Commission, "Final Safety Evaluation for Westinghouse Owners Group Topical WCAP-16083-NP, "Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," January 10, 2006.
	14. Letter from F.P. Schiffley, II, Westinghouse Owners Group, to US Nuclear Regulatory Commission, "Transmittal of WCAP-16083-NP, Revision 0 'Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,'" July 30, 2004.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
529th FULL COMMITTEE MEETING

February 9-11, 2006

TODAY'S DATE: February 9, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	THOMAS HERRITY	NRR/DSS/SBPB
2	GARETH PARRY	NRR/DRA
3	Don Marksberry	RES/DRAA/OERAB
4	Gary DeMoss	RES/DRAA/OERAB
5	RICHARD BARRETT	RES/DRAA
6	JIM YEROKUN	RES/DRAA
7	Doris Lewis	RES/DSARE
	Michelle Simmons	RES/DSARE
9	Ronaldo V. Jenkins	RES/DRAA
10	R Pettis Jr	NRR/DE
11	G CRANSTON	NRR/DSS/SBWB
12	D Thatcher	NRR/DE/EQVA
13	Paul Prescott	NRR/DE/EQVA
14	Margaret Chernoff	NRR/ADRO/DORL
15	Charles Ader	RES/DRAA
16	Greg Galletti	NRR/DE/EQVB
17	Bill Rogers	NRR/DE/EQVB
18	TOM ALEXION	NRR/PGCB
	Lambros Lois	NRR/DSS/SBWB.
20	Steven Jones	NRR/DSS/SBPB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
529th FULL COMMITTEE MEETING

February 9-11, 2006

TODAY'S DATE: February 9, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Matthew A. Mitchell	NRR/DCI/CVIB
2	Ramin Assa	RES/DLPM
3	Jenny Tobin	OPA
4		
5		
6		
7		
9		
10		
11		
12		
13		
14		
15		
16		
17		
18		
20		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
529th FULL COMMITTEE MEETING
February 9-11, 2006

February 9, 2006
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

	NAME	AFFILIATION
1	JOHN FORESTER	SANDIA LABS
2	ALAN KOLACZKOWSKI	SAIC
3	Djordje Vojnovic	SNSA
4	DAVID T. LANGLEY	TVA
5	Robert Smoll	TVA
6	P. SHANNON LOVVORN	TVA
7	Deann Raleigh	LIS, Scientech
8	KARL FEINTUCH	DORL
9	~~Jenny Tobin~~	~~OPA~~
10		
11		
12		
13		
14		
15		
16		
17		
18		
19		
20		

①

ITEMS OF INTEREST

529th ACRS MEETING

FEBRUARY 9-11, 2006

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
529th MEETING
February 9-11, 2006

RECENTLY ISSUED SIGNIFICANT ENFORCEMENT ACTIONS

NRC NEWS

INSIDE NRC

REGULATORY INFORMATION CONFERENCE 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-06-001

SECURITY ISSUES ASSOCIATED WITH RADIOACTIVE MATERIALS LICENSEES

Peter B. Lyons
Commissioner
Nuclear Regulatory Commission

before the
Midyear Meeting of the Health Physics Society
G. William Morgan Lecture
Scottsdale, Arizona
January 23, 2006

Introduction

Thank you for the invitation to join you here today. This provides another opportunity, in addition to our past interactions, to discuss issues of importance to the Health Physics Society. I also want to thank you for the honor of being selected as the G. William Morgan Lecturer. I am honored to be among those who previously were selected.

Let me start with some personal background, which for the last 40 years has involved work with a wide range of radiation and radiation-based technologies. My graduate training at California Institute of Technology focused on nuclear physics and its applications to astrophysics. From Cal Tech, I went to Los Alamos, where I spent my first 15 years supporting our nation's nuclear weapons test programs with extensive work in diagnostics at the Nevada Test Site. Whether developing and calibrating instrumentation for tests or working at the Site, it was a rare day when I wasn't working around or with ionizing radiation in some form. In later years at Los Alamos, I managed many projects that involved the same basic technologies.

After almost 30 years in Los Alamos, I was presented with an amazing opportunity to move to Capitol Hill to serve first as science advisor to Senator Pete Domenici, and later on the staff of the Senate Energy and Natural Resources Committee, chaired by Senator Domenici. During my 8 years on Senate Staff, I interacted on a regular basis with representatives of your Society.

With Senator Domenici's keen interest in our nation's energy security and his particular focus on the role that nuclear energy might play in achieving that security, I had many opportunities to help in the development of legislation that would advance relevant disciplines, including health physics, required for any future renaissance of nuclear power in this country.

Now at the NRC, my responsibility is, among other things, to promulgate, review, and enforce regulations governing civilian uses of radioactive materials. Many of you are involved in research that informs these regulations and some of you use radioactive materials that are regulated by the NRC. My NRC appointment is both a great honor and a demanding assignment, made all the more challenging in today's world by the concerns raised by global terrorism.

The traditional focus of the regulation of radioactive sources was the protection of workers and the public from their misuse or from accidents. Security measures were also a concern, but with the principal aim of preventing petty theft or accidental loss. The events of September 11, 2001, however, changed the way in which we must think about sources. Our perspective must now encompass the possible malevolent use of radioactive materials in weapons of terror. As a result, past practices must be modified to reflect the threat environment.

One of our concerns, of course, is that a high-risk radioactive source might be combined with conventional explosives and used in a radiological dispersal device (or RDD). Now as far as I know, RDDs are not part of the military arsenal of any country for the simple reason that they are not very good weapons. Our analyses verify that such devices would not cause large numbers of fatalities. However, RDDs might nonetheless meet a terrorist's objectives to cause panic and potential environmental contamination that could seriously disrupt normal activities in the affected area or cause significant economic impact. Thus, we must protect the public from malevolent use of high-risk radioactive sources.

The task is challenging because of the widespread use of radioactive materials throughout the world in medical practice, research, and numerous industrial applications. Moreover, domestic and international commerce in these sources is extensive, and existing controls on imports and exports, particularly for sources of low to moderate risk, are minimal.

Pre-September 11

Prior to 9/11, accidents such as those in Brazil (1987) where an abandoned radiotherapy machine containing 1400 curies of Cesium-137 was opened by junkyard workers resulting in four deaths and more than 244 persons being contaminated, and Estonia (1994) where a Cesium-137 Source was stolen from a radioactive waste facility which resulted in one fatality and four injuries, highlighted the risk from orphan sources and served as an impetus for several initiatives, both internationally and domestically. Those efforts were aimed at improving safety by recovering orphan sources as they were discovered and at increasing controls to prevent future orphan sources. The International Conference on the Safety of Radioactive Sources and the Security of Radioactive Material in Dijon, France (1998) led to the first draft of the International Atomic Energy Agency's (IAEA) "Action Plan for Safety of Radioactive Sources and the Security of Radioactive Material" (Action Plan).

From the Action Plan, further conferences and technical sessions led to the development of the IAEA "Code of Conduct for the Safety and Security of Radioactive Sources" (Code of Conduct). In addition, regional international, trans-border issues among Canada, Mexico, and the United States led to agreements prior to 9/11 to hold a series of so-called Trilateral Meetings. These Meetings were intended to establish protocols and coordination on several issues including enhanced communications regarding lost or stolen sources near borders, communications about trans-boundary shipments, and coordination of national positions on conventions dealing with radioactive materials.

Code of Conduct for the Safety and Security of Radioactive Sources

The Code of Conduct is the IAEA's framework for international cooperation in reducing the risks from radioactive sources including orphan sources. Elements of the Code of Conduct that apply to the recovery or prevention of orphan sources are:

- development of a national source registry;
- modification of import/export programs to ensure better tracking of sources;
- control over orphan sources, including promoting awareness of orphan source issues among external stakeholders; and
- management of disused sources, including the establishment, where applicable, of agreements for the return of such sources to manufacturers.

On-going NRC Efforts on the Code of Conduct

The NRC has ongoing efforts to meet the commitments made by our Government's endorsement of the Code of Conduct. These efforts have resulted in new rulemaking for import and export controls for radioactive sources and the development of a national source tracking system.

Import /Export

Strengthening the import and export controls for high-risk sources was one of the primary tenets in the IAEA Code of Conduct. In June of 2005 the NRC issued revisions to 10 CFR Part 110, "Export and Import of Nuclear Equipment and Material." The final rule became effective at the end of 2005. These additional import and export controls add reporting requirements, and determinations that the parties importing or exporting high-risk radionuclides (whether sealed or unsealed) are authorized to conduct these activities by competent authorities in the respective country.

National Source Tracking

After 9/11, there were numerous requests for the number and types of radioactive sources that existed in the United States that could be of interest to terrorists. In addition, the NRC was issuing security advisories and wanted to ensure that they were received by the appropriate NRC and Agreement State licensees. Furthermore, working groups focusing on efforts to improve security of high-risk sources needed to know the numbers and types of such sources.

Attempts to respond to the need for this type of information highlighted that there was no central database for high-risk sources. The regulations at that time simply did not require tracking of sources. Instead, the NRC and some Agreement States issued licenses with total possession limits, not a possession limit for individual sources. Regulators relied on inventory, receipt, and disposal records for licensees to provide some aspects of a paper trail.

Post 9/11, these shortcomings were readily visible. Thus, the need and resolve for national source tracking was established. Based on the U.S. Government's endorsement of the Code of Conduct, recommendation from the DOE/NRC joint report, a mandate in the Energy Policy Act of 2005 (EPAct 2005) and an NRC commitment to Congress, there is now a firm path toward implementation of a national source tracking system. However, the process of rulemaking and developing the system takes time and does not meet the nation's immediate information needs.

The short-term solution involved creation of an interim voluntary database relying on licensees to make good faith efforts to provide accurate information. Approximately 2600 NRC and Agreement State licensees were contacted to provide a "snapshot" inventory of discrete sealed sources that contained IAEA Category 1 and 2 sources and even Category 3 sources if there was a potential to have a large aggregation that would trip the Category 2 quantity. The response was outstanding. An interim database has been established and is currently being used to inform NRC efforts to improve security and better track high-risk sources. Until the national source tracking system is established, the interim database will be updated on a periodic basis.

The proposed schedule for implementing the national source tracking system reflects the need for rulemaking. A Proposed Rulemaking was issued for public comment in July of 2005. The final rule is scheduled to be published in August 2006 consistent with the EPAct 2005. After the final rule, there will be a phased implementation of the tracking system beginning in the spring of 2007.

When the proposed rulemaking was noticed, the Commission directed staff to solicit comments on the potential addition of Category 3 sources to the NSTS. To date, most of the comments opposed to the inclusion of Category 3 sources cite the increased burden that would be imposed on licensees and the NRC. Some comments, including the HPS position statement issued this month, favor inclusion of Category 3 sources; they note that these sources can be aggregated to levels well above Category 2 sources and that failure to include them will introduce a loophole. The Commission will deal with these differing points of view when the rule is finalized.

Security Measures

Additional Security Measures (ASM) have also been promulgated by NRC orders, such as those issued to panoramic irradiator licensees (June 2003) and source manufacturer or distributor licensees (January 2004). It is my understanding that during the development of these orders, the Health Physics Society played a key role in facilitating the comment process and a request for additional meetings. NRC also issued Radioactive Material Quantities of Concern (RAMQC) transportation orders to applicable licensees.

The radionuclides and the threshold limits in the RAMQC transportation orders were consistent with the IAEA Code of Conduct. These measures require background investigations, protection of sensitive information, license verification, documentation of domestic shipments and transfers, and intrusion detection and response systems. They also require the establishment of a security zone(s), access controls, coordination with local law enforcement authorities to ensure a timely response if needed, background investigations for certain employees, and protection of sensitive unclassified information. Implementation of these measures must be completed this month.

Involvement of Agreement States Enhancing Security

Another issue, which I learned about during my Senate service, involved the perspective of many States that they should play a strong role in security of sources, not just in safety.

This sensitive topic has been examined by the Commission in recent months. I believe that we have responded appropriately with an inclusive and thoughtful approach to involve the States and achieve our common objective to enhance controls over certain radioactive materials while enhancing protection of public health and safety. The approach involves recognition of the integrated nature of safety and security. I believe that the Agreement States' response to this Commission initiative will further enhance the level of mutual trust and partnership between the Agreement States and the Commission.

Both NRC and the Agreement States will continue to issue the requirements, as new licensees are identified, for authorizations to possess material above the threshold quantities. The Agreement States will inspect and enforce the requirements for their licensees. NRC will continue to coordinate with the States to assure consistent implementation. As of December 2005, NRC and all 33 Agreement States had issued a legally binding requirement for increased controls.

Implementation of the Energy Policy Act of 2005

The enactment of the EPAct 2005 added NRC regulatory authority over certain types of radioactive material that were previously excluded – specifically, certain accelerator-produced material, discrete sources of radium-226, and certain discrete sources of naturally occurring radioactive materials (other than source material). In time, this will help provide a more coherent national framework for regulation of most radioactive materials. And because the EPAct 2005 provided the Commission with authority to grant limited time waivers, the Commission has been able to maintain the "status quo" with respect to regulatory responsibilities of the States through issuance of these waivers.

However, by February 2007, NRC must issue final regulations addressing the newly covered material. Issuance of the regulations will require each State to compare its regulatory program against NRC's requirements. NRC is consulting with the States and other stakeholders in developing these regulations and, to the maximum extent practicable, will use existing model State standards in promulgating the regulations.

The EPAct 2005 contained many activities, some of which require significant cooperation between NRC and the State Radiation Control Programs to accomplish. Recognizing this, the NRC established a multi-organizational Task Force to integrate the activities. Task Force members include representatives from the NRC and State Radiation Control Programs as sponsored by the CRCPD and the Organization of Agreement States.

The Task Force is chartered to develop a framework under which activities will be planned, managed, and implemented. The Task Force is developing a detailed action plan to ensure timely and complete implementation. Task Force responsibilities related to NRC regulation of Naturally occurring and Accelerator-produced Radioactive Material (NARM) include: (1) the technical basis for the rulemaking to establish a regulatory framework for the expanded definition of byproduct material; (2) the transition plan required in the Act to assert the expanded regulatory authority and permit assumption of the authority by Agreement States; (3) development of guidance for the NARM rulemaking; and (4) regulatory program changes related to NARM.

We are hoping to make this transition as smooth as possible, both for regulators and licensees. In issuing the regulations, the Commission will also prepare and publish a transition plan describing the conditions under which States may continue to exercise authority over the newly covered byproduct material. The transition plan will provide that any Agreement between the Commission and a State covering byproduct material and entered into before the date of publication of the transition plan will be considered to include the newly covered byproduct material. Non-Agreement States that wish to regulate the newly covered material have the option of making an application to the NRC for Agreement State status.

Research and Test Reactors

Before I close, I'd like to address one other area that many members of your Society probably have recently confronted. This involves the national attention focused by the ABC television network on the security of research and test reactors.

Long prior to 9/11, security plans and procedures were required of research reactors. These requirements employed a defense-in-depth approach, which enabled the licensee to detect, delay, assess, and respond to security events. After 9/11, the NRC ensured that numerous additional security-related measures were instituted at these reactors to enhance protection against facility sabotage or theft of nuclear material. In addition to these actions, the NRC re-assessed the security of the research reactors to further determine whether any additional security measures are warranted. Results to date indicate that there are no credible scenarios that could result in significant radiological consequences to the public.

As this audience well appreciates, the radiological consequences of an attack on research reactors would be low due to the small quantities of radioactive material present, the reactor structure and shielding designs, and the safety and security measures in place. Also, attempts to sabotage the facility or steal the nuclear material would trigger a rapid armed response and activate pre-established emergency response plans. Even if a sabotage attack were attempted against a research reactor, we are convinced that the potential for significant radiation-related health effects to the public is highly unlikely.

Late in 2005, ABC aired a "Prime Time" story related to research reactor security that portrayed many current practices at research reactors to be grave national security risks. However, our evaluations to date of these concerns have not concurred with most of the so-called "security vulnerabilities" identified in the program.

As one example, ABC showed that some doors to buildings housing reactors were open and unmonitored. However, the NRC verified that the specific doors in question are to publicly accessible classroom and office buildings, which are not required to assure adequate security of the reactor. Another example from ABC was that so-called "guards" were not always present or appeared to be asleep. However, the traffic control and monitoring personnel identified by ABC to be "guards" are not required or considered by NRC for security or any other regulatory purpose.

In our evaluations, each specific concern from ABC for each research reactor is being assessed through NRC's allegation review process. Based on these assessments, NRC continues to conclude that in most cases security plans, procedures, and measures are adequate to protect public health and safety from the potential radiological effects of research reactors. In one case, implementation of security requirements was not acceptable and the NRC is ensuring that corrective actions effectively address the problem.

Furthermore, we recently issued letters to each research reactor licensee to obtain additional information and re-emphasize our expectations for maintaining effective security in the current threat environment. The information we requested will help the NRC to re-validate that the existing security requirements, as supplemented by the additional security measures conveyed to the research reactor community after 9/11, are implemented to help protect public health and safety. In addition, we have requested that ABC make any additional video or other relevant information they obtained during their study available to us in order for the NRC to ensure that all risk-significant items are addressed.

Based on our continuing review of site-specific security and our knowledge of the potential risks and threats, we continue to believe that the research reactors remain safe and secure. If as a result of the continuing research reactor oversight activities, any additional security measures are necessary to assure the health and safety of the public, the NRC will not hesitate to implement additional security measures as appropriate.

In conclusion, I want to commend the Health Physics Society for its national leadership in providing responsible, scientific evaluations of the real health risks presented by radiation and radioactive materials. During my service in the Senate and now during my service with the NRC, I have learned to value the measured, carefully developed opinions of this Society on issues of mutual interest. I look forward to many more years of these interactions.

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-05-019

IMPROVEMENTS TO THE UNITED STATES NUCLEAR REGULATORY COMMISSION'S OPERATING EXPERIENCE PROGRAM

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

before the
International Conference on Operational Safety Performance
International Atomic Energy Agency
Vienna, Austria
November 30, 2005

The subject of this conference, operational safety performance, is one of fundamental importance to the U.S. Nuclear Regulatory Commission. Thus, I am especially pleased to join you for this conference.

Operational safety performance is a keystone of the NRC's regulatory framework. It has many "tentacles" extending into areas such as maintenance, engineering, and security, as well as into operations. It can also be seen as both originating from, and feeding back to, a plant's design. The importance of this area at the NRC derives from our overall Strategic 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.***

To accomplish that Objective, we identified six key strategies to assure safety, one of which is to:

> ***Evaluate and utilize domestic and international operational experience and events to enhance decision making.***

This strategy not only enhances our decision-making process, but it is also a fundamental component of knowledge management – meaning the process by which the NRC is enabling transfer of knowledge and lessons learned from current regulators to a generation of newly hired regulators.

In international documents, this strategy may be expressed somewhat differently, but the underlying message is always fundamentally the same:

Learn from your experiences and those of others.

For example, the IAEA expresses it as "... actively promote feedback on the lessons learned from past experience ." An NEA document, adopting a prior IAEA position, adds as one aspect of regulatory effectiveness, "Strives for continuous improvements in ... performance" and emphasizes the use of a "learning organization" model .

This learning should not be confined to lessons only from the nuclear industry. Nuclear regulatory agencies should also learn from experiences in other industries and organizations that have a strong focus on safety, such as the transportation industry and space flight programs. But the question of just how a regulatory agency can optimize the process of gathering, analyzing, and using operational experiences to help ensure nuclear safety is certainly a complex issue worthy of examination at this conference.

Although the use of operating experience has long been a part of the NRC's activities (and those of its predecessor, the Atomic Energy Commission), the agency's programs have been shaped by several past events. In the late 1970s, the NRC was primarily focused on the licensing of new plants and the inspection of plant construction and commissioning. At that time, only one headquarters division and one branch in each of the five regional offices focused on operating reactors. Such limited resources did not enable any systematic method for evaluating the growing volume of licensee-reported event information. In addition, licensees themselves did not have the resources to systematically evaluate operating experience, nor at that time was any industry group available, such as the Institute of Nuclear Power Operations (INPO) or the World Association of Nuclear Operators (WANO), to perform such a function.

Critical reviews of the NRC immediately following the Three Mile Island (TMI) accident in 1979 included NRC task forces to examine our analysis and evaluation activities. In response, NRC created the Office for Analysis & Evaluation of Operational Data, or AEOD, as an internal but independent office to collect operational data, to systematically analyze and evaluate these data, to feed back lessons to improve the safety of licensed operations, to assess the effectiveness of the agency-wide program, and to act as a focal point for interaction with outside organizations for data analysis and evaluation of operational experiences. At about the same time, U.S. industry also took action to create INPO, in part to provide an independent capability to evaluate operating experience and feed back lessons learned to licensees.

During the 1990s, further evaluations were performed, resulting in a set of recommendations aimed at eliminating unnecessary functions and duplication. In 1999, as part of its initiative to streamline NRC's infrastructure, the agency implemented a significant strategic change and dissolved AEOD, transferring its core operating experience functions to two separate offices. The Office of Nuclear Reactor Regulation, or NRR, was assigned short-term operating experience functions, and the Office of Nuclear Regulatory Research, or RES, was assigned long-term efforts. During this period, the agency continued to support evolutionary improvements to the systematic processes for collecting and evaluating operating experience and communicating the lessons learned to the NRC staff and the regulated industry.

The 2002 Davis-Besse reactor vessel head degradation event was another significant event in the history of the NRC and forced another comprehensive re-evaluation of our key processes. An NRC inter-office task force in 2003 found substantial shortcomings in the agency's operating experience activities. Throughout the NRC, it was acknowledged that our operating experience programs needed reassessment. The shortcomings noted by the Davis-Besse task force were similar to those noted in the evaluations and reviews conducted after the TMI accident.

During 2004, the NRC staff developed a plan for implementing the task force recommendations and completed the framework and infrastructure for our new operating experience program for reactors and launched it on January 1, 2005. The program established a centralized clearinghouse to systematically collect, communicate, and evaluate operating experience information. It also makes significant use of information technology to make related information readily available to internal users and to the public.

A new database was created for managing all reported events, and a new Operating Experience Information Gateway Web site was launched that consolidates a large collection of individual databases and Web sources of information onto a single Web access page. We have also made it easier for the public to search operating experience in generic communications, event reports, and preliminary event notifications.

A new communication tool to promptly notify NRC staff members of new operating experience in their areas of expertise has been developed. This tool may also be used to examine emergent operating experience in selected areas. We have created teams of technical review groups to systematically and periodically assess operating experience in their specialized areas to identify trends and insights and to recommend actions. This program appears to be off to a good start. I'd like to share with you some recent examples where this program has been successful in capturing, evaluating, and disseminating operating experience information.

- The Hope Creek BWR plant experienced circumferential cracking of the recirculation pump shaft. The new operating experience program directed an increased vigor in acting on such issues in a thorough and timely manner and as a result the staff promptly issued an Information Notice to inform industry and the public of the issue. The staff continues to interact with vendor groups to identify additional issues and further regulatory actions.

- The Hatch Unit 2 BWR plant experienced safety relief valve Tee-Quencher support bolt failures. The Tee-Quenchers are the T-shaped ends of the pressure relief system that discharges reactor steam into the suppression pool when the safety relief valves are lifted. They are designed to minimize the instability associated with the large dynamic forces that occur during discharges. The staff issued a Morning Report, a very timely public information dissemination tool, and contacted General Electric and all other domestic BWR licensees to determine their Tee-Quencher configuration. Facilities with a bolted configuration performed operability determinations and determined that their systems remained operable.

- The Millstone 3 PWR plant experienced a reactor trip as a result of "tin whiskers," which are fine threads of soft metal that grow on electronic circuit boards and can cause short circuits. The staff first performed an internal Operating Experience Briefing to inform NRC management and to facilitate evaluation of the issue. The staff subsequently issued an Information Notice to inform industry of the cause of this event and related operating experience from international and non-nuclear industry sources. The NRC's Office of Nuclear Regulatory Research is currently evaluating whether tin whiskers should be identified as a new Generic Safety Issue.

Although the staff acted quickly and forcefully to this latter event, I should note my personal view that this is an issue that might well have been anticipated much sooner within the nuclear industry, before revealing itself in a plant trip. These phenomena were well known outside of the nuclear industry. We clearly must continue to make progress in our efforts to gather relevant information and must continue to improve our ability to look beyond our own industry for useful lessons.

Other recent focus areas of the NRC's operating experience program for reactors include:

- Gas intrusion or voiding in safety systems continues to be a concern at some PWRs, notably the Palo Verde and Indian Point plants. We take every one of these events seriously. Development of a Generic Letter to obtain information from domestic licensees on the subject has been approved and is underway.

- Significant design deficiencies in existing plants continue to appear, although they appear to be decreasing in frequency. One recent example was a degraded condition identified at the Kewaunee plant involving the potential loss of safety-related systems as a result of postulated flooding in the turbine building. This issue was preliminarily rated as Level 2 in the International Nuclear Event Scale (INES) and was reported to the IAEA. A similar condition has also been identified at the Surry plant as well, and an Information Notice has been issued.

- Today, we also have a heightened sensitivity to passive component degradation. As one example, a through-wall crack and leak were recently identified in FitzPatrick's torus. A Special Inspection was conducted and the event was preliminarily rated as an INES Level 2 and reported to IAEA. As another example, the increasing amount of operating experience involving degradation of underground cables has led to development of a Generic Letter.

- One other area of significance to operational safety is grid reliability. Since the August 2003 electrical grid blackout in North America, which resulted in loss of offsite power at a number of reactors, the agency has increased its attention in this area. Additional monitoring has been introduced, especially during high-power demand situations like hot summers, to ensure licensees have prompt communication mechanisms and appropriate procedures with transmission operators to minimize the impact due to any future grid disturbances. The development of a Generic Letter to obtain information from domestic licensees on the subject of grid reliability has been approved.

- NRC has also been very active with external events arising from natural phenomena this year due to domestic and international operating experiences involving the Asian tsunami and the recent hurricanes named Katrina, Rita, and Wilma. NRC is conducting thorough followup studies to identify and act upon the lessons learned from these experiences.

These are some of the areas where events and degraded conditions of actual or potential risk significance have been recently observed. For these and other areas, the agency is increasing its attention by applying the operating experience lessons learned, insights, and observations. Such applications include timely and effective internal and external communication of the relevant operating experience through briefings, Web postings, the development of generic communications, and other communication mechanisms depending on significance and generic applicability. Additional inspections are performed as necessary for events and degraded conditions of safety significance.

Even more broadly, an insight we gained from the Davis-Besse head degradation event was that NRC needed a better process to institutionalize significant lessons. To address this, we have started developing an agency-wide corrective action program to better capture, track, and document the significant lessons that must be institutionalized and that must remain understood and be carefully evaluated by future generations of NRC staff.

In addition, NRC's use of, and participation in, international operating experience forums is systematic and extensive. These experiences, such as those received through the Incident Reporting System, or IRS, and the INES, jointly developed by IAEA and NEA, are now a formal element of the NRC's operational experience screening process and are available on our internal Web site. NRC has been participating in the INES since 1993 and has fully participated in the initiative since 2001. All daily events are screened and rated, and those events that are rated Level 2 or higher are reported to IAEA typically within two business days.

NRC has also participated since the early 1980s in the IRS for the efficient exchange of operating experience. In addition to posting generic communications on our public Web site, NRC also submits all generic communications pertaining to reactor operating experience to IAEA on a quarterly basis.

Internally, the INES events and IRS reports from the international community are systematically screened and evaluated for applicability to U.S. plants. In 2005, a number of international events reported from these and other sources have been disseminated to appropriate NRC staff. A few of these events have been identified for detailed evaluation and potential applicability to our domestic reactors. For example, the circumferential break of the Essential Service Water pipe at Vandellos-2 (Spain) while operating at rated power and the shutdown of Kalpakam-2 (India) following the tsunami are currently under staff evaluation. NRC also exchanges operating experience with individual countries and the international community through routine interfaces, meetings, and agreements.

In addition, many operating experience sources are made available to the pubic and accessible by domestic and international stakeholders through the NRC public Web site and the agency's document management system which can be accessed through our Electronic Reading Room (www.nrc.gov/reading-rm.html).

In conclusion, NRC's management and use of operating experience have evolved over many years. We intend to maintain continued strong vigilance in collecting and using operating experience across related industries and across international borders. To further assure success, the Commission has specifically requested periodic updates from the staff on the agency's progress in developing a rigorous corrective action program to institutionalize the lessons we learn from our experience. And, as I noted at the beginning of my talk, this is one of the key strategies of the Commission for success in our mission.

Throughout our three decades of operation, the NRC has continued to learn from operating experience. However, we clearly must continue to improve in this key area. International sharing and use of operating experience continue to play a critical supporting role in the safety of nuclear power plants worldwide.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs Telephone: 301/415-8200
Washington, DC 20555-001 E-mail: opa@nrc.gov
Web Site: http://www.nrc.gov

No. S-05-018

"Public Confidence and the Nuclear Regulatory Commission"

Prepared Remarks by
The Honorable Gregory B. Jaczko
Commissioner
U.S. Nuclear Regulatory Commission
before the
Nuclear Power and Global Warming Symposium
Warrenton, VA
November 8, 2005

INTRODUCTION

I am glad to be here today.

I know that the subject of this conference is Nuclear Power and Global Warming. You have been engaged in discussions about whether the expansion of nuclear power offers a safe and viable alternative to the effects the burning of fossil fuels have on the environment.

While these are important issues, it is not appropriate for me, in my job as an independent regulator, to discuss the proper role of nuclear power. Decisions about contracting or expanding nuclear power are for the public to make through the actions of the Administration, the Congress, and ultimately the private sector.

The role of the Nuclear Regulatory Commission in my view is not to promote or discourage this initiative but rather to ensure that any new plant that may get built will be safe and secure. The mission of the NRC is to "license and regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment."

The most important requirement for the NRC to accomplish that mission is to ensure public confidence in what we do. The public demands it, the industry needs it, and it is our job.

I am afraid that there is a lot more work to do in this area. For example, I am often asked by members of the public if nuclear power plants are safe. This question illustrates my point. The fact that there is concern about the safety of nuclear power plants shows the NRC still has a long way to go to convince the public that it is an effective independent regulatory body that can be trusted to ensure the safe use of nuclear materials.

Working to improve public confidence is something we can all agree needs to be done. It is even more important today as we work to maintain effective regulatory oversight of the current fleet of 103 operating nuclear reactors in 33 States — as well as thousands of radioactive materials licensees throughout the country — while preparing to review applications for new nuclear power plants for the first time in decades.

I can confirm for you that the talk about a potential "nuclear Renaissance" is serious and real. Currently, it appears likely that over the next three years the NRC will receive applications from numerous utilities and consortiums to construct new nuclear reactors. In this environment I believe that the only way for the agency to proceed is to ensure that NRC staff are **wedded to safety – not shackled to schedules**. Doing this will require additional resources and a strong commitment to safety culture.

RESOURCES

Let me begin with the issue of resources, which could shackle the NRC staff if not managed correctly.

The industry should expect an efficient and effective NRC process for reviewing applications for new plants. This will require the hiring and training of hundreds of new NRC staff and additional resources for the NRC to develop guidance on enhanced margins of safety utilizing innovative measures and new policy on incorporating security into new reactor designs.

I have encouraged the Commission to work with the Congress to secure additional resources needed to achieve these goals and to ensure that reviewing new applications will not negatively affect on-going safety work.

PUBLIC'S ROLE

The public must also play its critical role in developing sound government policy as new licenses are considered. The NRC is made up of dedicated civil servants who come to work every day wanting to make the right safety decisions, and they need to hear from the public to help them do their jobs.

Of course, that dialogue can be productive only if the NRC is open and transparent in every step of the process. The NRC must be open with information and transparent in the processes we use to make decisions. In a post-September 11th world, we can not always fully achieve our goal of openness, but we can always be transparent as an agency – both to the public and to the licensees. In other words, while specific pieces of information may need to be protected for the NRC to accomplish its public safety and security mission, the *process* the Commission uses to make policy decisions should always be open, accessible, and well understood by all.

For the NRC to do its job, our stakeholders must see an unbiased agency whose primary goal is ensuring the safe use of nuclear materials.

LICENSEES' ROLE

The industry can also help improve public confidence and avoid the shackles of schedules. Any applications licensees submit for new reactors must be thorough and high-quality. The burden is on the industry to convincingly address all of the necessary safety and security issues.

The NRC should be clear and firm about its standards and must not be afraid to reject applications that do not meet them. Prematurely accepting inadequate applications will only create scheduling pressures on the NRC staff.

Only with the necessary resources – and through consistent responsible actions on the part of the NRC staff, the industry, and the public – can we be certain to break the shackles of arbitrary schedules and ensure we are ensconced in a happy marriage with safety.

SAFETY CULTURE

Beyond resources, there is another issue that we must focus on to ensure that there are no shackles on NRC staff and the industry. We must show the public that we value a questioning attitude. We must reinforce a culture at the agency and in the industry in which everyone feels empowered, emboldened and encouraged to ask the next question, the difficult question, and not to simply accept what is presented to them.

If public confidence is the key to effectively regulating the nuclear industry, the foundation is achieving an environment focused on safety and security – a concept known as safety culture. The NRC considers "safety culture" to involve a work environment where management and employees are dedicated to asking questions and promoting safety.

Safety culture at the NRC is like a pot beginning to boil. You are familiar with the proverbial "watched pot" just when you begin to see individual bubbles forming. Those first bubbles are like the divergent views at the NRC. Unfortunately, in my view, the NRC has a tendency to take the pot off of the stove before it reaches a full boil. I would like to see a raging boil of divergent views reach its way directly to the Commission to ensure we have access to all of the information we need.

If we look at the history of the nuclear industry, we find that problems almost inevitably appear as a result of a loss of this questioning attitude, a deteriorating safety and security culture. One of the biggest challenges in this arena is complacency, and unfortunately, complacency is most likely to be recognized only after it seeps in and contributes to a degraded safety and security environment.

DAVIS-BESSE

The most recent and well-investigated example of this can unfortunately be found at the Davis-Besse Nuclear Power Station in Ohio.

On March 5, 2002, the licensee for Davis-Besse discovered cracks and corrosion in the reactor pressure vessel head, which is the top of the reactor coolant system pressure boundary. During repair of the identified cracks, a cavity the size of a football was discovered that extended completely through the 6-inch thick carbon steel cap all the way down to a thin stainless steel liner.

Even after years of operating experience and armed with the information about a potential problem that the NRC provided, the industry as a whole failed to implement an effective corrective action program to identify and manage this type of cracking and corrosion. The licensee failed to effectively implement its operating experience review program and catch this corrosion before it became a serious safety issue. The NRC failed to ensure that the safety issue was identified and corrected even though it knew about generic problems with this important component of a plant.

As a result, the NRC instituted a Davis-Besse Lessons Learned Task Force and recommendations from this task force have been implemented. But our work is far from over. This event did not occur decades in the past at the infancy of this industry and the NRC, but rather only a few years ago with a mature regulator and a mature industry relying on a record of safety that led to complacency.

The Davis-Besse incident is a clear example of why the public lacks confidence in the industry and why the questioning attitude at the heart of safety culture is essential for continued nuclear reactor safety. Employees - both of the NRC and the industry - must feel empowered to ask the difficult questions. Ensuring this happens is at the core of safety culture.

EMERGENCY PREPAREDNESS

I want to wrap up my talk with an important topic that I believe serves as a barometer for how we can measure the public confidence in the NRC – emergency preparedness. After all, the emergency planning effort is the most tangible way the nuclear industry affects its neighbors.

When I travel to nuclear power plants I always try to meet with local elected officials and citizen groups. One of the most frequent issues I hear from these stakeholders is concern about the emergency preparedness plans in the 10-mile zones around the plants.

It is the NRC's responsibility to evaluate a licensee's onsite emergency plan and the agency relies on the Federal Emergency Management Agency – FEMA – to provide recommendations about the adequacy of State and local emergency plans. This system makes sense because FEMA is the agency with the emergency management expertise and the relationships with state and local governments to address all hazards.

I do believe, however, that the NRC should take prompt action to eliminate any doubts or concerns about *radiological* emergency plans. Input from FEMA is crucial but the NRC has the ultimate authority and responsibility to ensure the adequate protection of public health and safety around nuclear power plants. The Commission and the public should not be left to wonder if alert and notification procedures are in place, transportation resources are available, and reception and care centers are arranged.

I want to be able to visit any of the 65 nuclear power plant sites in this country and hear – not only from the licensees, but also from the public – that there is complete confidence in the emergency plans in place. No other outcome will more clearly demonstrate to the public that the NRC is wedded to safety and committed to improving public confidence .

CONCLUSION

As I conclude my remarks I hope to have helped frame and clarify some of the issues you are pursuing here this week.

The NRC must work to improve the confidence of the public in its capabilities and intentions to effectively regulate the nuclear industry in whatever shape it takes in the future. We can all agree that our goals should be a safe and secure future in which the health of our families and communities is guaranteed and our environment is protected. Working together – industry, the public, and the NRC – is the best way to avoid the arbitrary shackles of schedules and ensure the industry and the NRC staff remain wedded the imperative of safety.

Again, I thank you for the invitation to speak to you today, I commend for your efforts to learn more about and report on these important issues, and I look forward to any questions you may have.

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-05-017

Comments of Commissioner Jeffrey S. Merrifield
Saxton Nuclear Power Plant
November 8, 2005

- Good Morning. On behalf of the United States Nuclear Regulatory Commission (NRC), it is indeed, a pleasure to be here today to participate in the celebration of the completion of the decommissioning of Saxton Nuclear Power Plant.

- Nearly 44 years ago, on November 15, 1961, to be precise, our predecessor, the Atomic Energy Commission (AEC), issued a license to the Saxton Nuclear Experimental Corporation to operate an experimental pressurized water reactor.

- The reactor, which first went critical on April 12, 1962, was not built primarily to generate electricity. Instead, it was intended as a research and developmental program to demonstrate how a nuclear reactor could be operated under utility operating conditions.

- AEC documents dating to the early 60's describe the Saxton effort as "'generating knowledge' about getting more heat, and hence more electricity, out of nuclear fuel and thereby reducing the future costs of power generation."

- While Saxton only generated power at 23.5 thermal megawatts, a mere fraction of a modern nuclear power plant, it laid the foundation for understanding how "better and more powerful reactors" could be built in the future.

- Saxton served as a pioneer in the nuclear industry through its use of boron in cooling water to control the chain reaction and it was the first privately owned reactor to use plutonium as fuel.

- Operating more than 11 years until it shut down in May of 1972, Saxton was distinguished by the fact that it operated with neither fanfare nor serious incident. Decommissioned at a time when our current 104 reactor fleet was in its boom years, Saxton quickly faded in the memory of the AEC.

- As I was preparing to come here today, I was struck by two facts. The first is, that as far as we can tell, I am one of the first, if not the first Commissioner of either the AEC or the NRC to have visited this site.

- Part of that reason results from tradition. As a matter of practice, Commissioners typically did not attend reactor groundbreaking or commissioning ceremonies, because of a concern that this would be perceived as an endorsement of promotion of nuclear power.

- Beginning in the mid-1980's it became more of a habit for NRC Commissioners to visit operating nuclear plants to oversee their safe operations, but obviously, by that time, Saxton was long shut down.

- The second fact that struck me is that counter to my intuition, at the time Saxton was first conceived and built, virtually no consideration was given as to what to do with the reactor site when power operations were completed.

- In our society today, it would be inconceivable to think that a nuclear power plant could be licensed and built with virtually no consideration about what to do with the radioactively contaminated building after its useful life was complete. Yet that is precisely what happened in 1961.

- As a side note, Saxton did not even have to face one of the most difficult issues confronting many other reactors that have gone through decommissioning. The fuel used at Saxton was owned by the federal government and consequently was returned to the Savannah River site in South Carolina when the reactor ceased operations in 1972. Other reactors have not been so lucky in resolving the issue of where spent fuel will be sent to complete the decommissioning process.

- At the time of Saxton's shut down in 1972, the AEC was only in the very early stages of deciding what to do with these decommissioned reactors.

- In 1977, in testimony before the House Committee on Science and Technology, the NRC stated that virtually all of the 11 test reactors that had closed by that time had chosen mothballing as the alternative for decommissioning. Referencing Saxton in particular, the NRC expected that the "the residual radioactivity may be removed after about 50 years" – or about the year 2027.

- It was not until 1988, after more than 11 years of effort, that the NRC issued a final rule that required utilities to specify how they would assure that adequate funding was available to clean up a site after a plant ceased operation. In addition, this rule required them to outline how they would conduct the decommissioning, how long it would take, and how they would protect public health and safety in the process.

- For the first time, the Commission, following the lead of communities like this one, began to ask for a more robust explanation of how decommissioning would result in the unrestricted use of the site and a greater justification for choosing options such as entombment or mothballing for decommissioning sites instead of returning the site to its original condition.

- Today, if you look upon the field where a power reactor used to sit, it is hard to believe that our predecessors could have been so short sighted. While the promises of nuclear power certainly gleamed in the eyes of many Americans, it is unfortunate that it took so long for the final pages in the history of Saxton to be turned.

- Yet today is a day of celebration. Through the dedication of many local residents who participated as members of the Saxton Citizens Task Force, attended one of the many meetings held regarding decommissioning, or cheered on those who did, this effort resulted from significant community involvement and planning.

- Likewise, General Public Utilities (GPU), which built and operated the reactor, and which is now represented by First Energy, took the responsibility for the decommissioning of this site, at a cost many times in excess of the cost to build it in the first place.

- This is also an important event for my Agency, the NRC, for it represents the fulfillment of our obligation to license nuclear facilities in a manner that protects public heath, safety and the environment. Today, unlike our predecessor the AEC, environmental stewardship is a much more important element of our mission.

- Like its pioneering days of the early 1960's, Saxton is also one of the pioneers in a new effort: providing for a decommissioning that allows for productive reuse of the site by the local community. This site can be used safely for any number of activities, which is a goal we would like to achieve for every decommissioned site.

- In our nation today, we are on the precipice of a number of utilities considering the decision of whether or not to build new nuclear reactors in the United States. After a long dormancy, as many as 6-8 utilities may seek combined operating license applications with the NRC in the next few years.

- As Saxton helped to create the conditions for the operation of large nuclear reactors, the efforts of this community, this utility, and our Agency, which resulted in the decommissioning of Saxton, have also set a new stage for nuclear power. While many questions may be asked about the cost or need to build a nuclear reactor, Saxton has answered the question as to whether reactors can be fully dismantled after they fulfill their useful life. Communities all across America will benefit from the hard-fought lessons learned here in Saxton.

- Again, I want to thank you for allowing me to join you today.

January 30, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

Karen D. Cyr
General Counsel

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

SUBJECT: STAFF REQUIREMENTS - SECY-05-0203 - REVISED PROPOSED RULE TO UPDATE 10 CFR PART 52, "LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS"

The Commission has approved withdrawal of the previously published proposed rule and publication of this revised notice of proposed rulemaking, subject to the comments and changes noted below.

The staff (EDO and OGC) should give high priority to complete this rulemaking activity on schedule and provide the proposed final rule to the Commission no later than October 2006. To support this schedule, the staff may provide a proposed final rule to the Commission without review by the Committee to Review Generic Requirements. In a manner that supports the schedule, the staff should seek Advisory Committee on Reactor Safeguards feedback on technical issues, if any, during the public comment period.

Concurrently, the EDO and OGC must ensure that the three 10 CFR Part 73 security rulemakings are completed on schedule or earlier. Therefore, a strict plan and resources are to be established and managed to the set timetables for both the security and Part 52 rulemakings.

To facilitate stakeholder comments on the proposed rule, the staff should hold at least one public workshop as soon as practicable after the proposed rule is issued (within two to three weeks) to discuss the major proposed revisions and answer stakeholder questions. The staff should be open in its approach to the workshop and should seriously consider all comments and alternatives before preparing the final rule package.

The staff should solicit stakeholder comments on handling updates to environmental and emergency preparedness information, including an update to the early site permit (ESP) some time prior to submission of a COL application and other potential alternatives that contribute to the predictability and the flexibility of the process for updating environmental and emergency preparedness information in the ESP. In soliciting comments, the staff should be careful to recognize the distinctive nature of these two issues, that may necessitate potentially different alternatives.

In parallel with the issuance of the proposed rule, the staff should develop guidance to clarify the type of environmental information addressed by the 'new and significant' standard ("new and significant information on the site or design to the extent that the information differs from, or is in addition to, the information discussed in the ESP EIS").

The staff should solicit stakeholder comments on a potential requirement that Part 52 licensees update the PRA periodically throughout the life of the facility, perhaps on a schedule similar to the schedule for UFSAR updates.

The staff should solicit public comments on NEI's seven conforming changes as noted in the NEI letter dated December 14, 2005, so as to allow the agency to incorporate them in the final rule scheduled for October 2006, as appropriate.

The staff should ensure that the NEI comments made on the AP1000 Design Certification and that have general applicability are posed as questions in the Federal Register Notice in a manner that would allow the agency to incorporate them in the final rule scheduled for October 2006, as appropriate.

The staff should include a question in the statement of considerations that seeks public comment on adding a provision to the rule that would require COL applicants to submit a detailed schedule for completion of ITAAC at a specific point in time prior to fuel load.

The staff should also engage industry and public stakeholders to identify any generic regulatory process changes that could enhance the efficiency and effectiveness of preparation of COL applications in situations where a change to an applicable regulation may occur prior to completion of the staff's associated review.

The scope and methods of a PRA to be submitted should be addressed in guidance documents, not the regulations.

The language of the Atomic Energy Act does not explicitly require a mandatory hearing in connection with the issuance of a reactor manufacturing license. The staff should solicit comments on the need for mandatory hearings for issuance of manufacturing licenses, and modify the SOCs to clearly indicate that the Commission may consider changing this approach in the final rule. Should the Commission change its approach, an opportunity for hearing would still be provided, though a hearing would not then be mandatory. The staff should update the regulatory analysis to include the estimates of costs and benefits of providing a mandatory hearing for issuance of a manufacturing license.

The staff should revise 10 CFR 2.340 [initial decision in contested proceedings; immediate effectiveness of initial decision directing issuance of a CP or OL] to address (1) issuance of an ESP; (2) issuance of a COL; and (3) issuance of a finding pursuant to 10 CFR 52.103.

The staff should include in 10 CFR 52.79(a) requirements for COL applications to contain information demonstrating how the applicant will comply with 10 CFR 50.62 [ATWS requirements] and 10 CFR 50.68 [criticality accident requirements].

The staff should revise 10 CFR 50.62(d) [ATWS implementation schedule requirements] to make it applicable only to licenses issued before the effective date of the final Part 52 rule.

The proposed 10 CFR 52.63 language could be enhanced to allow the Commission to amend design certification rules to address generically any of the Design Acceptance Criteria. The staff should include a discussion of this potential enhancement in the *Federal Register* notice and request comments on whether and how the rule might be revised to incorporate such a provision. The staff should also solicit comment on the extent to which backfit-like provisions should apply when amending a design certification rule.

NRR and OGC should provide the Commission with proposed resource and organization plans for the next four years, as well as proposed strategies for staff review of expected applications and support for COL hearings before the Atomic Safety and Licensing Board Panel.

The staff should solicit public comment on whether 10 CFR Part 21 reporting requirements of defects and noncompliance should be imposed on ESP or design certification applicants.

<u>Changes to the *Federal Register* Notice</u>

1. Page 4, paragraph 2, revise line 5 to read '... CFR part 52 allow for resolving ~~resolved~~ safety ....'

2. Page 5, last paragraph, revise lines 1 and 2 to read 'A ~~Following the close of the public comment period on the July 2003 proposed rule, a~~ number of factors led the NRC to question whether the July 2003 ~~that~~ proposed ....'

3. Page 6, 1st full paragraph, revise line 13 to read '... are not adequately addressed in this proposed rule ~~(or are not adequately addressed),~~ ....'

4. Page 7, revise line 14 from the top to read '... to the design and not site-specific.'

5. Page 18, revise line 7 from the top to read '... it would be inconsistent ~~make no sense~~ for the ....'

6. Page 21, paragraph 1, revise lines 6 and 7 to read '... could be added to the appropriate sections in part 52 for combined licenses and manufacturing licenses. Inasmuch as ....' Revise line 10 to read '... with the ~~inclusion of an analogous~~ provision ....'

7. Page 26, last paragraph, revise line 12 to read '... an exemption with regard to design certification information ~~from one or more elements~~ must ....' Revise lines 13 and 14 to read '... licensee seeks an exemption from other provisions of Subpart B or other provisions of a particular standard design certification rule ~~compliance with those criteria~~ , then ....'

8. Page 27, revise line 1 from the top to read '... § 52.7. ~~However, the~~ The exemption ....'

9. Page 27, last paragraph, line 6, correct the spelling of "*Siegel*".

10. Page 31, paragraph 1, revise line 10 to read '... plant may ~~will~~ be built ....'

11. Page 32, line 5, correct the spelling of "meteorological". Revise line 10 from the top to read '... Design Criterion~~a~~ 2 ....'

12. Page 33, 1st full paragraph, revise line 4 to read '... ~~multi-unit~~ sites which already have

on them one or more licensed units to include ....'

13. Page 36, 1st full paragraph, revise line 2 to read ' ... work activities ~~, i.e., a limited work authorization (LWA),~~ at the ....' Revise line 6 to read ' ... perform ~~LWA~~ activities ....' Revise line 10 to read '... perform limited work ~~LWA~~ activities ....' Revise line 11 to read ' ... perform ~~LWA~~ activities ....'

14. Page 38, last paragraph, revise line 7 to read ' ... Administrative Procedure~~s~~ Act ....'

15. Page 43, 1st full paragraph, revise lines 3 and 4 to read ' ... would clarify ~~what~~ "elements" for which a variance ....'

16. Page 47, last paragraph, revise line 1 to read ' ... to be ~~more~~ consistent ....'

17. Page 49, revise the last line to read ' ... to all ~~future~~ applications ....'

18. Page 56, revise the last line to read ' ... incorporating DCR [Design Certification Rule] general ....'

19. Page 73, revise line 5 from the top to read ' ... proposal would not ~~wouldn't~~ require ....'

20. Page 79, revise line 3 from the top to read ' ... requirement to obtain ~~get~~ NRC ....'

21. Page 100, 2nd full paragraph, revise line 4 to read ' ... requalification program that ....'

22. Page 101, 1st full paragraph, revise line 4 to read ' ... introductory text ~~test~~ to ....'

23. Page 102, 1st full paragraph, revise line 4 to read ' ... requirements ~~for licensees~~ in 10 CFR part 21. As discussed with respect to the ....'

24. Page 104, revise the last line to read ' ... in the applicable ~~respective~~ design ....'

25. Page 114, last paragraph, revise line 7 to read ' ... hazards consideration determination ~~finding~~ for ....'

26. Page 127, last paragraph, line 4, correct the spelling of "licenses".

27. Page 133, add a period at the end of the 1st paragraph.

28. Page 135, revise line 1 to read ' ... management excellence. The ....'

29. Page 135, 1st full paragraph, revise line 10 to read ' ... original applicant~~ion~~ for that ....'

30. Page 140, last paragraph, revise line 6 to read ' ... ERA is ~~are~~ much ....'

31. Page 143, last paragraph, revise line 2 to read ' ... in 2003 sets forth ....'

32. Page 152, revise line 1 to read ' ... under § 52.103(g) for ....' Revise line 8 from the top to read ' ... one of the principal ~~principle~~ differences ....'

33. Page 157, last paragraph, revise line 5 to read ' ... firm "constructing ~~construction~~ ,

owning, ...'

34. Page 159, last paragraph, revise line 1 to read '... proposed Subpart ~~Part~~ F of ....'

35. Page 164, last paragraph, revise line 10 to read '... facility has been completed would increase~~d~~ the cost ....'

36. Page 169, last paragraph, revise line 1 to read '... explicitly addresses whether ....'

37. Page 170, revise lines 3 and 4 from the top to read '... action occurs~~red~~ when the NRC issues~~d~~ the combined license, which includes~~d~~ the authority ....'

38. Page 171, revise line 3 from the top to read '... of § 2.206~~2~~. A ....'

39. Page 171, last paragraph, revise the last line to read '... or otherwise do not ~~don't~~ require ....'

40. Page 181, revise line 7 to read '... Commission, in accordance with the ....'

41. Page 184, last paragraph, revise line 2 to read '... that each operator~~s~~ of a nuclear power reactor~~s~~ licensed ....' Revise line 3 to read '... an operator's license ....'

42. Page 188, last paragraph, revise line 3 to read '... license authorizes the ....' Revise line 6 to read '... processes authorizes the ....'

43. Page 191, 2nd full paragraph, revise line 5 to read '... NRC staff's views.' Revise line 9 to read '... in subpart A ~~B~~ of part ....'

44. Page 193, last paragraph, revise line 3 to read '... that Subpart ~~Part~~ A of ....'

45. Page 220, § 2.1(d), revise to read '... Administrative Procedure~~s~~ Act ....'

46. Page 230 revise line 1 from the top to read '... manufactured may ~~will~~ be ....'

47. Page 231, paragraph (4), revise line 1 to read '... will be Rockville ~~Bethesda~~, ...'

48. Page 234, paragraph (12), revise line 1 to read '... subpart A ~~B~~ of ....'

49. Page 243, last paragraph, revise lines 2 and 3 to read '... writing to the Chief, New Reactor Licensing Branch ~~Director, New, Research and Test Reactors Program~~, U.S. Nuclear ....'

50. Page 246, § 2.819(b), revise the last line to read '... on the denial ~~withdrawal~~ will ....'

51. Page 258, § 21.2(2), revise line 3 to read '... for ~~the~~ manufacture, ...'

52. Page 264, paragraph (ix), revise lines 1 and 2 to read '... permit was ~~sold or~~ transferred.'

53. Page 285, 1st full paragraph, revise line 5 to read '... possess R~~r~~estricted D~~d~~ata or ....'

54. Page 285, 2nd full paragraph (67.), revise line 3 to read ' ... and Regulatory Approvals.'

55. Page 321, last paragraph, revise line 5 to read ' ... that if ~~If~~ the ....'

56. Page 340, last paragraph, revise the last line to read ' ... 100 continue~~s~~ to ....'

57. Page 350, paragraph (5), revise line 2 to read ' ... assessment, or an entity participating in the proceeding pursuant to § 2.315(c), may take ....'

58. Page 353, 1st full paragraph, revise line 1 to read ' ... license ~~covered by § 51.20~~ ....'

59. Page 426, paragraph (18), revise lines 1 and 2 to read ' ... with § 50.69 of this chapter, the information ....'

60. Page 445, 2nd full paragraph, revise line 5 to read ' ... licensee may use ~~using~~ any ....'

61. Page 456, § 52.155(a), revise line 2 to read ' ... under this subpart ....'

62. Page 460, paragraph (13), revise lines 1 and 2 to read ' ... with § 50.69 of this chapter, the information ....'

63. Page 468, revise lines 3 and 4 from the top to read ' ... the manufactured reactor, the costs and benefits of SAMDAs, and the bases for not incorporating SAMDAs into the design of the reactor to be manufactured. ~~and the environmental impacts of operation of the manufactured reactor.~~'

64. Page 469, last paragraph, revise the last line to read ' ... 10 CFR 2.104 ~~2.309~~.'

65. Page 551, paragraph (k), line 1, delete the comma after "under".

66. The staff should replace references to § 50.34 in § 52.3(b)(4) with references to § 52.79.

67. The staff should delete the requirement in § 52.156 to comply with § 50.33a (antitrust reviews) to be consistent with the Energy Policy Act of 2005.

68. The staff should add a provision to § 52.177 that "An application for renewal must contain all information necessary to bring up to date the information and data contained in the previous application."

69. The staff should correct a typo in Appendix A, Section III.A. There is a reference to 10 CFR Part 51 in line 4 of this paragraph that should be a reference to 1 CFR Part 51.

Changes to the Regulatory Analysis

70. Page 6, paragraph 1, revise lines 3 and 4 to read ' ... Part 52. In ~~The NRC issued~~ a staff requirements memorandum issued on January 14, 1999, the Commission approved~~ing~~ the NRC ....' Revise the last line to read ' ... incorporate stakeholder ~~shareholder~~ comments.'

71. The staff should revise the Regulatory Analysis (RA) assumptions regarding the number

of COL applications that will reference an ESP in the next three years to assume that only three COL applicants will reference an ESP in the next three years.

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

December 21, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-05-0187 - STATUS OF SAFETY CULTURE INITIATIVES AND SCHEDULE FOR NEAR-TERM DELIVERABLES

Consistent with previous direction provided in SRM-SECY-04-0111, the staff should continue to interact with external stakeholders, build from enhancements already made to the ROP in response to the Davis Besse Lessons Learned Task Force, and develop a process for determining if an evaluation of safety culture is warranted when a plant falls into the degraded cornerstone column of the ROP action matrix.

The staff should keep the Commission offices fully and currently informed of the status of this activity, inform the Commission offices of the key elements of the process before finalizing it, and complete this activity by May 2006. The staff should complete requisite training of inspectors on the enhancements to address safety culture by the end of CY 2006.

Significant changes to the ROP addressing safety culture should be documented in the ROP guidance and/or basis documentation.

The staff should ensure that resulting modifications to the ROP are consistent with the regulatory principles that guided the development of the ROP, such that overall assessments of licensee performance remain transparent, understandable, objective, predictable, risk-informed and performance-based.

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

December 20, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-05-0219 - ISSUANCE OF NUCLEAR REGULATORY COMMISSION GENERIC LETTER 2005-XX, "GRID RELIABILITY AND THE IMPACT ON PLANT RISK AND THE OPERABILITY OF OFFSITE POWER"

The Commission has disapproved the immediate issuance of the Generic Letter. Instead, the staff should hold a public workshop as soon as practicable in early January, then inform the Commission offices of the results of the workshop and any changes to the draft Generic Letter. These actions should be completed not less than three business days prior to issuance of the final Generic Letter and not later than January 27, 2006. The workshop should include our licensees, other utilities, the North American Electric Reliability Council (NERC), Regional Transmission Organizations (RTOs), Independent System Operators (ISOs), the Federal Energy Regulatory Commission (FERC), as well as the State Public Utility Commissions (PUCs), to the extent practicable, and should seek to provide a full explanation of what is intended by the questions, what we expect for answers, and how we anticipate using the information as we move forward to ensure that nuclear power plants continue to have access to reliable offsite power. Given this date, the staff should make all appropriate efficiencies to maintain our schedule to prepare for the 2006 peak cooling season.

The Office of the Secretary will plan a public Commission meeting of the NRC and FERC. The objective of this NRC/FERC meeting is to discuss the most effective role of each respective Commission in addressing grid reliability issues and assure an integrated approach of the two key Commissions in accomplishing our national missions.

SRM M050426 directed the staff to determine whether another round of TI inspections will be needed for summer of 2006, in part by evaluating licensee responses to the generic letter, and to inform the Commission of this determination by April 28, 2006. The staff should use the best available information in making this determination, even if it does not include a complete evaluation of licensee responses to the generic letter.

IN RESPONSE, PLEASE
REFER TO: M051208A

December 20, 2005

MEMORANDUM FOR: John T. Larkins
Executive Director, ACRS

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

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, 1:00 P.M., THURSDAY, DECEMBER 8, 2005, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards (ACRS) to discuss the Committee's activities and current focus. Following its retreat in January 2006, the ACRS should inform the Commission how the Committee plans to manage the increased workload resulting from the anticipated receipt of new reactor designs and combined license applications. The ACRS shall make among its highest priorities its role in the resolution of GSI-191. The staff shall expedite efforts to provide the ACRS with information necessary to make its assessment and recommendations. The Commission continues to value the independent technical views of the ACRS on significant matters under consideration by the agency.

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

IN RESPONSE, PLEASE
REFER TO: M051121B

December 19, 2005

ЛEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

Karen D. Cyr
General Counsel

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

ꞩUBJECT: STAFF REQUIREMENTS - BRIEFING ON THE STATUS NEW REACTOR ISSUES, 9:30 A.M. AND 1:30 P.M., TUESDAY, NOVEMBER 21, 2005, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

ı the morning session, the Commission was briefed by representatives of the nuclear industry, new reactor vendors, and a nancial expert, on the status and projected schedules for submitting applications for Early Site Permits (ESPs), Design ertifications (DCs), and Combined Licenses (COLs) . During the afternoon session, the Commission was briefed by the NRC taff on the status of its new reactor licensing activities and the strategies the staff has developed to address current and ıture challenges.

he staff should pro-actively engage the industry on industry's plans for future applications in the new reactor licensing rea. The staff should use industry's projections in ongoing efforts to prepare the agency for the review of these pplications, including the efforts to utilize integrated scheduling and resource planning. The staff should continue to ncourage industry's efforts to maximize standardization of licensing applications, designs, and construction activities by :veraging, to the extent practicable, such standardization in the agency's review efforts.

o support the increased interest in new reactor licensing, the staff, in coordination with OGC should examine options to ccelerate the rulemaking schedule for revisions to 10 CFR Part 52, "Early Site Permits, Standard Design Certifications, and ombined Licenses for Nuclear Power Plants." The options examined should include:

- expediting the proposed rule that is currently before the Commission, and
- creating and fast tracking a greatly reduced scope version of the current proposed rule.

ɿe staff should provide the Commission with a discussion of the regulatory advantages, and time and resources needed for ach option, including the need for and timing of public workshops, with the goal to assure that an enhanced and stable :gulatory framework will be in place to support applicant preparation of potential COL applications to be submitted in ɔ07.

(EDO) (SECY Suspense: 12/30/05)

c: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons

Recently Issued Significant Enforcement Actions

Reactor Actions

1. Entergy Nuclear Operations, Inc. (Indian Point Units 2 & 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 & 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.

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 finding was associated with a White Significance Determination Process (SDP) finding involving the licensee's failure to properly utilize 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, who rely on information provided by the facility licensee, might not have been able to take initial offsite response measures in as timely a manner had the event degraded further.

Dominion Energy Kewaunee (Kewaunee Power Station) EA-05-176

On December 21, 2005, a Notice of Violation was issued for a violation associated with a Yellow SDP finding involving the licensee's failure to ensure that the safety-related function of the auxiliary feedwater pumps, the 480 volt safeguards buses, the safe shutdown panel, the emergency diesel generators, and the 4160 volt safeguards buses, each Class 1 systems or components, would be protected from serious flooding or excessive steam releases as a result of random or seismically induced failures of non-Class 1 systems in the turbine building. The violation cited the licensee's failure to implement design control measures as specified in 10 CFR Part 50, Appendix, B, Criterion III, "Design Control".

Nuclear Management Company, LLC (Point Beach 1 & 2) EA-05-191

On December 16, 2005, a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $60,000 was issued for a Severity Level III violation of 10 CFR 50.9 involving the licensee's failure to provide accurate information to the NRC associated with a critique of an August 2002 Emergency Preparedness drill.

Nuclear Management Company, LLC (Point Beach 1 & 2) EA-05-192

On December 16, 2005, a Notice of Violation was issued for a violation associated with a White Significance Determination Process (SDP) finding. The violation of 10 CFR 50.47 associated with a White finding involved the licensee's failure to self-identify the untimely declaration of an Alert classification during an August 2002 emergency preparedness (EP) drill.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

2443 Warrenville Road
Lisle IL 60532
Web Site: http://www.nrc.gov E-mail: opa3@nrc.gov

No. III-06-003 January 20, 2006

CONTACT: Jan Strasma (630) 829-9663
Viktoria Mitlyng (630) 829-9662

NRC STATEMENT ON JUSTICE DEPARTMENT DAVIS-BESSE ACTION

The Nuclear Regulatory Commission (NRC) appreciates the attention that the U.S. Attorney's office has devoted to this issue which culminated today with the actions announced by the Justice Department. Central to this matter was a failure by the licensee to ensure that information provided to the NRC was complete so it can effectively regulate nuclear power plant safety.

The NRC took tough, aggressive action against FirstEnergy, levying the largest fine in NRC history ($5.45 million) for violations associated with the damage to the reactor vessel head at Davis-Besse and for deliberately providing inaccurate and incomplete information to the NRC on reactor conditions. Action also has been initiated against five individuals.

The Davis-Besse actions by the NRC, and DOJ and its Environmental Crimes Section, send a strong message to the industry, emphasizing that appropriate safety margins must be maintained and that the NRC will not tolerate the failure of licensees and individuals to provide it with accurate and complete information.

The failure to comply with NRC regulations and provide accurate information led to a two-year shutdown of the Davis-Besse plant for extensive repairs, major management changes, and improvements to the safety culture of the plant staff. Only after extensive inspections and oversight did the agency permit the plant to restart in March 2004. Since that time, the plant has operated safely and successfully.

The NRC also instituted a "lessons learned" task force following the event to evaluate the agency's regulatory processes on reactor vessel head integrity and recommend improvements for both the NRC and industry. All of the nearly 50 task force recommendations have been implemented.

###

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs Telephone: 301/415-8200

Washington, DC 20555-001 E-mail: opa@nrc.gov

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

No. 06-007 January 19, 2006

DIAZ NAMES JEANNE LOPATTO FEDERAL/INTERNATIONAL ASSISTANT

Nuclear Regulatory Commission Chairman Nils J. Diaz has appointed Jeanne Lopatto, a veteran of 22 years in federal government, to the newly created position of special assistant for Federal and International Programs.

Ms. Lopatto, whose most recent government service was as Director of Public Affairs for the U.S. Department of Energy, will report directly to Diaz.

"I'm pleased to have someone with Ms. Lopatto's breadth of experience on my office staff. Her background will enable the NRC to have a more cohesive federal and international liaison effort," Diaz said in announcing her appointment to his personal staff.

In her position at the Energy Department, Ms. Lopatto served as the spokesperson for then Secretary Spencer Abraham, as well as other department officials, and managed DOE's media and public relations programs. She was a member of official delegations to a variety of international conferences on topics including oil and gas development, oil supply issues, nuclear power, climate change, nuclear non-proliferation programs and technology development. She organized a number of media events at meetings of the International Atomic Energy Agency, international energy forums, the International Energy Agency, meetings of the G-8 energy ministers and U.S.-Russia Commercial Energy Summits.

As Director of Public Affairs at DOE, she worked closely with other government agencies, including the Departments of State, Commerce and Homeland Security, the Environmental Protection Agency and the National Security Council.

Prior to that Ms. Lopatto had a long career on Capitol Hill, including her service as the press secretary for the U.S. Senate Judiciary Committee under Chairman Orrin G. Hatch, R-Utah. She also worked in media positions on Sen. Hatch's personal staff as well as the Senate Labor and Human Resources Committee.

She holds a B.A. in American Studies from Dickinson College in Carlisle, Pa.

###

Inside NRC

Volume 28 / Number 3 / February 6, 2006

Emergency core cooling reg guide previewed before ACRS members

The NRC's regulatory guide for implementation of alternative, performance-based emergency core cooling requirements in proposed rule 10 CFR 50.46(a) will be largely finalized by the end of the year, agency staff told members of the Advisory Committee on Reactor Safeguards last month. The reg guide should be ready for public comment in July and for review by the full ACRS in December, Timothy Collins of the Office of Nuclear Reactor Regulation (NRR) said at a Jan. 25 joint meeting of the ACRS subcommittees on regulatory policies and practices and thermal-hydraulic phenomena. The proposed rule is scheduled to be reviewed by the ACRS in September, he said.

Existing guidance on radiological consequences of lossof-coolant accidents (LOCAs) would remain "valid" under the proposed rule, and NRC staff doesn't anticipate that licensees will have to make many changes in this area if they adopt it, Collins said. However, "one caution" is that "proposed plant changes could invalidate assumptions in current guidance" on radiological consequences, he said. For example, modifying criteria for activation of sprays in a post-LOCA containment, which has been noted by some in the industry as a desirable option that would be opened by the proposed rule (INRC, 18 April '05, 1), might require updating a unit's radiological consequences analysis, he said.

"LOCA frequency estimates used to support 50.46(a) assumed historical operating conditions," and "significant changes...could invalidate applicability" of those estimates, which were developed by a two-year expert elicitation process, Collins said (INRC, 28 Nov. '05, 1). "Significant changes need to be assessed for impact on LOCA frequency," which NRC staff "will have to work out on a case-by-case basis," he said.

Acceptable methods for emergency core cooling systems (ECCS) analysis generated a great deal of discussion at the meeting. Under the proposed rule, pipe breaks larger than a transition break size (TBS) specified in the proposed rule (INRC, 8 Aug. '05, 1) could be analyzed using criteria in Appendix K of 10 CFR 50, in Regulatory Guide 1.157, or "another analytical approach" selected by the licensee, Ralph Landry of NRR said in his presentation. NRC staff has "not defined what that [alternate approach] is" in the draft reg guide, and licensees would not be required to submit their approach for agency approval, Landry said. However, they would be required to "maintain documentation available for NRC audit."

ACRS Chairman Graham Wallis challenged the way in which uncertainty analyses would be used under the proposed rule, suggesting that even a high degree of confidence in those calculations might not be adequate. Current cooling regulations state that a post-LOCA reactor core "MUST remain amenable to cooling," a requirement with "no probability at all," Wallis emphasized repeatedly. "These are very clear statements," and "there's no consumer product or safety product that would ever be marketed with a 70% probability of working," he noted.

For containment analyses under the proposed rule, acceptance criteria for breaks below the TBS would be the "same as used today," Edward Throm of NRR said in his presentation. Analysis of breaks above the TBS would "use currently approved computer models" with "realistic initial conditions" and "realistic treatment of break flow and heat structures," but would not include single-failure or off-site power criteria for engineered safety systems, he said. "Nonsafety grade equipment may be credited, but must be maintained available and be capable of performing [its] credited function under the associated accident conditions," Throm said.

Acceptance criteria for containment in the case of breaks above the TBS would specify that the containment and containment structures be able to "withstand peak pressure without loss of integrity based on ASME [American Society of Mechanical Engineers] code limits," and on criteria in NRC's standard review plan for concrete and steel containments, Throm said.

"Strong containments are really important to the mitigation of severe accident consequences," and staff should "avoid something that decrease the effectiveness of containments" under its proposed rule, ACRS member Richard Denning said.

Hans Ashar of NRR said his office is conducting studies with the agency's Office of Nuclear Regulatory Research on "effects of various types of containment degradation." Ashar said that the criteria in the standard review plan "will be of the same robustness as before" the proposed rule.
Wallis said he found it "a bit strange" that the proposed rule and regulatory guide only apply to current reactors. Mark Rubin of NRC's probabilistic risk assessment branch responded that this is because some new designs have already completed the certification process and hence are "sort of frozen in a time warp." Collins said the NRC is requesting public comment on this issue and "everything's up for change at this point."

RISP analyses
One of the larger changes under 50.46(a) would be its requirement that "a licensee who wishes to make changes the facility or procedures or to the technical specifications shall perform a RISP (risk-integrated safety performance) assessment," Stephen Dinsmore of NRR said in his presentation. This assessment "must demonstrate that all plant changes satisfy the acceptance criteria in the rule: acceptable changes in risk; defense-in-depth is maintained; adequate safety margins are maintained; and adequate performance management programs are implemented." For those that choose to adopt the new rule, RISP analyses "should be applied to every change the licensee makes, regardless of the mechanism," and not merely changes under the new rule, Dinsmore said.

"The proposed rule authorizes licensees to make facility changes without prior NRC approval" under current regulations (10 CFR 50.59) "when the increase in the estimated risk is minimal compared to the overall plant risk profile," Dinsmore said. Quantitative guidelines to define "minimal" risk are needed, he said, because the proposed rule "introduces the consideration of the change in risk into every decision." Because RG 1.174, the NRC's guidance on risk informed regulation "does not provide any guidance about when a proposed risk-informed change need not be approved by NRC," a new guideline was needed, he said. The comment period for the proposed rule has been extended to March 8, NRC said in a Jan. 25 Federal Register notice. NEI and the Westinghouse Owners Group had requested the extension in December. The comment period was originally scheduled to close Feb. 6. The extension also applies to comments on the agency's report on seismic considerations for the transition break size, NRC said (INRC, 26 Dec. '05, 12).

Comments may be submitted online at http://ruleforum.llnl.gov.—*Steven Dolley, Washington*

All commissioners but McGaffigan on board for latest Part 52 revisions

In a 4-1 vote, the NRC commissioners approved the staff's request to publish a revised rule on new plant licensing processes and directed that the rulemaking be given high-priority so that a final proposed rule can be completed by October.

Commissioner Edward McGaffigan was the lone dissenter. While the other commissioners agreed to the staff's proposal to withdraw a 2003 proposed rule revising 10 CFR Part 52 and start anew with a comment-and-response period on another version, McGaffigan favored salvaging the three-year-old proposal. Part 52, issued in 1989, provided alternatives to the traditional two-step licensing process under 10 CFR Part 50. The licensing processes in Part 52 include the early site permit (ESP), combined construction permit-operating license (COL), and design certification.

Four designs have now been certified under Part 52—the ABWR (May 1997), System 80+ (May 1997), AP600 (Decembe2006). The other two licensing processes have never been fully executed, although three companies have applied for an ESP and several utilities are starting to prepare COL applications.

McGaffigan said he was persuaded by arguments made by the industry that the latest Part 52 proposal is too problematic and should be significantly pared down. In his Jan. 5 vote sheet, McGaffigan discussed his internal agency battle to get background documents on the proposal publicly released and described what he viewed as an obstinate staff response to the commission's urging that the rule be condensed and put on a fast track. "My colleagues, perhaps reluctantly after being backed into a corner by the staff, have decided to go forward with this dump truck of a proposed rule hoping that, with public commenters' help, we can find the jewel box of needed Part 52 changes somewhere in the dump truck before the commission issues a final rule," McGaffigan wrote.

But McGaffigan was skeptical that the public comment process would be able to assist in the search for jewels in the mammoth rule.

He questioned whether the agency should have anything less than a 120-day comment period for a rulemaking package containing 551 pages. He pointed out that the White House Office of Management & Budget process for clearance of information collection requirements would take time, and he also said he would be reluctant to allow the Advisory Committee on Reactor Safeguards (ACRS) to forgo a review of the final rule. But he said he would not object to waiving the Committee to Review Generic Requirements (CRGR) review on backfit issues.

The Jan. 30 staff requirements memorandum (SRM) allows the staff to bypass the CRGR review and directs it to seek ACRS input during the public comment period rather than conduct a separate review.

The SRM tells the staff to schedule a public workshop as quickly as possible—within two to three weeks—to discuss the revisions and allow for input, which should be considered for incorporation in a final proposed rule.

The commissioners said in the SRM that three rulemakings on updates to 10 CFR Part 73 security requirements should be given equal priority to the Part 52 revisions so that they can be completed "on schedule or earlier."

Too complex, NEI says

Marvin Fertel, senior vice president and chief nuclear officer for the Nuclear Energy Institute (NEI), told the commission last year that the industry largely feared that trying to undertake such an extensive revision of the rule would destabilize the licensing process, particularly since some companies were about a year away from developing their

COL application. He said the complexity of the rule would make it difficult for the industry, let alone members of the public and others, to "constructively and effectively" provide feedback.

"A major problem is that the proposed rule changes the fundamental principles of the Part 52 regulation," Fertel wrote in a Dec. 14 letter to NRC Chairman Nils Diaz. Whereas the existing rule references administrative and technical requirements found in other parts of NRC's regulations, the staff's proposed changes appear to steer away from that principle and instead partially reference and incorporate other requirements into the rule.

Fertel asserted the "sheer number of changes (more than 150) makes the overall impact of these changes difficult to discern." Fertel attached to his five-page letter multiple attachments with the industry's early analysis of the changes. He organized the comments into sections breaking down the changes into those considered to be "conforming" or beneficial and those having "no clear benefit or need."

Commission views

The commissioners took note of the comments and acted on NEI's suggestion for holding a public workshop. In its SRM it also told the staff to seek public comments on seven conforming changes noted by NEI and its comments on the AP1000 design certification and others that have "general applicability."

Diaz voted twice on the staff's proposal. In his first set of comments, he said the revised proposed rule "will contribute to the clarity and predictability of Part 52." But he also included areas of the rule that he believed should receive further public comment. In supplemental comments, Diaz said language should be modified so that the NRC could amend design certification rules to address any of the design acceptance criteria "generically when such an amendment would improve the specificity, certainty, or clarity of a certified design."

Commissioner Jeffrey Merrifield said in his vote sheet that the ACRS and CRGR reviews were not needed in this rulemaking because the proposed changes were "process-oriented rather than technical in nature." While supporting the new proposed rule revisions, Merrifield said he didn't believe all the changes were needed.

Commissioner Gregory Jaczko said the debate over the rule changes should be expanded beyond the internal and NEI discussions. He urged that the rule be published so that all members of the public could participate and contribute their views. Commissioner Peter Lyons disagreed with the industry that moving forward with the proposed rule would inject regulatory uncertainty because the first wave of COL applicants would be ready to file around the time the rule is issued. He said the changes would provide long-term stability and that there would never be a "perfect" time to make changes if there is a "progression of COL applications," as is anticipated.—*Jenny Weil, Washington*

RIC 2006

Program and Schedule

Updated: February 2, 2006 (9:27am)

Tuesday, March 7, 2006

sion | Time | Room	Session Information	Contact Information
9:00 - 9:30 am Room D & E	**Conference Opening and Welcome** • Jim Dyer, Director (D), Office of Nuclear Reactor Regulation (NRR), Nuclear Regulatory Commission (NRC) • Luis A. Reyes, Executive Director for Operations (EDO), NRC • Carl J. Paperiello, D/Office of Nuclear Regulatory Research (RES), NRC	**Session Coordinator (SC):** Tilda Y. Liu 301-415-1315 email: TYL1@nrc.gov
P2 9:30 - 10: 30 am Room D & E	**Plenary Session: Presentation | Q&A Session** *NRC Chairman Nils J. Diaz*	**SC:** Richard P. Croteau 301-415-1750 email RXC2@nrc.gov
	BREAK 10:30 - 11:00 am	
P3 11:00 - 11:30 am Room D & E	**Plenary Session: Regulatory Trends** • Jim Dyer, D/NRR/NRC	**SC:** Tilda Y. Liu 301-415-1315 email: TYL1@nrc.gov
P4 11:30 am - 12:30 pm Room D & E	**Plenary Session: Presentation | Q&A Session** *Commissioner Edward McGaffigan, Jr.*	
	LUNCH BREAK 12:30 - 2:00 pm (NO FORMAL LUNCHEON PLANNED)	
	BREAKOUT SESSIONS - Set #1: 2:00 - 3:30 pm Notes: 1st letter = session day - T=Tuesday, W=Wednesday, and Th=Thursday The number (1, 2, 3, 4, and 5) = the session set. The next letter(s) (BC, D, E, F, and GH) = the room	
BC Tuesday 2:00 - 3:30 pm Room B & C	**FUELS - *Cladding Behavior for Regulatory Applications*** **Session Sub-topics:** - Cladding Criteria for 50.46 - Spent Fuel Storage and Transportation - Reactivity Insertion Accidents **Chair:** Farouk Eltawila, D/Division of Systems Analysis and Regulatory Effectiveness (DSARE)/RES/NRC **Co-Chair:** Frank M. Akstulewicz Jr., Branch Chief (BC), Nuclear Performance and Code Review Branch (SNPB)/ Division of Safety Systems (DSS)/NRR/NRC **Panelists:** • Paul Clifford, Senior Reactor Engineer, Nuclear Performance and Code Review Branch, DSS/NRR/NRC • Robert Einziger, Senior Materials Engineer, Structural and Materials Section, Technical Review Directorate, Spent Fuel Project, Office of Nuclear Materials Safety and Safeguards (NMSS)/NRC • Albert J. Machiels, Senior Program Manager,EPRI • Ralph Meyer, Senior Technical Advisor, Safety Margins and Systems Analysis Branch, DSARE/RES/NRC • Rosa Yang, Technical Executive, EPRI	**SC:** Michelle E. Flanagan 301-415-6461 email MEF@nrc.gov

RIC 2006

Program and Schedule

Updated: February 2, 2006 (9:27am)

Tuesday, March 7, 2006

Session | Time | Room	Session Information	Contact Information
T1D Tuesday 2:00 - 3:30 pm Room D	**FIRE PROTECTION** ***Risk-Informed and Performance-Based*** **Session Sub-topics:** - NRC Activities in Support of NFPA 805 Implementation - NFPA 805 Implementation Guidance - NFPA 805 Transition **Chair:** James E. Lyons, D/Division of Risk Assessment (DRA)/ NRR/NRC **Panelists:** • Joe W. Donahue, VP, Nuclear Engineering and Services Department, Progress Energy • Alex Marion, Senior Director of Engineering, Nuclear Energy Institute (NEI) • Sunil D. Weerakkody, Chief, Fire Protection Branch (AFPB), Division of Risk Assessment (DRA), NRR/NRC	SC: Paul W. Lain 301-415-2346 email PWL@nrc.gov
T1E Tuesday 2:00 - 3:30 pm Room E	**LICENSING ISSUES** **Session Sub-topics:** - Power Uprates - Alternate Source Term - NOEDs - Licensing **Chair:** Catherine Haney, D/Division of Operating Reactor Licensing (DORL)/NRR/NRC **Panelists:** • Pamela B. Cowan, Licensing and Regulatory Affairs Director, Exelon Nuclear • John F. McCann, Licensing Director, Entergy Nuclear Operations, Inc. • Michael D. Tschiltz, Deputy Director (DD)/DRA/NRR/NRC	SC: Travis L. Tate 301-415-8474 email TLT@nrc.gov
T1F Tuesday 2:00 - 3:30 pm Room F	**LICENSE RENEWAL** **Session Sub-topic:** License Renewal for Future Applicants **Chair:** Frank P. Gillespie, D/Division of License Renewal (DLR)/NRR/NRC **Panelists:** • Kenneth Chang, BC, License Renewal Branch C, Division of License Renewal/NRR/NRC • Rani Franovich, BC, Environmental Branch B, Division of License Renewal/NRR/NRC • Patricia Lougheed, Lead License Renewal Inspector, Engineering Branch 2, Division of Reactor Safety, Region III (RIII)/NRC • Louise Lund, Branch Chief, License Renewal Branch A, Division of License Renewal/NRR/NRC • Garry G. Young, Manager, Project Management License Renewal Services, Entergy Nuclear	SC: Stephen T. Hoffman 301-415-3245 email STH@nrc.gov

RIC 2006

Program and Schedule

Updated: February 2, 2006 (9:27am)

Tuesday, March 7, 2006

Session | Time | Room	Session Information	Contact Information
TGH Tuesday 2:00 - 3:30 pm Room G & H	**RULEMAKING** **Session Sub-topics:** - Regulatory philosophy of rulemaking - Public advocates' views about the rulemaking process, and regulatory stability - Lessons learned from the Part 26 rulemaking transparency in the rulemaking process - E-rule benefits and the public interests OR - Petition Case Study - considering whether we want someone to discuss an emergency planning petition as an example of the transparency and safety culture interests. **Chair:** Christopher I. Grimes, D/Division of Policy and Rulemaking (DPR)/NRR/NRC **Panelists:** TBD •	**SC:** Michael A. Dusaniwskyj 301-415-1260 email MAD1@nrc.gov
BREAK 3:30 - 4:00 pm		
BREAKOUT SESSIONS - Set #2: 4:00 - 5:30 pm		
T2BC Tuesday 4:00 - 5:30 pm Room B & C	**SEVERE ACCIDENT RESEARCH** **Session Sub-topics:** - International Co-op Program - Source Term **Chair:** Sher Bahadur, DD/Division of Systems and Regulatory Effectiveness, RES/NRC **Co-Chair:** James E. Lyons, D/DRA/NRR/NRC **Panelists:** • Robert Henry, Ph.D. - Senior VP, Nuclear Group, Fauske & Associates, LLC (FAI) • Thomas Kress,Member, Advisory Committee on Reactor Safeguards (ACRS)/NRC • Charles G. Tinkler, Senior Advisor for Severe Accident Research, Safety Margins and Systems Analysis Branch (SMSAB), Division of Systems Analysis and Regulatory Effectiveness (DSARE)/RES/NRC • Michel Vidard, Sr. PM, Service d'Etudes et Projets Thermiques et Nucléaires (Service of Studies and Thermal and Nuclear Projects) (EDF-SEPTEN), Electricité de France	**SC:** Daniel C. Forsyth 301-415-5674 email DCF1@nrc.gov

Tuesday, March 7, 2006

Session | Time | Room	Session Information	Contact Information
T2D Tuesday 4:00 - 5:30 pm Room D	**NEW REACTOR LICENSING, PREPARING FOR COMBINED LICENSE REVIEWS** **Session Sub-topics:** - Update on Energy Bill and Standby Support for COL Applicants Industry Perspective on New Reactor Licensing Issues - Licensing Infrastructure: Regulatory Guidance for COL Applications and SRP Update - Design Centered Approach to COL Application Reviews **Chair:** David B. Matthews, D/Division of New Reactor Licensing (DNRL)/NRR/NRC **Co-Chair:** Jose A. Calvo, DD/DNRL/NRR/NRC **Panelists:** • Joseph Colaccino, Senior Project Manager, Division of New Reactor Licensing, New Reactor Licensing Branch, NRC • Joseph D. Hegner, Licensing Lead, North Anna COL Project, Dominion • Phillip Ray, Senior Project Manager, Division of New Reactor Licensing, New Reactor Infrastructure Planning Branch, NRC • Rebecca Smith-Kevern, Acting Associate Director for Nuclear Power Technology, Safety and Security, Office of Nuclear Energy, Science, and Technology, U.S. Department of Energy (DOE)	**SC:** Eric R. Oesterle 301-415-1365 email ERO1@nrc.gov
T2E Tuesday 4:00 - 5:30 pm Room E	**USE OF OPERATING EXPERIENCE (Regulator/Operator/Licensee)** **Session Sub-topics:** - Use and application of current operating experience from the perspectives of several stakeholders (Regulator, Industry, Foreign Regulator) **Chair:** Patrick L. Hiland, DD/Division of Inspection and Regional Support (DIRS)/NRR/NRC **Panelists:** • Vincent Coulehan, Manager of Operating Experience, Entergy Nuclear Northeast • John Kauffman, Sr. Reactor Systems Engineer, Division of Systems Analysis and Regulatory Effectiveness (DSARE), Office of Nuclear Regulatory Research (RES)/NRC • Barry Kaufer, Deputy Head, Nuclear Safety Division, Nuclear Energy Agency (NEA) • Mary Jane Ross-Lee, BC, Operating Experience Branch, Division of Inspection and Regional Support (DIRS), NRR/NRC • Mark Satorius, DD, Division of Reactor Projects, RIII/NRC	**SC:** Brett A. Rini 301-415-3931 email BAR3@nrc.gov

RIC 2006

Program and Schedule

Updated: February 2, 2006 (9:27am)

** DRAFT **

Tuesday, March 7, 2006

<table>
<tr><th>sion | Time | Room</th><th>Session Information</th><th>Contact Information</th></tr>
<tr><td>T2F

Tuesday
4:00 - 5:30 pm

Room F</td><td>ALLEGATIONS

Session Sub-topics:
- Alternative Dispute Resolution (ADR)

Chair: Eugene V. Imbro, DD/Division of Engineering (DE)/NRR/NRC
Co-Chair: Lisamarie Jarriel, Agency Allegations Advisor, Office of Enforcement (OE)/NRC

Panelists:
• Samuel J. Collins, Regional Administrator (RA), Region I (RI)/NRC
• J. Bradley Fewell, Lead Counsel, Exelon Nuclear
• Billie Pirner Garde, Attorney at Law, Clifford & Garde
• Nick Hilton, Senior Enforcement Specialist, OE
• Rocco Scanza, Director, Institute on Conflict Resolution at Cornell University</td><td>SC:
Russell J. Arrighi
301-415-0205
email RJA1@nrc.gov</td></tr>
<tr><td>T2GH

Tuesday
4:00 - 5:30 pm

Room G & H</td><td>EMERGENCY PREPAREDNESS
Lessons Learned from Hurricanes Katrina and Rita

Session Sub-topics:
- Emergency Preparedness Program Update
- Path Forward

Chair: Michael J. Case, D/DIRS/NRR/NRC
Co-Chair: Eric J. Leeds, D/Division of Preparedness and Response, Office of Nuclear Security and Incident Response (NSIR)/NRC

Panelists:
• Michael S. Beeman, D/National Preparedness Division/Acting Chief of External Affairs, FEMA Region II
• Linda L. Howell, Chief, Response Coordination Branch, Region IV (RIV)/ NRC
• Tab Troxler, Director of Emergency Preparedness, Department of Emergency Preparedness, St. Charles Parish, Louisiana
• Joe Venable, Vice President, Operations - Waterford 3, Entergy Operations</td><td>SC:
Yen-Ju Chen
301-415-5615
email YJC@nrc.gov</td></tr>
</table>

TUESDAY RECESS 5:30 pm

RIC 2006

Program and Schedule

Updated: February 2, 2006 (9:27am)

Wednesday, March 8. 2006

Session | Time | Room	Session Information	Contact Information
P5 **8:30 - 9:00 am** **Room D & E**	**Plenary Session: RES** • Carl J. Paperiello, D/RES/NRC	**SC:** C. E. (Gene) Carpenter 301-415-7333 email CEC@nrc.gov
P6 **9:00 - 10: 00 am** **Room D & E**	**Plenary Session: Presentation | Q&A Session** *Commissioner Jeffrey S. Merrifield*	**SC:** David L. Skeen 301-415-1850 email DLS@NRC.gov

BREAK 10:00 - 10:30 am

BREAKOUT SESSIONS - Set #3: 10:30 am - 12:00 pm
Notes: 1st letter = session day - T=Tuesday, W=Wednesday, and Th=Thursday. The number (1, 2, 3, 4, and 5) = the session set. The next letter(s) (BC, D, E, F, and GH) = the room

Session | Time | Room	Session Information	Contact Information
W3BC **Wednesday** **10:30 am - 12:00 pm** **Room B & C**	**RISK-INFORMED REGULATORY STRUCTURE FOR FUTURE REACTORS** **Session Sub-topics:** - NRC Technology Neutral Framework (TNF) - ASME Related Codes & Standards - DOE Framework - Rulemaking & Implementation **Chair:** Charles E. Ader, D/Division of Risk Analysis and Applications (DRAA)/RES/NRC **Co-Chair:** Michael D. Tschiltz, DD/Division of Risk Assessment (DRA)/NRR/NRC **Panelists:** • Kenneth R. Balkey, Vice President, The American Society of Mechanical Engineers (ASME) Nuclear Codes and Standards, Consulting Engineer, Westinghouse Electric Company • Mary T. Drouin, PRAB/DRAA/RES • Mark R. Holbrook,Advisory Engineer, Nuclear Regulatory Support Programs, Idaho National Laboratory • Eileen M. McKenna,Branch Chief,Financial, Policy and Rulemaking Branch (PFPB), Division of Policy and Rulemaking (DPR), NRR/NRC	**SC:** Todd A. Hilsmeier 301-415-6788 email TAH1@nrc.gov
W3D **Wednesday** **10:30 am - 12:00 pm** **Room D**	**GSI 191** **Chair:** Brian W. Sheron, Associate Director for Engineering and Safety Systems (ADES)/NRR/NRC **Panelists:** • John C. Butler, Sr. Project Manger (PM)/NEI • Maurice E. Dingler, Technical Staff Engineer, Wolf Creek Nuclear Operating Corporation • Paul A. Klein, Sr. Materials Engineer, Steam Generator Tube Integrity and Chemical Engineering Branch (CSGB), Division of Component Engineering (DCI)/NRR/NRC • Thomas O. Martin, Director, DSS/NRR/NRC • Robert L. Tregoning, Group Leader, GSI-191 Research Activities, Engineering Research Applications Branch (ERAB), Division of Engineering Technology (DET)/RES/NRC	**SC:** Sean E. Peters 301-415-1842 email SEP@nrc.gov

RIC 2006

Program and Schedule

Updated: February 2, 2006 (9:27am)

Wednesday, March 8, 2006

ssion | Time | Room	Session Information	Contact Information
E Wednesday 10:30 am - 12:00 pm Room E	**SAFETY CULTURE INITIATIVES & IMPLICATIONS** **Chair:** Ho K. Nieh, Jr., DD/Division of Policy and Rulemaking (DPR)/NRR/NRC **Co-Chair:** Michael R. Johnson, D/OE **Panelists:** • Bruce A. Boger, Associate Director for Operating Reactor Oversight and Licensing (ADRO)/NRR/NRC • Michael T. Coyle, VP Operations, Nuclear Energy Institute (NEI) • David Lochbaum, Director, Nuclear Safety Project, Union of Concerned Scientists (USC)	**SC:** Sara M. Bernal 301-415-1027 email SMB5@nrc.gov
W3F Wednesday 10:30 am - 12:00 pm Room F	**INTERNATIONAL PERSPECTIVES** - New Reactor Design Reviews **Session Sub-topics:** - Licensing of Advanced Reactors **Chair:** William W. Borchardt, DD/NRR/NRC **Co-Chair:** Janice Dunn Lee, D/Office of International Programs (OIP)/NRC **Panelists:** • André-Claude Lacoste, Director General, DGSNR, France • Li Ganjie, Director General, National Nuclear Safety Administration, State Environmental Protection Administration, China • Jukka Laaksonen, Director General, Radiation and Nuclear Safety Authority, Finland	**SC:** Michael C. Cullingford 301-415-1276 email MCC@nrc.gov
GH Wednesday 10:30 am - 12:00 pm Room G & H	**YUCCA MOUNTAIN** **Session Sub-topics:** Spent Fuel Disposal **Chair:** Jack Strosnider, D/Office of Nuclear Materials Safety and Safeguards (NMSS)/NRC **Co-Chair:** Cynthia A. Carpenter, D/Program Management, Policy Development and Planning Staff (PMAS)/NRR/NRC **Panelists:** • Paul Golan, Principal Deputy Director, Office of Civilian Radioactive Waste Management, Department of Energy (DOE) • Steven Kraft, D/ Waste Management, Nuclear Energy Institute (NEI) • Robert Loux, Executive Director, Nuclear Waste Project Office, State of Nevada • Bill W. Reamer, D/Division of High Level Waste Repository Safety, Office of Nuclear Material, Safety and Safeguards (NMSS)/NRC	**SC:** Steven L. Baggett 301-415-8584 email SLB@nrc.gov
	BREAK 12:00 pm - 12:30 pm	
	LUNCH BREAK 12:30 pm - 2:00 pm (NEI LUNCHEON OR LUNCH BREAK)	
	BREAK 2:00 - 2:30 pm	
P7 2:30 - 3: 30 pm Room D & E	**Plenary Session: Presentation | Q&A Session** *Commissioner Gregory B. Jaczko*	**SC:** Josh Batkin 301-415-1820 email JCB3@nrc.gov

Wednesday, March 8, 2006

Session | Time | Room	Session Information	Contact Information
BREAK 3:30 - 4:00 pm		
BREAKOUT SESSIONS - Set #4: 4:00 - 5:30 pm		
W4BC **Wednesday** **4:00 - 5:30 pm** **Room B & C**	**ADVANCED REACTORS - GEN IV** ***Research and Licensing Strategies for the NGNP*** **Session Sub-topics:** - NRC Adv Rx Program - DOE/INL NGNP - HTGR Technology - Energy Policy Act - Joint Licensing Strategy **Chair:** James G. Danna, Acting Chief, Advanced Reactors and Regulatory Effectiveness Branch (AREAB)/DSARE/RES/NRC **Co-Chair:** Laura A. Dudes, Chief (C)/New Reactor Licensing Branch (NRBA)/(DNRL)/NRR/NRC **Panelists:** • Trevor Cook, NGNP Program Manager, DOE • Andrew Kadak, Professor, Massachusetts Institute of Technology • Larry Parme, GT-MHR Licensing Manager, General Atomics • Stuart Rubin, Sr. Technical Advisor, Advanced Reactor Program Manager, AAREB/DSARE/RES/NRC • Edward G. Wallace, Sr. General Manager US Programs, PBMR Pty Ltd.	**SC:** Kent B. Welter 301-415-5740 email KBW@nrc.gov
W4D **Wednesday** **4:00 - 5:30 pm** **Room D**	**RISK INFORMED ACTIVITIES - *Status and Direction*** **Session Sub-topics:** - Industry and NRC perspectives on 50.46a - Standards Development - Progress, Benefits and Issues - 50.69 - Lessons Learned - Update on the Training Needs for Risk Analysts **Chair:** Gary M. Holahan, AD for Risk Assessment and New Projects (ADRA)/NRR/NRC **Panelists:** • William E. Burchill, Department Head, Department of Nuclear Engineering, Texas A&M University • Robert Lutz, Fellow Engineer, Risk and Reliability Assessment, Westinghouse Electric Co. LLC • Tony Pietrangelo, Sr. Director of Risk Regulation, NEI • Gerry Sowers, PRA Section Leader, Palo Verde Nuclear Generating Station • Michael D. Tschiltz, DD/DRA/NRR/NRC	**SC:** Michele N. Laur 301-415-3719 email mnl1@nrc.gov

RIC 2006

Program and Schedule

Updated: February 2, 2006 (9:27am)

Wednesday, March 8, 2006

ssion | Time | Room	Session Information	Contact Information
E **Wednesday** **4:00 - 5:30 pm** **Room E**	**ROP** ***Assessment Program Update*** **Session Sub-topics:** - Cross cutting issues, - Significance Determination Process **Chair:** Bruce A. Boger, ADRO/NRR/NRC **Panelists:** • James Andersen, BC, Performance Assessment Branch (IPAB), Division of Inspection and Regional Support (DIRS • Daniel Dorman, Deputy Director, Division of Nuclear Safety, NSIR/ NRR/NRC • Steve Floyd,VP Regulatory Affairs, NEI • Anton Vegel, DD, Division of Reactor Projects, Region IV	**SC:** Lois M. James 301-415-1112 email LMJ@nrc.gov
W4F **Wednesday** **4:00 - 5:30 pm** **Room F**	**CURRENT SEISMIC ISSUES & ASSOCIATED RESEARCH** **Session Sub-topics:** - NRC's Seismic Issues Technical Advisory Group (SITAG) Activities - Update of Probabilistic Seismic Hazard Assessment (PSHA) - Performance-based Approach in Seismic Design Analysis - Tsunami and Its Effects - NRC Perspective on Seismic Issues **Chair:** Richard J. Barrett, Deputy Director, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research (DRAA/RES) **Co-Chair:** Eugene V. Imbro, DD/DE/NRR/NRC **Panelists:** • Goutam Bagchi, Senior Advisor, Division of Engineering, Office of Nuclear Reactor Regulation (DE/NRR) • Lloyd S. Cluff, D, Geosciences Department Pacific Gas and Electric Company • Robert P. Kennedy, President, RPK Structural Mechanics Consulting Inc. • Andrew J. Murphy, Senior Advisor, Engineering Research Applications Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research (ERAB/DET/RES) • Carl J. Stepp, Principal, Earthquake Hazards Solutions	**SC:** Anthony H. Hsia 301-415-6933 email AHH@nrc.gov

RIC 2006
Program and Schedule
Updated: February 2, 2006 (9:27am)

Wednesday, March 8, 2006

Session | Time | Room	Session Information	Contact Information
W4GH Wednesday 4:00 - 5:30 pm Room G & H	**SPENT FUEL MANAGEMENT** **Session Sub-topics:** - Spent Fuel Management - Spent Fuel Transportation - NAS Transportation Study **Chair:** E. William Brach, D/Spent Fuel Project Office (SFPO)/NMSS/NRC **Co-Chair:** Cynthia A. Carpenter, D/PMAS/NRR/NRC **Panelists:** • Kevin Crowley, Executive Director, Nuclear and Radiation Studies Board, National Academy of Sciences • Robert Halstead, High-level Radioactive Waste Transportation Advisor, State of Nevada Agency for Nuclear Projects • Steven Kraft, D/Waste Management/NEI • Gary Lanthrum, D/Office of National Transportation, Office of Civilian Radioactive Waste Management, DOE • John Parkyn, Chairman of the Board and CEO, Private Fuel Storage LLC	**SC:** Steven L. Baggett 301-415-8584 email SLB@nrc.gov

WEDNESDAY RECESS 5:30 pm

Thursday, March 9, 2006

Session | Time | Room	Session Information	Contact Information
BREAKOUT SESSIONS - Set #5: 9:00 - 10:30 am		
Th5BC **Thursday** **9:00 - 10:30 am** **Room B & C**	**MATERIALS DEGRADATION** **Session Sub-topics:** - EPRI's R&D Programs on Materials Degradation in LWRs - A Technical Basis for Revision of the PTS Rule (10CFR50.61) - Overview of Non-Destructive Testing Research - Studies on Irradiation Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in LWR Environments - Materials Research Needs in the Foreseeable Future *"Proactive Materials Degradation Management"* *"A Risk-Informed Re-evaluation of the Pressurized Thermal Shock Rule"* **Chair:** Jennifer L. Uhle, C/Materials Engineering Branch, DET/RES/NRC **Co-Chair:** William H. Bateman, DD, Division of Component Integrity (DCI)/NRR/NRC **Panelists:** • Omesh Chopra, Sr. Metallurgist, Argonne National Laboratory (ANL) • Karen Gott, Coordinator Material and Chemistry Research, Swedish Nuclear Power Inspectorate (SKI) • Robin L. Jones, Technical Executive, Materials Science & Technology, Electric Power Research Institute (EPRI) • Mark T. Kirk, Sr. Materials Engineer, DET/MEB/CIS/RES/NRC • Wallace E. Norris, Sr. Materials Engineer, DET/MEB/AIT/RES/NRC	**SC:** Samantha Crane 301-415-5888 email SXC3@nrc.gov
Th5D **Thursday** **9:00 - 10:30 am** **Room D**	**DIGITAL INSTRUMENTATION & CONTROL** *Diversity and Defense-in-Depth for Digital Systems* **Session Sub-topics:** - Current Regulatory Concerns - International Perspective - US Domestic Industry Perspective and Initiatives **Chair:** Michael E. Mayfield, D/DE/NRR/NRC **Co-Chair:** Michele G. Evans, C/Engineering Research Applications Branch/DET/RES/NRC **Panelists:** • T. Preston Gillespie, Jr., Oconee Nuclear Station • Allen G. Howe, Branch Chief, Instrumentation and Chemicals Branch (EICB)/DE/NRR/NRC • Jukka Laaksonen, Director General, Radiation and Nuclear Safety Authority, Finland • Raymond C. Torok, Sr. Project Manager, Instrumentation and Control, EPRI	**SC:** Paul J. Rebstock Jr. 301-415-3295 email PJR1@nrc.gov

RIC 2006

Program and Schedule

Updated: February 2, 2006 (9:27am)

Thursday, March 9. 2006

Session | Time | Room	Session Information	Contact Information
Th5E **Thursday** **9:00 - 10:30 am** **Room E**	**ROP Inspection Program Update** **Session Sub-topics:** - Inspection Program Realingment - Temporary Instructions - Engineering Inspection Revision - Performance Deficiencies **Chair:** Stuart A. Richards, DD/DIRS/NRR/NRC **Panelists:** • Russell Gibbs, BC, IRIB/DIRS/NRR/NRC • Cynthia Pederson, D, Division of Reactor Safety, RIII/NRC • Tony Pietrangelo, Senior Director, Risk Regulation, Nuclear Energy Institute • Jerry Roberts, Manager - Nuclear Safety Assurance, Cooper Nuclear Station	**SC:** F. Paul Bonnett 301-415-4107 email FPB@nrc.gov
Th5F **Thursday** **9:00 - 10:30 am** **Room F**	**CONSTRUCTION INSPECTION PROGRAM & INSPECTION, TESTS, ANALYSES AND ACCEPTANCE CRITERIA (ITAAC)** **Session Sub-topics:** - ITAAC-CIP Overview - ITAAC Closeout - The role of ITAAC and the QA Program **Chair:** Michael J. Case, D/DIRS/NRR/NRC **Co-Chair:** William D. Beckner, DD/Division of New Reactor Licensing (DNRL)/NRR/NRC **Panelists:** • Mary Ann M. Ashley, Team Leader, Construction Inspection Program, Reactor Inspection Branch, NRR/NRC • Joseph Colaccino, Sr PM, New Reactor Licensing Branch, Division of New Reactor Licensing, NRR/NRC • Ben J. George, Manager Nuclear Licensing, Southern Nuclear Operating Company • Peter S. Hastings, Licensing Manager, Nuclear Projects, Duke Energy	**SC:** John E. Thorp 301-415-6584 email JET3@nrc.gov
Th5GH **Thursday** **9:00 - 10:30 am** **Room G & H**	**SECURITY** ***Security Program Update and Path Forward*** **Session Sub-topic:** Security initiatives and challenges on the short term horizon **Chair:** Cornelius F. Holden Jr., DD/Division of Operating Reactor Licensing (DORL)/NRR/NRC **Co-Chair:** Roy P. Zimmerman, D/NSIR/NRC **Panelists:** • Daniel H. Dorman, Director, Division of Nuclear Security Operations, Office of Nuclear Security and Incident Response, NRC • Morgan Rafferty, Board Member, San Luis Obispo Mothers for Peace • Robert B. Stephan, Assistant Secretary, Office of Infrastructure Protection, U.S. Department of Homeland Security	**SC:** Carol A. Harris 301-415-7368 email CAH4@nrc.gov

BREAK 10:30 - 11:00 am

Program and Schedule

Updated: February 2, 2006 (9:27am)

Thursday, March 9, 2006

Session | Time | Room	Session Information	Contact Information
11:00 am - 12:00 pm Room D & E	**Plenary Session: Presentation | Q&A Session** *Commissioner Peter B. Lyons*	**SC :** Josie M. Piccone 301-415-8430 email: JMP1@nrc.gov
P9 12:00 - 12:30 pm Room D & E	**Plenary Session: Wrap Up** • Jim Dyer, D/NRR/NRC • Carl J. Paperiello, D/RES/NRC	**SC:** Tilda Y. Liu 301-415-1315 email: TYL1@nrc.gov C. E. (Gene) Carpenter 301-415-7333 email CEC@nrc.gov
LUNCH 12:30 - 1:30 pm (NO FORMAL LUNCHEON PLANNED)		
REGIONAL BREAKOUT SESSIONS - Set #6: 1:30 - 4:00 pm		**Regional Breakout HQ Contact:**
RG1BC Thursday 1:30 - 4:00 pm Rooms B & C	**Region I Breakout** **Session Sub-topics:** – NOED Process **Chair:** Samuel J. Collins, RA, RI **Panelists:** • Michael R. Kansler, President, Entergy Nuclear Northeast • Jim Dyer, D/NRR/NRC	**SC:** Kevin A. Mangan 610-337-5335 email KAM1@nrc.gov
RG2D Thursday 1:30 - 4:00 pm Room D	**Region II Breakout** **Session Sub-topics:** – NOED Process **Chair:** William D. Travers, RA, Region II **Panelists:** • Ashok Bhatnagar, Senior VP, Nuclear Operations, Tennessee Valley Authority • Bruce A. Boger, ADRO/NRR/NRC	**SC:** Malcolm T. Widmann 404-562-4550 email MTW@nrc.gov
RG3E Thursday 1:30 - 4:00 pm Room E	**Region III Breakout** **Session Sub-topics:** – NOED Process **Chair:** James L. Caldwell, RA, RIII **Panelists:** • Mano K. Nazar, Senior Vice President, Chief Nuclear Officer, American Electric Power • Brian W. Sheron, Associate Director for Engineering and Safety Systems (ADES)/NRR/NRC	**SC:** Allan R. Barker 630-829-9579 email ARB3@nrc.gov

Program and Schedule

Updated: February 2, 2006 (9:27am)

Thursday, March 9, 2006

Session | Time | Room	Session Information	Contact Information
RG4GH Thursday 1:30 - 4:00 pm Rooms G & H	**Region IV Breakout** **Session Sub-topics:** NOED Process **Chair:** Bruce S. Mallett, RA, RIV **Panelists:** [illegible], Associate Director for Risk Assessment and New [illegible]	SC: David N. Graves 817-8608147 email DNG@nrc.gov

BREAK 4:00 - 4:15 pm

Session | Time | Room	Session Information
P10 **4:15 - 4:45 pm** **Room D**	**Plenary Session: Inter-Regional**

CONFERENCE RECESS 4:45 pm

Evaluation of Human Reliability Analysis (HRA) Methods Against HRA Good Practices (NUREG-1792)

Erasmia Lois (USNRC)
John Forester (SNL)
Alan Kolaczkowski (SAIC)

Presentation to the Advisory Committee on Reactor Safeguards

Rockville, MD
February 9, 2006

Sandia National Laboratories

SAIC
An Employee-Owned Company

Outline

- Background
- Evaluation of methods
  - Approach
  - Summary of results
  - Brief description of each method and some observations
  - Comparison of methods against some key characteristics
  - Implications - What methods should be used when?
- Plans for next steps

Background

- The NRC has developed the "PRA Action Plan for Stabilizing PRA Expectations and Requirements," (SECY-04-0118) to address PRA quality issues
- Guidance for performing/reviewing HRAs is part of the plan
- Guidance is developed in two phases:
  - Phase 1: HRA Good Practices--NUREG-1792, completed
  - Phase 2: Evaluation of methods against the Good Practices, in progress
- Status of methods evaluation
  - Draft report submitted for internal review, including ACRS
  - Address comments from ACRS sub- and full committees and others: February 2006
  - Submit for public comment: March 2006
  - Revise/submit to publication: September 2006

HRA Methods Reviewed

- Technique for Human Error Rate Prediction (THERP) (NUREG/CR-1278)
- Accident Sequence Evaluation Program (ASEP) HRA Procedure (NUREG/CR-4772)
- Human Cognitive Reliability (HCR)/Operator Reliability Experiments (ORE) Method (EPRI TR-100259)
- Cause-Based Decision Tree (CBDT) Method (EPRI TR-100259)
- EPRI HRA Calculator
- Standard Plant Analysis Risk HRA (SPAR-H) Method (NUREG/CR-6883)
- A Technique for Human Event Analysis (ATHEANA) (NUREG-1624, Rev. 1)
- Success Likelihood Index Methodology (SLIM) Multi-Attribute Utility Decomposition (MAUD) (e.g., NUREG/CR-3518)
- Failure Likelihood Index Methodology (FLIM)
- A Revised Systematic Human Action Reliability Procedure (SHARP1, EPRI TR-101711)

Summary of Results

- Most HRA methods are quantification tools for estimating human error probabilities (HEPs)
  - Provide guidance for obtaining HEPs
  - As such are not dealing with the HRA process per se and hence many of the good practices
- A few touch on some aspects of how to do an HRA, but how to do a good HRA is left to analysts
- An exception is ATHEANA, and to some extent THERP, that provide both HRA guidance and a quantification approach
- SHARP and SHARP1 are guidance document on how to do an HRA
- The HRA Calculator is a computerized tool that guides quantification

Summary of Results (cont.)

- All HRA quantification methods have strengths and weaknesses
- Methods reflect an evolution of how to quantify human failure
  - Early methods more simplistically address human behavior
  - Progression of methods reflects efforts to better understand/incorporate advances in behavioral and cognitive science and operational experience
  - Different approaches/capabilities for translating qualitative information into human error probabilities
- Different methods developed for different purposes (detailed versus scoping analysis)
- Some can be applied much easier than others, but at a cost (less breadth and depth of analysis)

Summary of Results (cont.)

- Strengths, e.g.,
  - Some provide clear/good technical basis of underlying model
  - Good step-by-step guidance on how to use the tool
  - Traceable analysis
- Weaknesses, e.g.,
  - Weakness in technical basis make the use of some methods questionable
  - Some address only a limited set of performance shaping factors (PSFs) and context (plant conditions)
  - Methods not always applied as intended

Summary of Results (cont.)

- Overall perspective: Methods can be viewed as providing a "tool box" :
  - Some provide a tool for detailed analyses; others for screening analyses
- Using the right method for the right application is very important
- Therefore, we should use those methods that provide the best capabilities for the application
- Should use methods as they are intended to be used
- Drop any method(s) found to have unjustified technical basis

Technique for Human Error Rate Prediction (THERP)
(NUREG/CR-1278)

- THERP is a method for identifying, modeling and quantifying human failure events (HFEs) in a PRA.
  - How to incorporate into PRA not covered
  - Emphasis on decomposing operator tasks into subtasks
- THERP has probably been used more than any other HRA technique
- Guidance for quantification of pre- and post-initiator HFEs
- Diagnosis contribution to error is handled with time reliability curves (TRCs) that provide no insights. Response execution HEP is added on.
- Basic HEP adjusted by PSFs
- Only a relatively small subset of PSFs actually addressed in quantifying HEPs (how to handle other PSFs left to analyst)
- Few HEPs and quantitative factors have an empirical basis

Accident Sequence Evaluation Program (ASEP) HRA Procedure (NUREG/CR-4772)

- A quantification technique for pre- and post-initiator human failure events
- Provides both screening and nominal human error probabilities for both pre- and post-initiators
- Otherwise, a simplified version of THERP meant to produce more conservative HEPs, but useable by PRA analysts with limited HRA background
- Basic HEP adjusted by PSFs
- Only a relatively small subset of PSFs actually addressed in quantifying HEPs (how to handle other PSFs left to analyst)

Human Cognitive Reliability (HCR)/Operator Reliability Experiments (ORE) Method (EPRI-TR100259)

- EPRI developed quantification technique for estimating non-response probability of post-initiator actions only
- Simulator measurement-based, time/reliability correlation (TRC) for diagnosis portion of human action
  - Does not explicitly address potential causes of human errors in diagnosis
- Needs relatively significant number of simulator exercises to produce reasonable results
- Evidence supporting use of the lognormal distribution, and thereby the standard normal distribution tables for obtaining non-response probability, is not available for public scrutiny
- Addresses both screening and nominal HEPs
- Includes Cause-Based Decision Tree (CBDT) method for longer time-frame events

Cause-Based Decision Tree (CBDT) Method
(EPRI-TR100259)

- Originally developed by EPRI to:
  - Address when HCR/ORE produces very low probability values
  - Address actions with longer time frames where "extrapolation of HCR/ORE TRC could be extremely optimistic"
- Quantification technique for estimating non-response probability of post-initiator human actions only
  - Causal approach allows consideration of 8 potential error mechanisms and factors that could contribute to those failures (diagnosis is assessed) through use of decision trees
- In more recent years, the CBDT method has frequently come to be used as a "stand alone" method
- No guidance for use under time-limited conditions
- Quantification data extrapolated from THERP, based on expert judgment

Standard Plant Analysis Risk HRA (SPAR-H) Method (NUREG/CR-6883)

- A quantification technique addressing both diagnosis and execution aspects human events
- Can be used for pre- and post-initiator events
  - SPAR-H does not use that classification nor distinguish
- Designed to provide reasonable estimates for regulatory uses
  - Accident sequence precursor program (ASP)
  - Phase 3 of the Significance Determination Process (SDP)
- Assumes basic HEP, adjusted to reflect ~8 PSFs
  - Nominal value for some PSFs usually assumed for control room actions
  - HEPs based on extrapolation of THERP and comparison with other methods
- Resolution of PSFs not appropriate for detailed HRA analysis (without expert judgment on part of the analyst)

A Technique for Human Event Analysis (ATHEANA)

(NUREG-1624 Rev. 1 & *Reliability Engineering & System Safety*, *83*: 207-220, 2004 Article on Quantification)

- Identification, modeling, and quantification of post-initiator human actions, including treatment of errors of commission
  - Concepts applicable to pre-initiators, but little specific guidance provided
- Addresses potential cognitive and implementation failures for a human action and the situations that could cause them
  - Strives to address a wide range of scenario and performance conditions (context) and unsafe actions
  - Intent is to address both nominal and deviation scenarios (i.e., not just "near-average" PRA context)
- Formal, facilitator-led expert elicitation process for quantification
- Guidance for addressing broad range of factors relevant to the <u>nominal case</u> needs to be strengthened (emphasis is on error forcing context)
- Detailed context development to determine the most appropriate influencing factors can be complicated and time and resource intensive
- If deviation scenarios need to be identified, analysis will take additional time

Success Likelihood Index Methodology (SLIM) Multi-Attribute Utility Decomposition (MAUD) and Failure Likelihood Index Methodology (FLIM)

- Quantification methods with a primary focus on post-initiator diagnosis failures
- Assumes that relative importance weights and ratings of PSFs, obtained from expert judges and related to a task, can be multiplied together and then summed across PSFs to arrive at the Success Likelihood Index (SLI).
  - FLIM (developed by PLG) is similar but provides scaling guidance for a suggested 7 PSFs (in some applications more)
- Requires events with known HEPs as calibration events (anchor values), and an assumption of a logarithmic-linear relation between the desired HEP and the SLI
- Identifying appropriate calibration data can be problematic
- Questions exist regarding the appropriateness of the linear model to reflect the experts' judgments
- Software tool for SLIM/MAUD not available

EPRI HRA Calculator

- Software tool – not a method
- Automates HCR/ORE, CBDT, THERP annunciator response model to address diagnosis of post-initiator HFEs
  - No guidance for which method to use
  - Includes aspects of SPAR-H for comparison purposes
- Uses THERP for response execution portion
- Uses THERP and ASEP to quantify pre-initiator HFEs
- Relies on SHARP1 as the HRA framework
- Not all PSFs discussed/addressed appear to be handled within the software quantification (this is being improved)
- Limited flexibility to address other PSFs (focus on standardization)

A Revised Systematic Human Action Reliability Procedure (SHARP1, EPRI TR-101711)

- SHARP1 is a guidance document for performing many aspects of an HRA in the context of a PRA (including identification and modeling issues)
- Covers both pre- and post-initiator human actions
- While it does not provide a quantification method for either, it does provide a summary of quantification methods available at the time.
- Generally consistent with the ASME standard for performing an HRA and with the NRC's HRA good practices guidance
- Insufficient guidance on identification of PSFs and context and on the consideration of errors of commission

Comparison of Methods Against Some Key Characteristics

- Discuss important (selected) HRA characteristics
- Address how the different methods cover those characteristics
  - Characteristics of quantification process

Overall Quantification Approach

- Uses concept of a basic/initial HEP that is subsequently adjusted and/or set tables, curves (generally limited and fixed set of PSFs) – THERP, ASEP, CBDT, SPAR-H
- Estimates HEP directly based on context & experience/judgment – SLIM/FLIM, ATHEANA
- Based on empirical or judged measures of timing for actions – HCR/ORE

Addresses Dependencies

- Has a model to address dependencies – THERP (among subtasks), ASEP uses simplified version of THERP, SPAR-H and sometimes FLIM uses THERP
  - Generic model requiring expert judgment to assess the level of dependence
- Discussed and to be considered as part of the context and included in the estimated HEP for given HFE – ATHEANA and to some extent SLIM/FLIM
- Discussed, but specific quantitative estimates not proposed. Effect on quantification left to the analysts – HCR/ORE, CBDT.
- SHARP1 provides overall good discussion, but does not address quantification of dependencies

Range of Contexts Considered

Mainly the plant- related characteristics (plant conditions) that might vary for a given PRA scenario

- Largely an expected context based on PRA definition of scenario - THERP, ASEP, CBDT, SPAR-H, SLIM/FLIM – Depends on analyst to some extent
- Investigates nominal and related but different contexts (including so-called deviation scenarios) that all fit within the PRA definition of scenario ATHEANA
- Context not explicitly addressed other than as represented in the simulator runs (when used) Range of contexts requires many simulator runs - HCR/ORE.

Range of Specific PSFs Considered

- Most methods cover a relatively small range of PSFs and commonality is less than might be expected.
- THERP – For diagnosis, discusses a wide range of PSFs, but model addresses only a few - time available, event specific training, task load, redundant signals, stress, experience.
- SLIM and ATHEANA do not specify a fixed set - ATHEANA provides range of examples
- Only ATHEANA (and SLIM if modified) considers potential interactions between PSFs

Implications for Use of HRA Methods
(What methods should be used when?)

It all depends on the issue and decision being made

- HRA process
  - When issue/decision clearly affects just one or very few already identified HFEs with no need to worry about dependencies nor interactions with the rest of the PRA, then detailed identification and modeling processes etc. are not important
  - When issue/decision affects multiple HFEs or requires interactions with the rest of the PRA to be accurate (e.g., need to account for dependencies and the correct component rankings), then following the HRA process correctly becomes more important

Implications for Use of HRA Methods

(What methods should be used when?) (cont.)

It all depends on the issue and decision being made (cont.)

- <u>HRA quantification and qualitative analysis of HFE</u>
  - When the risk-related decision being made is not very sensitive to the specific qualitative and/or quantitative results from the method because, for instance, screening analysis or sensitivity studies show that the conclusions do not change, or
  - When level of PRA analysis (extent of PRA conditions being considered) is not intended to include detailed HRA considerations (e.g., ASP analyses), or
  - When, based on prior experience, seems likely that the most important influencing factors affecting the human action of interest are easily and directly handled using a less detailed, easier to use method,

  then simpler quantification methods may be used.

Implications for Use of HRA Methods

(What methods should be used when?) (cont.)

- Simpler quantification methods can provide helpful answers to the decision process as long as:
  - the primary weaknesses of a method are avoided
  - the method is not asked to give answers it cannot provide, for example,
    - determine causal influences to a diagnosis error using a simple TRC
    - assess the potential effects of communications when "communications" is not addressed directly by the method or easily interpreted as part of another factor that is covered by the method.

Implications for Use of HRA Methods
(What methods should be used when?) (cont.)

It all depends on the issue and decision being made (cont.)

- HRA quantification and qualitative analysis of HFE
  - The more the decision requires the "best" answer we can provide because the decision is very sensitive to the probabilistic inputs and the associated results from the HRA, the more important it is that the HRA process be rigorously followed and that a more detailed, broader scope quantification method needs to be used.
    - Whatever quantification method is used, it needs to be justified as to why it is appropriate for the decision being made
  - If one needs, for example,
    - A reasonably accurate estimate of the HEP - whether the probability of failure is high or low
    - To understand what the drivers for success/failure are and what conditions could create problems for the crew (so as to identify fixes),

    Then a detailed analysis that considers a reasonably broad range of conditions is needed

Implications for Use of HRA Methods

(What methods should be used when?) (cont.)

- Analysts/reviewers/users should avoid selecting a method first and then making the decision/issue *fit* the method
- The HRA process should be the other way around
  - Determine what is needed from the HRA to address the decision/issue
  - Select the appropriate method(s) accordingly AND justify the selection as well as the assumptions and judgments made in implementing the method(s)
  - Perform sensitivities to make results even more robust

Next Steps

- Continue improvement of methods thru reviews of non USA methods and interactions with their developers
- Strive for convergence in HRA technology
  - Develop common frameworks—work with domestic and international experts/practitioners
  - Address the ISPRA study results
- Improve the technical bases of selected NRC methods
  - Test/compare methods thru simulator experiments and other means
- Improve quantification capability and validate with experience where possible
- Expand knowledge base as needed

Technical Discussion of SRP 14.2.1 – "Generic Guidelines for Extended Power Uprate Testing Programs"

ACRS Committee Presentation

Paul Prescott

Robert Pettis

Office of Nuclear Reactor Regulation
Division of Engineering
Quality and Vendor Branch A and B

February 9, 2006

3

Presentation Overview

- Purpose is to provide ACRS:
  - Recent changes to SRP 14.2.1.
  - Staff evaluation of EPUs using SRP 14.2.1.
  - Overview and technical discussion of SRP Section III.C., "Justification for Elimination of EPU Power Ascension Tests."

Recent Changes to SRP 14.2.1

- Most changes to SRP 14.2.1 since last ACRS review were editorial in nature.

- Most significant change was addition of paragraph to clarify Section III.C.c., "Facility Conformance to Limitations Associated with Computer Modeling and Analytical Methods."
  - Ensure setpoint and parameter changes and modifications do not invalidate analytical methods.
  - If analytical methods are inadequate, the secondary review branches may need transient testing performed to make its final safety conclusion.

Staff Evaluations of EPUs Using SRP 14.2.1

- Staff ensures EPU license amendment request fully addresses SRP Section 14.2.1.

    - Initial test program review
    - Plant modifications
    - Power ascension test elimination justifications
    - Proposed EPU testing plans

Overview and Technical discussion of SRP 14.2.1 Section III.C.

- Staff guidance acknowledges that licensees may propose justification for not performing certain testing. Supplemental guidance provided in SRP Section III.C. for staff evaluation of justification.

Overview and Technical discussion of SRP 14.2.1 Section III.C.

- Some of the factors considered by the staff under Section III.C.
  - Operating experience
  - Thermal-hydraulic phenomena or system interactions
  - Computer modeling
  - Plant operations and use of procedures

Overview and Technical discussion of SRP 14.2.1 Section III.C.

- Staff has previously accepted the following justifications for not performing LTTs for EPUs.
  - Licensees' test program will monitor important plant parameters during EPU power ascension.
  - Tech Spec surveillance and post-mod testing will confirm the performance capability of the modified components.
  - Operating history at the facility and experience at other LWRs.
  - LTTs not needed for Code analyses benchmarking.

Overview and Technical discussion of SRP 14.2.1 Section III.C.

- One plant has a proposed license condition requiring transient testing of the condenstate/feedwater system to confirm consistency with analytical results.

- One plant has proposed a manual turbine trip from 30% power in its EPU test program.

(4)

FERRET

A Least Squares Best Estimate Evaluation

for

Reactor Vessel Dosimetry

Lambros Lois
NRR/DSS/SBWB
Feb. 9, 2006

This Presentation Will Discuss:

- GDC 30; Requirements
- Calculated and Measured Fluence Values
- The “Old” FERRET
- Questionable Applications
- FERRET Review
- The “New” FERRET
- Conclusions

GDC 30, Requirements

- The RCPB shall be designed.... to behave in a non-brittle manner with consideration for the uncertainties in determining the material properties.

- To assure that the RCPB behaves in a non-brittle manner through the operating period of the plant the staff calculates projected fluence values to the end of the operating licence.

- In addition to material embrittlement, fluence is used to predict IASCC, material weldability, and PT limits.

Calculated and Measured Fluence

- Fluence values are required on the inside surface of the PV and 1/4T and 3/4T of the vessel thickness.
- Fluence cannot be measured directly in the locations of interest.
- Instead, we measure dosimeter foil activation and convert to neutron flux using a calculated neutron spectrum at the location of irradiation.
- Appendix H to 10 CFR 50 provides for surveillance capsules to be removed at specified intervals.
- Calculated to Measured (C/M) dosimeter value ratios were significantly different from 1.0.

Questionable Applications

- The staff was presented with Dosimetry Adjustments that did not seem correct

- Using Cu, Ti, Ni, Fe and U-238 dosimeters Fe indicated the lowest value. The adjustment was essentially on the value for Fe.

- Reports of Vessel Dosimetry Measurements indicated large discrepancies of the C/M ratios by as much as 30% to 40%.

- C/M values are outside the expected range of 4% to 5% uncertainty.

- Staff calculations indicated that the dosimeter location may not be correct.

FERRET Review

- The staff requested that FERRET be submitted for staff review, in between

- The staff refused to accept licensing submittals that used FERRET best estimate fluence values

- In 2005 Westinghouse submitted for review its first version of FERRET

- Acceptance Review, Indicated that the large discrepancies were not treated nor recognized as a problem. The staff responded that: "... is reluctant to initiate review of this report, that appears technically correct in its least squares method but is seriously flawed in its physics ..."

- Westinghouse issued a supplement to the original report.

UNITED STATES NUCLEAR REGULATORY COMMISSION

The "New" FERRET

- Least squares spectral based adjustment
- There is no covariant matrix
- The report includes results of about 70 capsules with uncertainties in the range of 5%
- The staff approval of FERRET includes a limitation that limits its applicability to cases with uncertainties within the bounds of the data base

Conclusions

- Reactor vessel dosimetry from Westinghouse was poor
- The staff accepts fluence values either calculated using an approved and benchmarked code or a FERRET adjusted value within the limits of the data base
- The staff does not have any information how the data base was corrected or revised

Least Squares Application to Spectrum Adjustment

$$R_i \pm \delta_{Ri} = \Sigma_g (\sigma_{ig} \pm \delta_{\sigma\, ig})(\phi_g \pm \delta_{\phi g})$$

- Measured reaction rates and associated uncertainty for each sensor in the measurement dosimeter set
- Energy dependent dosimeter reaction cross section and associated uncertainties for each sensor in the measurement dosimeter set
- Calculated neutron energy spectrum and associated uncertainties at the measurement location

Note: RG 1.190, includes guidance for the calculation of $\phi_g \pm \delta_{\phi g}$ and acceptable sources for $\sigma_{ig} \pm \delta_{\sigma\, ig}$.

Application of the FERRET method to the Westinghouse data base, reduces the standard deviation from 13% to 7%.

- DOT $\Rightarrow$ Discrete Ordinates Transport
- RAMA $\Rightarrow$ Radiation Analysis Modeling Application
- FERRET $\Rightarrow$ Not an Acronym

(5)

G:PPHandout

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
529	10	10.1

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

Authors:
JOHN T. LARKINS

List of Documents Attached	
PLANNING & PROCEDURES MINUTES	**10**

Instructions to Preparer	From Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	JOHN T. LARKINS

February 9, 2006 (1:10pm)
G:\PlanPro(ACRS)\ppmins.529.wpd

INTERNAL USE ONLY

SUMMARY/MINUTES OF THE ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING February 8, 2006

The ACRS Subcommittee on Planning and Procedures held a meeting on February 8, 2006, in Room T2B-3, 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 10:00 a.m. and adjourned at 11:00 a.m.

ATTENDEES
G. Wallis
W. Shack
J. Sieber

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

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

Member assignments and priorities for ACRS reports and letters for the February ACRS meeting are attached (pp. 5-6). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the February ACRS meeting be as shown in the attachment (pp. 5-6).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through April 2006 is attached (pp. 7-8). The objectives are to:

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

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action (pp. 9-12).

RECOMMENDATION

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

3) Response to the Staff Requirements Memorandum (SRM)

In the December 20, 2005 SRM (pp. 13), resulting from the ACRS meeting with the NRC Commissioners on December 8, 2005, the Commission requested the following:

a) Following its retreat in January 2006, the ACRS should inform the Commission how the Committee plans to manage the increased workload resulting from the anticipated receipt of new reactor designs and combined license (COL) applications.

b) The ACRS shall make among its highest priorities its role in the resolution of GSI-191 [The staff shall expedite efforts to provide the ACRS with information necessary to make its assessments and recommendations]

Regarding Item (a), during its January 26-27, 2006 Planning and Procedures Subcommittee meeting, the members discussed a plan proposed by the ACRS staff for handling anticipated heavy workload in the areas of advanced reactors and COLs. A draft response to the Commission SRM on this matter will be provided to the Committee for discussion and endorsement during the March 2006 ACRS meeting. The due date for responding to the Commission SRM is March 31, 2006.

Regarding Item (b), the ACRS Subcommittee on Thermal-Hydraulic Phenomena is scheduled to hold a meeting on February 14-16, 2006 to discuss interim results of the chemical effects tests and industry responses to the Generic Letter on PWR sumps. This matter is also scheduled for full Committee discussion during the March 2006 ACRS meeting. The Subcommittee and the full Committee will continue to discuss issues related to PWR sump performance as further progress has been made by the staff.

RECOMMENDATION

The Subcommittee recommends the following:

- The ACRS staff should provide a draft response to the Commission SRM with regard to ACRS plans for handling the anticipated increased workload in the areas of advanced reactors and COLs for discussion by the Planning and Procedures Subcommittee and the full Committee.

- Subsequent to its February 14-16, 2006 meeting, the Thermal-Hydraulic Phenomena Subcommittee should develop a draft report documenting its concerns for consideration by the full Committee during its March 2006 meeting.

4) Letter from Mr. Paul Blanch Regarding Vermont Yankee Extended Power Uprate

Mr. Paul Blanch, Energy Consultant, sent a letter (pp. 14-20) to the ACRS Chairman on January 20, 2006 documenting his views about the ACRS review of the Vermont Yankee extended power uprate. He expresses concern about whether Vermont Yankee will meet all applicable regulatory requirements at the extended power uprate conditions. He requests that the ACRS provide a statement, supported by objective evidence, that Vermont Yankee will be operated in compliance with applicable regulatory requirements thus assuring public safety. Dr. Denning, Chairman of the ACRS Subcommittee on Power Uprates, and Dr. Wallis prepared a draft response (pp. 21) to Mr. Paul Blanch for consideration by the Committee.

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the proposed response to Mr. Paul Blanch.

5) ACNW Meeting on Radiation Protection Program

The ACNW is scheduled to hear presentations by and hold discussions with representatives of the Office of Nuclear Regulatory Research (RES) regarding the Radiation Protection Program during the April 18-19, 2006 ACNW meeting. Exact timing of this session will be provided later. ACNW invites interested ACRS members to participate in this session.

RECOMMENDATION

The Subcommittee recommends that those ACRS members who are interested in attending the ACNW meeting session involving discussion of the RES Radiation Protection Program inform the ACRS Executive Director.

6) Actions, Agreements, Commitments, and Follow-up Items Resulting from the ACRS Retreat

An expanded meeting (Retreat) of the ACRS Subcommittee on Planning and Procedures was held on January 26-27, 2006 to discuss various issues.

A summary of the actions, agreements, commitments, and follow-up items resulting from this meeting are included in the attachment (pp. 22-25).

RECOMMENDATION

The Subcommittee recommends that the Committee provide feedback and endorse the actions, agreements, commitments, and follow-up items listed in the attachment.

7) ACRS Conference Room Upgrade

Arrangements are being made to upgrade the ACRS conference room audiovisual equipment. The upgrade will begin on March 13, 2006 and is expected to be completed on or before April 24, 2006. We are making arrangements to hold the Subcommittee meetings and the April 6-8, 2006 full Committee meeting in different locations, including the Commissioners' conference room.

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director and Jenny Gallo keep the Committee informed of the location of the ACRS meetings.

8) Interview of Candidates to Fill the Vacancy on the Committee

During the March ACRS meeting, members and the ACRS Member Candidate Screening Panel will interview best-qualified candidates for membership on the ACRS. An interview schedule along with the resumes of the candidates to be interviewed will be provided to the members prior to the March ACRS meeting.

In addition, a draft Federal Register Notice and Press Release seeking candidates with expertise in various disciplines, have been sent to the Commission for approval for publication. This will help maintain a pool of candidates with expertise in various areas and will also minimize the time required to fill future vacancies on the Committee.

9) Member Issue

Dr. Kress has prepared a draft report to the Commission on Risk-informed Criteria for Acceptability of Power Uprates from a Site Suitability Perspective for consideration by the full Committee during the February 9-11, 2006 ACRS meeting. Since this item is not announced in the Federal Register notice for the February meeting, the committee cannot discuss this matter at the February meeting. Some members have already provided comments to Dr. Kress.

RECOMMENDATION

The Subcommittee recommends that Dr. Kress' report be scheduled for discussion during a future ACRS meeting and that the members continue to provide feedback to Dr. Kress following the February ACRS meeting. After discussion at a future meeting, if the Committee does not endorse sending a report of this nature, Dr. Kress may choose to send his report as a member of the ACRS, in accordance with the ACRS Bylaws.

ANTICIPATED WORKLOAD
FEBRUARY 9-11, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Thornsbury	Evaluation of HRA Methods against Good Practices in NUREG-1792	A	To support staff schedule	**Draft**
Bonaca	—	Flack	**SUBCOMMITTEE REPORT** - Safety Conscious Work Environment/ Safety Culture **[SUBC. Mtg 01/25/06]**	—	—	—
Denning	—	Lamb	Proposed Revisions to SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs"	A	To support staff schedule	**Draft**
	—	Santos	FERRET Reactor Vessel Fluence Methodology **[INFORMATION BRIEFING]**	—	—	—
Powers	All Members	Nourbakhsh/ Duraiswamy	Final ACRS Report to the Commission on the NRC Safety Research Program	A	To respond to SRM. Due date March 15, 2006	**Draft**
Ransom	Wallis	Caruso	**SUBCOMMITTEE REPORT** - Proposed Revision 4 to Reg. Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a LOCA" and Application of TRACG Code to ESBWR Stability **[SUBC. Mtg. 01/19/06]**	—	—	—
Wallis	—	Larkins/Thadani/ Flack	Response to the December 20, 2005, SRM Regarding ACRS plans to manage the Anticipated Increased Workload in the areas of Advanced Reactor Designs and COLs	P & P Subc.	To respond to SRM, Due Date March 31, 2006	**Draft**

(51)

ANTICIPATED WORKLOAD
FEBRUARY 9-11, 2006 (Cont'd)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Shack	—	Snodderly	**SUBCOMMITTEE REPORT** - Draft Reg. Guide, "An Approach for Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" **[SUBC. Mtg 01/25/06]**	—	—	—
Sieber	Bonaca	Lamb	**SUBCOMMITTEE REPORT** - Interim Review of the Brunswick License Renewal Application **[SUBC. Mtg. 2/8/06]**	—	—	—

6

ANTICIPATED WORKLOAD
MARCH 9-11, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	—	Santos	Final Review of the License Renewal Application and the Final SER for Browns Ferry Units 1, 2, and 3	A	To support staff schedule	—
Powers	—	Taylor/Snodderly	Final Review of the Clinton Early Site Permit Application and the Final SER	A	To support staff schedule	—
		Nourbakhsh/ Duraiswamy	Final ACRS Report to the Commission on the NRC Safety Research Program **[IF NOT COMPLETED IN FEBRUARY]**	A	To respond to SRM. Due date March 15, 2006	—
Sieber	—	Lamb	Draft Final Revision 4 to Reg. Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"	A	To support staff schedule	—
	—	Lamb	Evaluation of Precursor Data to Identify Significant Operating Events **[INFORMATION BRIEFING]**	—	—	—
Wallis	—	Caruso	Chemical Effects Test Results/ Industry Responses to the Generic Letter on PWR Sumps	A	To provide Committee's views	—
	—	Larkins/Thadani/ Flack	Response to the December 20, 2005, SRM Regarding ACRS plans to manage the Anticipated Increased Workload in the areas of Advanced Reactor Designs and COLs **[IF NOT COMPLETED IN FEBRUARY]**	P&P Subc.	To respond to SRM, Due Date March 31, 2006	—

7

ANTICIPATED WORKLOAD
APRIL 6-8, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	Denning	Lamb	Draft Final Reg. Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants"	A	To support staff schedule	—
Armijo	Shack	Santos	Review of 1994 Addenda for Class1, 2, and 3 Piping Systems to the ASME Code Section III and the Resolution of the Differences Between the Staff and ASME **[TENTATIVE]**	A	To provide Committee's views	—
Bonaca		Thornsbury	Safeguards and Security Matters **[CLOSED] [TENTATIVE]**	A	To provide Committee's views	—
		Flack	Safety Conscious Work Environment/ Safety Culture	A	To provide Committee's views	—
	—	Santos	Draft Final Generic Letter 2005-xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems"	A	To support staff schedule	—
Denning	—	Caruso	**SUBCOMMITTEE REPORT** - Power Uprate Application for Ginna Nuclear Plant and the Associated Safety Evaluation - **SUBC. Mtg. 2/14-15/06**	—	—	—
Wallis	—	Caruso	Application of TRACG Code for ESBWR Stability	A	To support staff schedule	—

8

Items Requiring Committee Action

1 **<u>Draft Regulatory Guides on ASME Code Cases</u>**

Member: William Shack **Engineer:** Cayetano Santos

Estimated Time:

Purpose: Determine a Course of Action

Priority:

Requested by: RES W. Norris

The staff has prepared draft revisions to the following Regulatory Guides regarding ASME code cases:

(1) Proposed Revision 34 of Regulatory Guide 1.84 (DG-1133), Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Division 1

(2) Proposed Revision 15 of Regulatory Guide 1.147 (DG-1134), Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1

(3) Proposed Revision 2 of Regulatory Guide 1.193 (DG-1135), ASME Code Cases Not Approved for Use

The proposed revisions to Regualtory Guides 1.84 and 1.147 are being incorporated by reference to 10 CFR 50.55a. In a January 10, 2006 memorandum from Mark Cunningham to John Larkins, the staff requested ACRS review of these draft regulatory guides. The staff has provided these draft guides in advanced of the proposed rule since the ACRS has typically chosen not to review these regulatory guides at the draft stage.

The Planning and Procedures Subcommittee recommends that Dr. Shack propose a course of action.

2 Proposed Recommendations by RES to Resolve GSI-188, "Steam Generator Tube Leaks/Ruptures Concurrent With Containment Bypass" (Open)

Member: John Sieber **Engineer:** Cayetano Santos

Estimated Time:

Purpose: Determine a Course of Action

Priority:

Requested by: RES T. Mintz

The principle assertion of GSI 188 is that dynamic loads from secondary side breaks could affect the integrity of degraded steam generator tubes and result in increased steam generator tube leakage. Task 3.1 of the Steam Generator Action Plan (SGAP) was added as a result of an ACRS recommendation made in NUREG-1740 (Voltage-Based Alternative Repair Criteria). This task also outlined the tasks needed to resolve the assertion of GSI 188. A May 21, 2004 committee report on the SGAP concluded that "the analyses of the effects of depressurization during a MSLB on tube integrity have been completed and item 3.1 is appropriately closed out."

A memorandum dated December 16, 2005, from Carl. Paperiello, Director Office of Nuclear Regulatory Research, to Luis Reyes, Executive Director for Operations, describes the resolution of GSI 188. The staff concluded that the dynamic loads from secondary side breaches do not cause additional steam generator tube leakage or ruptures beyond what would be determined using differential pressure alone. The staff recommended that no changes be made to the regulations or guidance associated with dynamic loads from main steamline or feedwater line breaks. The staff also concluded that the dynamic loads associated with these breaks do not need to be considered in evaluating the potential for multiple tube ruptures in GSI-163 (Multiple Steam Generator Tube Leakage).

The staff has prepared a draft NUREG report describing the technical assessment of GSI 188. This report contains the following statement:

"The ACRS agreed that 'the analyses of the effects of depressurization during an MSLB on tube integrity have been completed, and item 3.1 is appropriately closed out.' Therefore, the ACRS supports the close-out of the principal assertion of GSI-188."

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

3 **Draft Final Regulatory Guide 1127 (RG 1.92 Rev. 1),"Combining Modal Responses and Spatial Components in Seismic Response Analysis"** (Open)

Member: Dana Powers **Engineer:** Michael Snodderly

Estimated Time: 1.5 hours

Purpose: Determine a Course of Action

Priority: High

Requested by: RES T.Y. Chang, RES

DG-1127 will update Revision 1 of RG 1.92 which provides guidance concerning the seismic analysis and design of nuclear power plant structures, systems, components. DG-1127 incorporates improved guidance on the use of Gupta's method for combining modal responses. The NRC staff requested the Committee to defer its review of DG-1127 until after the public comments have been received and analyzed by the staff. The Committee issued a Larkinsgram on November 8, 2004 that agreed with the staff's recommendation and requested an opportunity to review the draft final version of this Guide after reconciliation of public comments.

The draft guide was issued for public comment on February 15, 2005. The staff received responses from four commenters. In a memorandum dated January 13, 2006, the staff provided copies of the draft final regulatory guide and the staff's responses to the public comments for ACRS review.

Bill Hinze, ACNW, reviewed the draft final regulatory guide and concluded that what the staff has done is appropriate.

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

4 **Anonymous Letter concerning the Incorrect EOS Solution in the TRACE code** (Open)

Member: Graham Wallis **Engineer:** Ralph Caruso

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: RES

On January 10, 2005, Dr. Wallis and Dr. Ransom received a new anonymous letter concerning the technical adequacy of the Equation of State (EOS) solution in the TRACE computer code. This is the third anonymous communication related to TRACE that has been sent to the ACRS. Dr. Wallis is requested review this letter and to recommend whether it should be forwarded to the staff for its consideration, as were the earlier communications.

Home > Electronic Reading Room > Document Collections > Commission Documents > Meeting Slides, Transcripts, and Meeting SRMs > 2005 > Meeting SRM M051208A

IN RESPONSE, PLEASE
REFER TO: M051208A

December 20, 2005

MEMORANDUM FOR: John T. Larkins
Executive Director, ACRS

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

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, 1:00 P.M., THURSDAY, DECEMBER 8, 2005, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards (ACRS) to discuss the Committee's activities and current focus. Following its retreat in January 2006, the ACRS should inform the Commission how the Committee plans to manage the increased workload resulting from the anticipated receipt of new reactor designs and combined license applications. The ACRS shall make among its highest priorities its role in the resolution of GSI-191. The staff shall expedite efforts to provide the ACRS with information necessary to make its assessment and recommendations. The Commission continues to value the independent technical views of the ACRS on significant matters under consideration by the agency.

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

13

Paul M. Blanch
Energy Consultant

January 20, 2006

The Honorable Graham B. Wallis
Chairman ACRS
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Wallis:

It was a pleasure having the opportunity to testify before the ACRS on November 16, 2005 in Brattleboro Vermont and to present my concerns related to the proposed power uprate of the Vermont Yankee Nuclear plant.

It was my understanding and the understanding of many other participants that the purpose of meeting in Vermont was to listen to any potential safety concerns and to respond to these concerns either during the meeting or in follow-up communication.

It is my strongest belief that the ACRS has the responsibility to the general public to review and report on safety studies and reactor facility license and license renewal applications and advise the Commission on the hazards of proposed and existing reactor facilities and the adequacy of proposed reactor safety standards.

I am enclosing a copy of my statements before your committee. My overriding concern is that the only means to assure public safety and the hazards of proposed and existing reactor facilities is to determine if the facility will be operated within the NRC's regulatory requirements. My request to the ACRS is to determine if the power uprate will assure adequate protection to the general public by determining compliance with the applicable regulatory requirements including the basic safety building blocks commonly referred to as the General Design Criteria (10 CFR 50, Appendix A). It is not material that the applicable GDC's be either the draft or the final GDC's but compliance with one or the other must be identified and verified.

In previous communication[1] from the ACRS to the Commission evaluating the safety aspects of the proposed power uprate for Dresden and Quad Cities plants, the ACRS made it very clear that the plants would be operated safely by stating "The staff has determined that the proposed EPUs meet all regulatory criteria." My sole purpose of traveling to Vermont and presenting my position to the ACRS was to obtain assurance that a similar determination by the ACRS is made assuring the safety of the nearby population.

---

[1] ACRS letter dated December 12, 2001

Your letter[2] to the Chairman recommended approval of the Extended Power Uprate however failed to provide assurance that the plant would be operated in compliance with the NRC's regulations.

I would greatly appreciate it if the ACRS responds to my concerns identified in my testimony before the ACRS and provide a statement to the public, supported by objective evidence, that the plant will be operated in compliance with applicable[3] regulatory requirements thus assuring public safety.

Paul M. Blanch

Paul M. Blanch
135 Hyde Rd.
West Hartford, CT 06117
860-236-0326

Cc: Mr. George Mulley
Office of the Inspector General

[2] ACRS letter dated January 4, 2006
[3] The applicable regulatory requirements should be identified by the ACRS and /or the NRC Staff

MR. BLANCH: Thank you, Mr. Chairman. Thank you, ACRS members and members of the public, to take time out to listen to this long session today and yesterday.

Again, my name is Paul Blanch. I reside in West Hartford, Connecticut. And I have about 40 years of nuclear experience, both with the utilities and with the Navy.

As far as this proceeding goes, I have no political or financial interests. And I am not being compensated whatsoever for any of my efforts related to the Vermont Yankee efforts.

Our first speaker yesterday was a former governor of Vermont. And he stated that the EPU should be approved "if all regulatory requirements are met." I know I'm going to get at this point some of the members of the public, but I don't disagree with that statement "if all regulatory requirements are met."

I have been concerned about the EPU primarily related to the containment overpressure and the interdependence of the barriers, meaning the failure of one barrier could result in the possible failure of another barrier.

I was very troubled and very surprised by Mr. Hobbs' statement this morning that there already is an interdependence of the barriers. He clearly stated -- and I believe I heard this correctly -- that the failure of the Torus -- and I assume he is talking about a catastrophic failure of the Torus -- will result in core damage in disabling some of the safety instruments, which would result in -- well, the failure of the Torus would result in failure of the ECCS, which would result in the failure of the fuel or fuel meltdown.

Now, either Mr. Hobbs does not understand the design basis of Vermont Yankee -- and he is the engineering supervisor. And I believe that that event -- I could be wrong, but catastrophic failure of the Torus I believe is outside of the design basis and is not considered.

If he believes it is inside the design basis, he is misinformed. Either he is misinformed or he was trying to mislead this group and members of the general public by trying to convince everyone that we already have this interdependence of the independent barriers that provide the defense in depth. That is extremely troubling to me.

I have reviewed the ACRS' mission. And I believe the ACRS reviews certain changes and license amendments and makes recommendations to the Commission. When I say "the Commission," I'm talking the five commissioners.

I have reviewed some of the ACRS letters and typically find words along the lines -- I'll paraphrase it -- the ACRS is satisfied that the licensee will comply with all applicable regulations. Those are not the exact words but words along those lines whenever they are commenting on a proposed change, be it life extension, power upgrades, other license changes that the ACRS elects to review. That's not their sole responsibility. I believe that it is one of their responsibilities.

So how does the ACRS determine that this plant is in compliance with the applicable regulations? The Atomic Energy Act and the Energy Reorganization Act -- and, again, I'm going to paraphrase this -- make the statement along the lines that adequate protection to the public is provided if the licensee complies with the regulations. Those are not the exact words. I do have the exact words available, but it's pretty much the thought.

We have numerous indications that neither the licensee nor the NRC is fully cognizant of the compliance with the regulations. We brought up an issue. And we have written to Senators Leahy and Jeffers about the general design criteria.

The general design criteria were developed back in the mid '60s. I look at them as sort of the Ten Commandments. How do you design a power plant?

And the other and then the old regulations and reg guides and bulletins, orders, and all the other documents are interpretations of those commandments, such as one of the commandments "Thou shalt not kill."

Well, how does that apply in wartime? And there is always the area of abortion. These things are very vague and need clarifications. And other regulations interpret them and support it by various other supporting documents produced by the NRC.

When we reviewed this initial application and the updated final safety analysis report, we found that there was no commitment to the general design criteria in any of the licensing documents. In fact, in appendix F to the updated final safety analysis report, Vermont Yankee clearly made the statement that in this appendix, these are for historical purposes only.

About a year and a half ago, Mr. Arnold Gunderson and I asked for some clarification. So we filed a 2.206 because it really, really was not clear what the applicable general design criteria were.

And part of that 2.206 is up on the screen. And it requests basically that the NRC seek from Vermont Yankee clear and unambiguous definition of the general design criteria applicable to Vermont Yankee and how the facility's design conforms or deviates from the 70 draft or 62 final -- actually, 55 final.

The 2.206 petition was rejected after a year. And it's really not clear to any of us -- when I say "us, " I mean the NRC and the licensee -- exactly what regulatory requirements are applicable.

To give you an example, the NRC in their safety evaluation report mentions 64 general design criteria, final general design criteria. And the NRC isn't aware that there are only 55 of these general design criteria.

And then the safety evaluation report, the draft safety evaluation report, goes on to talk about compliance with the 70 draft design criteria.

Well, I went through a computer search of the SER, and they only mention 48 out of 70 draft criteria, how the other 22 got dropped -- and, believe me, those other 22 are not addressed in any of the other documents the NRC claims they are. The general design criteria is an example of compliance with regulations.

There are many other examples. If one goes through ADAMS at the NRC Web site, you will find that there are literally hundreds of exemptions to various regulatory requirements, including appendix J to 10 CFR 50, which I believe has to do with containment leak testing; appendix R, which is a fire prevention.

There are literally hundreds of exemptions that on their own may have been evaluation in isolation, but combined, we don't know the combined effect of all of these deviations from the regulations.

The ACRS contemplates a letter to the Commission. However, I believe the ACRS must assure itself that Vermont Yankee poses no undue risk to the public. In order to make that call, I believe the ACRS needs assurance that VY is in compliance with NRC regulations and identify all regulatory noncompliance.

It is the decision of the ACRS as to how to accomplish this clarification, whether it be an independent safety assessment, a matrix produced by the NRC, or some other vehicle that the ACRS can assure themselves that this plant is in compliance with the regulations and, therefore, provides reasonable assurance of public safety. Further verification of compliance with the NRC regulation, there is no assurance that the public will be adequately protected.

I would be more than happy to respond to any questions the Committee may have. Thank you.

MEMBER WALLIS: I had a question for you. It's a clarification. You started out giving me the impression that the GDCs were not referred to at all. And then later on you gave a list which seemed to indicate that most of them were but there may be some still missing. Which of those is it?

And if you know which ones are missing, maybe you could let us know so we know more specifically which ones you're concerned about.

MR. BLANCH: Yes. I have actually produced a list. In fact, I could give the Committee the draft 70 criteria, which are not easy to find, by the way. And I have them circled as to which ones have not been addressed.

As far as addressing the general design criteria, we look at the safety evaluation report, the draft one, that was just recently issued. That is only the applicability of the draft general design criteria to this change. It's not the general applicability.

One of the draft general design criteria -- I believe it's number 22 -- is single failure. That is not addressed. And Ms. Hobbs this morning was talking about a single failure that could take out two of our three primary barriers protecting the public. That is very troublesome to me.

I think the ACRS really needs to determine the degree of compliance and, therefore, safety of the Vermont Yankee plant, with or without the uprate.

MEMBER WALLIS: If we're talking about an uprate, it might be that some of these criteria are not relevant to the uprate in some way and that the changes brought about by the uprate make no difference or something I don't know yet until I have looked at it.

MR. BLANCH: Well -

MEMBER WALLIS: But we're not talking about Vermont Yankee in total. We're talking about an uprate.

MR. BLANCH: Well, I think if I were adding 20 percent to a building out in California, I would want to make sure that if I were adding 2 floors to a 10-story building, I would want to make sure that that building before I put the 2 stories complies with today's seismic requirements. That's my point.

(Applause.)

MEMBER KRESS: Let me put the onus back on you. How would you advise the ACRS to assure itself that the Vermont Yankee is in compliance with the regulations?

MR. BLANCH: I'm sorry. My -

MEMBER KRESS: How would you tell the ACRS to go about assuring itself that the Vermont Yankee is in compliance with all the regulations?

MR. BLANCH: Well, again, it's the ACRS' decision on how they determine that there is reasonable compliance with the regulations. The ACRS could write or direct the Commission that the staff evaluate Vermont Yankee for its compliance with the regulations and identify where it complies and where it deviates.

The ACRS could recommend to the Commission that they have some type of team in there and they go in, rather than an engineering inspection that had no acceptance criteria, to have a checklist. How do you meet the single failure criteria? How do you meet criterion 64, which is effluent rad monitoring, and, again, containment penetrations, fuel clad temperature? They're all in the design criteria. It's not an easy task. And this is the same request the Vermont state legislature made of the NRC, and that was rejected.

And I will not be confident that this plant can operate safely unless someone can reasonably demonstrate to me that it is in compliance, hopefully with today's regulations, but they don't want to go there.

CHAIRMAN DENNING: Thank you. Well, I think we understand.

DRAFT2 PROPOSED RESPONSE

January 21, 2006

Mr. Paul M. Blanch
135 Hyde Rd.
West Hartford, CT 06117

Dear Mr. Blanch,

Thank you for your letter of January 20, 2006 in which you raise questions about the role of the Advisory Committee on Reactor Safeguards and how we fulfill our responsibility in advising the Commissioners

Your concept of the role of the ACRS is not precisely how we undertake our advisory responsibility. As was indicated to you in Brattleboro, we are limited in our consideration of an EPU to those aspects of the licensing basis that are affected by the EPU. It is the responsibility of the regulatory staff in their consideration of the application to assure that the elements of the licensing basis that are changed are consistent with the set of regulations that is applicable. The staff accomplishes this task according to a set of guidelines that we have previously reviewed. The ACRS then reviews the Safety Evaluation Report prepared by the staff, to obtain assurance that the staff has done a technically competent job. The ACRS effort is much smaller than the effort undertaken by the staff. In addition, the character of the ACRS review is substantially different from that of the regulatory staff. Although the ACRS members (supported by our staff) must be cognizant of the NRC's regulations, the ACRS members are not selected because they are specialists in the interpretation of regulations but because of their technical expertise in the diverse aspects of reactor safety. Our value to the Commissioners is in our critical review of the technical quality of the work performed by the staff and our assessment of whether an adequate level of safety will continue to exist after the change in licensing basis.

Determining whether a licensee is operating in compliance with regulations (or more directly with its licensing basis) is not a primary role of the ACRS. Compliance must be monitored on a continuing basis because a licensee that is in compliance on one day may no longer be in compliance on the next day. For this purpose, the NRC has an inspection and enforcement arm.

After considering the technical issues that were raised in the mattter of the proposed EPU at Vermont Yankee, the ACRS decided that sufficient evidence was provided by the applicant and by the staff to resolve these issues This was the basis for our recommendation that the Entergy application should be approved.

Sincerely,

Graham Wallis

SUMMARY/MINUTES OF THE ACRS SUBCOMMITTEE MEETING

ON PLANNING AND PROCEDURES

JANUARY 26-27, 2006

BETHESDA NORTH MARRIOTT HOTEL

5701 MARINELLI ROAD

NORTH BETHESDA, MD

The ACRS Subcommittee on Planning and Procedures held an expanded meeting (retreat) on January 26-27, 2006 at the Bethesda North Marriott Hotel, 5701 Marinelli road, North Bethesda, MD. The meeting was convened at 8:30 a.m. on January 26, 2006 and recessed at 4:45 p.m. It was reconvened at 8:30 a.m. on January 27, 2006 and adjourned at 10:45 a.m. A portion of this meeting was closed pursuant to 5 U.S.C. 552 (c) (b) to discuss personal information.

ATTENDEES

ACRS Members:
G. Wallis (Subcommittee Chairman)
W. Shack
G. Apostolakis
M. Bonaca
R. Denning
T. Kress
D. Powers

ACRS Staff:
J. Larkins
A. Thadani
S. Duraiswamy
R. Savio
J. Flack
M. Scott
M. Snodderly
H. Nourbakhsh
J. Lamb
R. Caruso
C. Santos
E. Thornsbury
J. Gallo
M. Afshar-Tous
S. Meador
T. Brown
B. White

Invited Experts:
O. Maynard
S. Armijo

NRC Staff:
R. Assa, RES (part-time)

22.

LIST OF TOPICS DISCUSSED

The Subcommittee discussed the following topics:

- Anticipated workload
- Technical Expertise Needed on the ACRS, ACRS staff, and Consultants
- Strategy for Handling Anticipated Heavy Workload
    - Increasing the ACRS membership
    - Establishing new Subcommittees
    - Expanding meeting days
    - Increasing the number of ACRS full Committee meetings
    - Increasing the number of senior staff engineers and senior technical advisors
- Strategy for Seeking Candidates for Future Membership on the ACRS
    - Establishing an Ad Hoc Subcommittee to seek potential candidates through interaction with industry and other sources
    - Maintaining a pool of candidates with expertise in different areas
- Proactive Initiatives
    - Advanced reactor design for hydrogen production
    - PRA model uncertainties
    - Impact of power uprates on the safety of nuclear plants in light of other ongoing regulatory activities
    - Differences in regulatory approaches between U.S. and other countries
- Status of Implementing Commitments Made to Address significant comments received from Stakeholders during ACRS Self-Assessment Survey
    - Early interaction with the NRC staff on the regulatory significance of complex technical issues
    - More ACRS members with industry and plant operating experience
    - frequent interruption by the ACRS members during NRC staff presentation and the need for enhanced understanding of regulatory issues and process
- Proposed Options for ACRS/ACRS staff Knowledge Management
- Assignments, arrangements, and schedule for providing abstracts for papers for the Quadripartite Meeting

- Significant Issues
  - Advanced reactor designs (significant technical challenges)
  - Early site permit issues
  - License renewal issues
  - Extended power uprate issues
  - Risk-informing 10 CFR Part 50
  - Safety conscious work environment/safety culture
  - Improvements to Regulatory Guide 1.174
  - Living PRAs

Summary of Actions, agreements, assignments, and follow-up items.

1. Need a member with expertise in the thermal-hydraulics area to replace Dr. Ransom. We also need a member to replace Dr. Denning who will be leaving in July 2006. In addition, we need a member with expertise in nuclear analysis and fuels. Consultants could be used to augment the ACRS expertise in the areas of digital I&C, structural engineering/seismic issues, and design engineering.

2. Prepare a draft Commission paper to respond to the Commission's request in the December 20, 2005 SRM that the ACRS, following its retreat in January 2006, inform the Commission of its plans to manage the increased workload resulting from the anticipated receipt of new reactor designs and COL applications. This paper should also address resource needs. It should seek Commission approval to:
   - Increase the number of ACRS members to the statutory limit of 15 by FY 2007
   - Add two senior technical advisors in FY 2006, two Senior staff engineers and one administrative staff in FY 2007.

3. Expand the number of ACRS meeting days, as needed, as has been done in the past.

4. Revise the ACRS Subcommittee structure to establish new Subcommittees to deal with specific advanced reactor designs and COLs.

5. Establish an Ad Hoc Subcommittee, Chaired by Dr. Apostolakis, to seek potential candidates and to maintain a pool of candidates with expertise in various disciplines for future membership on the ACRS.

6. Continue to explore international regulatory approaches and inform the Commission where there are significant differences in regulatory approaches and requirements between U.S. and other countries.

7. Identify significant model uncertainties in PRAs and how to deal with such uncertainties in the regulatory decisionmaking process.

8. Initiate new proactive projects such as Passive System Reliability and Multi-National Design Approval initiative subject to the availability of Committee time and resources.

9. Dr. Denning should prepare a White Paper on extended power uprate issues (steam dryer cracking, SRV performance, BWR material cracking) and lessons learned from the review of power uprate applications.

10. The members should try not to interrupt the presenters during the first 10 minutes of their presentations.

11. Develop a Knowledge Management Pilot Program Plan for the ACRS Office. After implementing such Pilot Plan, present it to the other Offices of the Agency to coordinate potential integration of the ACRS Plan with their Knowledge Management Programs.

12. Members with assistance from cognizant staff engineers, should provide abstracts of technical papers in their assigned areas by the end of March 2006 that will be presented at the Quadripartite meeting scheduled for October 18-20, 2006.

13. For the 2006 ACRS plant visit, the ACRS should consider selecting a plant that has risk monitors.

14. During the next retreat, make arrangements for a presentation by Mr. Sam Walker, NRC Historian, on the history of the ACRS.

6

G:\ACRS-SECRETARY\reconciliation.wpd

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
529th	11	1

Title **RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS**

List of Documents Attached

See attached list

11

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

Lead Staff Person
SAM DURAISWAMY

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

February 3, 2006

MEMORANDUM TO: Dana A. Powers

FROM: H. P. Nourbakhsh, Senior Staff Engineer

SUBJECT: ANALYSIS OF OFFICE OF NUCLEAR REGULATORY RESEARCH (RES) RESPONSE TO ACRS LETTER ON"ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS - FY 2005"

Attached for your perusal is a copy of the RES's December 7, 2005 letter of response to ACRS's November 4, 2005 letter providing the findings from an assessment performed by the Committee to evaluate the quality of selected NRC research projects. Copies of the RES's letter of response and the committee's November 4, 2005 letter are attached.

Committee Letter

In its letter, the Committee summarized the results of its quality assessment of the following projects:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program

  This project was found to be more than satisfactory. The results meet the research objectives.

- Steam Generator Tube Integrity Program at the Argonne National Laboratory

  This project was found to be satisfactory. The results meet the research objectives.

- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University

  This project marginally satisfied the research objectives. The Committee identified significant deficiencies.

The specific comments as well as the methods used by the Committee for the quality review of research projects are described in detail in a separate report accompanying the letter.

RES Response

The RES's response also touched on the Committee's letter of November 4, 2005, providing the Committee's report on the quality review of selected NRC research projects. Following are the staff's responses to the Committee's assessment:

1. The staff is pleased that the Committee found that a research project entitled (Station Blackout Risk Evaluation for Nuclear Power Plants" to be more than satisfactory and that a project entitled " Steam Generator Tube Integrity" to be satisfactory.

2. With respect to the third project entitled "Rode Bundle Heat Transfer (RBHT) Test Program" that the Committee found to be marginally satisfactory, the staff would like to make the following clarifications:

    - The RBHT Program at the Pennsylvania State University (PSU) was originally intended to generate rode bundle data for the reflood phase of a loss-of-coolant accident (LOCA). It has since been modified to provide experimental data for other thermal -hydraulic processes for which the staff has a need.

    - The "Interfacial Drag" test report reviewed by the ACRS was a preliminary draft documenting one of the RBHT test series, and represents only a small portion of the overall RBHT program.

    - The work scope for the "Interfacial Drag" tests was limited, a more comprehensive evaluation of these data is planned by the staff.

    - The staff are willing to meet with the ACRS or one of its subcommittees to discuss the RBHT program in order to resolve any concerns.

    In an enclosure to th the RES's December 7, 2005 letter of response to ACRS's November 4, 2005 letter, the staff provided the following specific comments on issues raised by the ACRS in its review of the report on RBHT test program:

    - The Staff agrees with the Committee that the term "interfacial drag" in the title of report can be confusing as it is not a quantity that can be directly measured. The staff will request that PSU change the title in the final draft to "RBHT Two-Phase Mixture Level and Uncovery Test Data Report."

    - The Staff agrees with the Committee that the preliminary draft had significant problems that made it confusing and difficult to use. The staff stated that in a revised draft, provided to the staff, a number of corrections were made to improve the clarity. The staff also stated that a separate section (Appendix B) was added to provide additional details on the data reduction.

    - The staff intends to re-examine these data and the data reduction in the future, before they are used for model and correlation development. The staff stated that questionable assumptions involving the treatment of fluid properties, flow patterns, and magnitude of the bundle pressure drop will be revised if those assumptions made by PSU are found to be inadequate.

- The staff will examine PSU's assumptions regarding flow patterns. The staff states that the discussion of flow patterns in the preliminary draft appears to be unrelated to their calculation of pressure drop components. The staff do not intend to use the output from their "Energy Balance" program in determining the actual flow pattern for these data, or as a means of evaluating the TRACE code's prediction of the flow pattern.

- The staff indicates that the effect of flashing has been addressed in Appendix G to the revised draft. The staff also states that the contribution of flashing to the total vapor generation in the bundle was found to be small, and neglecting that term in the energy balance was a reasonable assumption.

- The staff agrees that the additional work is needed to in order to explain the effect of spacer grids. Staff states that the grid effect on low void and low flow rates will receive additional consideration in future evaluations of these data.

- The staff acknowledge that the treatment of uncertainties and characterization of sensitivities was missing from the draft report. The staff stated that the revised draft now contains a section (Appendix F) on uncertainties.

3. In its continuing efforts to assess and improve the quality of research projects, the staff are performing a lessons learned review of this issue.

4. The staff appreciates the Committee's efforts in reviewing the quality of these selected research projects. The staff will soon propose a list of candidate projects for ACRS review in FY 2006.

Analysis

The RES's response is satisfactory. The RES has agreed on many issues raised by the ACRS in its review of the report on RBHT test program. Since the staff intends to use the RBHT data for model development and assessment of the TRACE computer code, the Committee will be afforded opportunities to provide further review and comment on this research program.

Attachments: As Stated

cc w/o att. (via E-mail):
ACRS Members
J. Larkins
J. Flack
M. Snodderly
R. Savio
S. Duaiswamy
ACRS Technical Staff

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

December 7, 2005

Dr. William J. Shack
Acting Chairman
Advisory Committee on Reactor Safeguards (ACRS)
Washington, D.C. 20555-0001

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NUCLEAR REGULATORY COMMISSION RESEARCH PROJECTS - FY2005

Dear Dr. Shack:

We received your letter of November 4, 2005, informing us of the results of your assessment of selected NRC research projects in FY 2005. We are pleased to know that a project entitled "Station Blackout Risk Evaluation for Nuclear Power Plants" was found to be more than satisfactory and that a project entitled "Steam Generator Tube Integrity" was found to be satisfactory. With respect to the third project entitled "Rod Bundle Heat Transfer (RBHT) Test Program" that was found to be marginally satisfactory, we would like to make some clarifications as indicated below.

The RBHT Program at Pennsylvania State University (PSU) was originally intended to generate rod bundle data for the reflood phase of a large break loss-of-coolant accident (LOCA). It has since been modified to provide experimental data for other thermal-hydraulic processes for which the staff has a need. The "Interfacial Drag" test report reviewed by the ACRS was a preliminary draft documenting one of the RBHT test series, and represents only a small portion of the overall RBHT program. The work scope for the "Interfacial Drag" tests was limited, and a more comprehensive evaluation of these data is planned by the staff. Overall, this experimental program has provided valuable data for assessment of models and correlations in a two-fluid code.

The enclosure addresses specific Advisory Committee on Reactor Safeguards review comments, and describes how the RBHT data has been used by the staff. We are willing to meet with the ACRS or one of its Subcommittees to discuss this report and the RBHT program in order to resolve any concerns.

In our continuing efforts to assess and improve the quality of research products, we are performing a lessons learned review of this issue.

We appreciate your efforts in reviewing these documents, and will soon propose a list of candidate projects for your review in FY 2006.

Sincerely,

Carl J. Paperiello

Carl J. Paperiello, Director
Office of Nuclear Regulatory Research

Enclosure:
Response to ACRS Comments
on the RBHT Report

RESPONSE TO ACRS COMMENTS ON A REPORT ENTITLED "ANALYSIS OF ROD BUNDLE HEAT TRANSFER FACILITY TWO-PHASE INTERFACE DRAG EXPERIMENTS AT THE PENNSYLVANIA STATE UNIVERSITY"

1. Introduction

The purpose of this enclosure is to provide specific comments on issues raised by the ACRS in its review of the report entitled "Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Pennsylvania State University." As the Committee noted, the report reviewed was a preliminary draft intending to document the results from one of several test programs conducted using the Rod Bundle Heat Transfer (RBHT) facility. The objective of the "interfacial drag" test program was to conduct a series of rod bundle experiments providing detailed measurements to determine the void fraction distribution for a range of pressures, inlet flow rates, inlet temperatures, and bundle power. The staff intends to use these data for model development and assessment of the TRACE computer code.

The title of the report would be more appropriate as the "RBHT Two-Phase Mixture Level and Uncovery Test Data Report" since the goal was to obtain and document the data. We agree that the term "interfacial drag" can be confusing as it is not a quantity that can be directly measured. The term was used because of the usefulness of these data for assessment of models and correlations for interfacial shear in a two-fluid code. We will request that Pennsylvania State University (PSU) change the title in the final draft.

The measurements taken by the RBHT Program are considered to be more detailed than in previous studies of two-phase level swell because of the larger number of differential pressure cells in the RBHT facility compared to that in other facilities such the ORNL-THTF and FLECHT-SEASET bundles. A description of the RBHT test facility is documented in another report which was made available to the ACRS Thermal Hydraulic Subcommittee. Locations of the pressure taps can be found in that document.

2. Documentation

The ACRS had several comments on the documentation, and assigned a consensus score of 4.33. We agree that the preliminary draft had significant problems that made it confusing and difficult to use. In a revised draft provided to the staff on September 30, 2005, a number of corrections were made to improve the clarity. A separate section (Appendix B) was added to provide additional details on the data reduction.

3. Identification of, and justification of major assumptions

One of the primary goals of the data report is to document the test results and data reduction so that they are scrutable. We intend to re-examine these data and the data reduction in the future, before they are used for model and correlation development. Questionable assumptions involving the treatment of fluid properties, flow patterns, and magnitude of the bundle pressure drop will be revised if those assumptions made by PSU are found to be inadequate.

As part of the staff's evaluation of these data, we will examine PSU's assumptions regarding flow patterns. The discussion of flow patterns in the preliminary draft appears to be

Enclosure

unrelated to their calculation of pressure drop components. We do not intend to use the output from their "ENERGY BALANCE" program in determining the actual flow pattern for these data, or as a means of evaluating the TRACE code's prediction of the flow pattern.

In the revised draft, the effect of flashing was addressed in Appendix G. The contribution of flashing to the total vapor generation in the bundle was found to be small, and neglecting that term in the energy balance was a reasonable assumption.

We agree that additional work is needed in order to explain the effect of the spacer grids. From these and other experiments, it is clear that spacer grids have a profound effect on a two-phase flow even at low void fractions. The spacer grid effect on low void and low flow rates will receive additional consideration in future evaluations of these data.

4. Soundness of technical approach and results

A discussion of models currently in TRACE and a direct comparison of the RBHT data to any particular model was above the PSU work scope. The intent of this research was to generate and document the experimental data, not make comparisons to theory.

The staff intends to perform a more comprehensive evaluation of these RBHT data as part of TRACE model development. Derivation of interfacial drag models, a critical examination of the data, and comparison of the data to TRACE calculations will be performed in future work. These were not part of the present scope of work.

5. Treatment of uncertainties and characterization of sensitivities

This was missing from the preliminary draft. The revised draft now contains a section (Appendix F) on uncertainties.

USAGE OF DATA FROM THE ROD BUNDLE HEAT TRANSFER TEST FACILITY

Rod Bundle Heat Transfer Test Description

Three types of tests have previously been conducted:

- Forced Reflood Tests – a series of 33 reflood tests with forced inlet flow conditions. The objective of this test series was to provide detailed data to aid in the development of an advanced reflood model for the TRACE code and to provide a comprehensive separate effects assessment data base (effects of pressure, flooding rate and inlet subcooling). Both the test matrix and the instrumentation were designed to enhance the value of these tests for model development as discussed below.

- Interfacial Drag Tests – steady state axial void fraction profiles were obtained for 120 different combinations of pressure, inlet flow and rod power. The objective of this test series was to provide a comprehensive low pressure data base for void fraction in rod bundles under conditions representative of a PWR LBLOCA and the long-term cooling phase of an advanced passive plant. This data base will be used to first check the adequacy of the TRACE interfacial friction model and, if necessary, provide the means for improving this model.

- Boil-Off Test – a low-pressure transient boil-off and recovery test was conducted in support of AP-1000 to help resolve the question of bundle inventory requirements to prevent core heatup. This test will also be used as part of the TRACE assessment matrix.

From FY 2005 through FY 2006, the following types of tests have been or are being conducted:

- Steam Cooling Tests – these tests were conducted to determine forced convection and mixed convection single-phase vapor heat transfer coefficients in a prototypic rod bundle geometry with mixing vane grid spacers. These tests will also be used to define a base line for the two-phase enhancement that occurs in dispersed flow film boiling (see droplet injection tests) and provide the effect of the grid spacers on single-phase convective heat transfer.

- Droplet Injection Tests – these tests are targeted at providing the data base necessary to develop a two-phase enhancement model, as well as, provide data for the drop-vapor interfacial heat transfer rate and the grid spacer effect. In addition to their role in limiting vapor superheat, the presence of dispersed droplets also serve to enhance the convective heat transfer.

Past and Current Use of RBHT Data

To date, the RBHT data has been used in three ways: 1) to directly support the AP-1000 licensing activity, 2) to contribute to the growing TRACE assessment matrix, and 3) to support international collaboration. A brief description of each of these activities is given below.

First, during the AP-1000 design certification review, calculations by both the staff and Westinghouse showed core collapsed liquid levels slightly greater than 55 percent of the total core height. There was a lack of low-pressure level swell data in the right range to determine the threshold at which core heatup should be expected. The interfacial drag tests then being conducted in the RBHT facility, were extended to the point where core heatup was initiated to determine this threshold. In addition, a special boil-off and recovery test was conducted to provide for code assessment of core inventories below this threshold value.

The second way that the RBHT data is currently being used is in the TRACE assessment effort. Now that the code consolidation effort has been essentially completed, the focus has shifted to improving code robustness, computational efficiency, and accuracy. Both the interfacial drag tests (all 120 data sets) and the reflood tests (8 of the 33 tests) are currently being simulated with TRACE and will become part of the permanent assessment matirx that will be repeated before every major code release. The first public release of TRACE (version 5.0) is scheduled for the end of calendar year 2006. A developmental assessment report will accompany the code release and will include both the RBHT interfacial drag and reflood tests. The interfacial drag tests provide a comprehensive low-pressure data base for void fraction in rod bundles and greatly extend the existing data base.

Similarly, the RBHT reflood tests, in addition to their detailed instrumentation designed for model development needs, serve to fill some gaps in the existing reflood data base. First, the RBHT facility uses a top-skew power profile that more closely represents a peak PCT case than the cosine power shape used in almost all other non-proprietary reflood tests[1]. Second, the RBHT test matrix includes a number of tests with low subcooling, as opposed to the high subcoolings (60-80 K) common to other separate effects reflood tests.

The third way that the RBHT data is being used is in the area of international collaboration. Indeed, two international agreements have been concluded in which the RBHT reflood data has been "bartered." First, a joint development project was initiated with Korea. Two Korean engineers have each had one-year assignments with the NRC where they performed simulations with the TRACE code.

Second, an agreement between the French IRSN and the NRC resulted in the exchange of the RBHT reflood data for a set of seven BETHSY integral effects tests covering a wide range of small to intermediate size LOCAs (1 inch up to 10 inches). These tests will become part of the permanent TRACE assessment matrix and will provide a significant contribution to ensuring code accuracy for break sizes up to the transition break size (TBS).

Use of RBHT Data in Model Development

Model development activities using the RBHT data have been delayed by approximately two years due to the emergence of the ESBWR and the 50.46 program. These emerging issues required the reassignment of the staff members originally tasked with this effort. The current plan calls for this model development activity to begin in January 2006, and be completed in

---

[1] The FLECHT-Skewed tests also have a top-skewed power distribution, however, the data files in the NRC data bank have the instrumentation tags scrambled rendering this data almost useless until it can be recovered from Westinghouse.

2008. Each of the TRACE models that will undergo detailed assessment and/or model development is listed below, along with a short description as to why the RBHT data are of particular value for the model being investigated.

- Entrainment – the current TRACE version does not explicitly model droplet entrainment, instead "entrainment" is simulated by modifying the interfacial drag between the vapor and liquid fields. This has some very real drawbacks with respect to the determination of droplet diameter and the acceleration of the droplets; this in turn, affects the droplet volume fraction, the interfacial heat transfer, and finally, the level of vapor superheat. A droplet field has just been added to TRACE but the necessary physical models remain to be developed before this new capability can be used. The RBHT tests provide data that will be used to determine a correlation for the entrainment rate and the drop diameter.

- Inverted Annular Film Boiling – the current database does not provide enough resolution of the axial void profile in the region downstream of the quench front to address what appears to be a strong function of the local void fraction heat transfer in this regime. Indeed, the best available data (FLECHT-SEASET) uses delta-P cells to infer the void fraction with a span of 12 inches which often encompasses the entire inverted annular regime. The RBHT facility was designed to include a region with finely spaced delta-P cells (every 3 inches) to provide this missing data. Furthermore, the RBHT test matrix was constructed to provide a parametric effect on local subcooling at the quench front in the region of finely spaced delta-P cells.

- Inverted Slug Film Boiling – no suitable correlation exists, and most codes use an interpolation between inverted annular and dispersed flow. As the flow quality increases, the inverted annular "core" is broken up into ligaments and droplets of various shapes and sizes. Heat transfer in this regime appears to be a function both of vapor flow rate and void fraction. In addition to the finely spaced delta-P cells mentioned above, the RBHT test matrix includes a number of tests with low subcooling to provide data for this regime.

- Convective Heat Transfer – the data from the RBHT steam cooling tests will be used to assess the current TRACE model for convective heat transfer in rod bundles as well as the mixed convection effect.

- Two-Phase Enhancement – in dispersed flow film boiling, the convective heat transfer coefficient is enhanced above the single-phase value. No suitable model exists for this effect. The RBHT droplet injection tests are designed to provide the data needed to develop this model. These tests quantify both the entrained liquid flow rate and initial droplet diameter, which normally have to be inferred, as well as provide a measurement of the superheated vapor temperature that is significantly better than that of previous tests.

- Dispersed Flow Film Boiling – in addition to the contribution to this regime made though the development of the two-phase enhancement model discussed above, both the droplet injection and reflood tests provide the data necessary to evaluate and improve the TRACE model for drop-vapor interfacial heat transfer.

- Transition Boiling and Maximum Heat Flux – no suitable model exists for the transition boiling regime that occurs during the quenching of fuel rods nor for the maximum heat flux that occurs during the quenching. Instead, pool boiling critical heat flux models, usually with the Griffith void fraction modifier, are generally used to model the maximum heat flux, and the transition boiling region is approximated through an interpolation scheme. The finely spaced delta-P cells in the RBHT facility, together with a thermocouple sampling rate of 10 Hz (for comparison, FLECHT-SEASET used 1 & 2 Hz) allow for the calculation of the rod heat flux and local fluid conditions during the quenching process and thus provide the data base necessary to develop models for this regime.

- Minimum Film Boiling Temperature (Tmin) – the RBHT tests will be used to evaluate the effects of fluid conditions upon Tmin. The temperature at which transition boiling begins is critical in reflood calculations. Literature models give very conflicting trends with pressure, mass flux, subcooling and/or void fraction, and material properties. There is evidence that highly oxidized rods can quench from a significantly higher temperature. Material property effects will have to be investigated in another program if they are deemed important enough.

- Grid Spacer Models – the presence of mixing vane grids, such as those in the RBHT facility, enhance the heat transfer downstream significantly more than do the egg-crate spacer grids used in the FLECHT reflood tests. Three phenomena are thought to be responsible for this increase: 1) convective enhancement, 2) grid rewet, and 3) droplet breakup. Convective enhancement is due to the flow disturbance caused by the grids and will be studied using the RBHT steam cooling data. When a spacer grid rewets, it provides a large relatively cool surface that helps to de-superheat the steam and thereby enhance the rod heat transfer. Data from both the droplet injection tests and the initial reflood test series will be used to investigate this. In addition to the uniqueness of the droplet injection tests, the RBHT facility has finely spaced rod thermocuples just downstream of several grids, a large number of vapor probes, and grid thermocouples so the time of grid quench can be determined.

- Rod Bundle Interfacial Shear – the original TRACE interfacial friction model was found to seriously over-predict void fraction in rod bundles at low pressure. The Bestion model was implemented in TRACE to address this deficiency. Should the assessment against the RBHT interfacial friction tests show that this model is not suitably accurate, a new model can either be selected or developed using this data base.

In summary, the primary contributions of the RBHT data to our programs to date have been the resolution of the bundle inventory threshold for core uncovery question in the AP-1000 review, the additional assessment effort derived from the international agreement with Korea, and the high quality integral effects test data (BETHSY) obtained from France as part of a data exchange. TRACE assessment activities using the RBHT interfacial drag and reflood data have just begun. Finally, the model development activities for which the RBHT experiments were designed, have been delayed but will begin in January 2006.

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

ACRSR-2160

November 4, 2005

Dr. Carl J. Paperiello
Director
Office of Nuclear Regulatory Research
Washington, D.C. 20555-0001

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS - FY 2005

Dear Dr. Paperiello:

Enclosed is our report on the quality assessment of the following research projects:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program
  - This project was found to be more than satisfactory. The results meet the research objectives.
- Steam Generator Tube Integrity Program at the Argonne National Laboratory
  - This project was found to be satisfactory. The results meet the research objectives.
- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University
  - This project marginally satisfied the research objectives. The Committee identified significant deficiencies.

These projects were selected from a list of candidate projects suggested by the Office of Nuclear Regulatory Research.

A fourth research project on reactor containment performance being conducted at Sandia National Laboratories will be evaluated later, once a particularly pivotal report on the research becomes available.

We anticipate receiving your list of candidate projects for review during the next 12 months.

Sincerely,

/RA/

William J. Shack
Acting Chairman

Enclosure: As stated

ACRS Assessment of the Quality of Selected NRC Research Projects

October 2005

U.S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards
Washington, DC 20555-0001

ABOUT THE ACRS

The Advisory Committee on Reactor Safeguards (ACRS) was established as a statutory Committee of the Atomic Energy Commission (AEC) by a 1957 amendment to the *Atomic Energy Act* of 1954. The functions of the Committee are described in Sections 29 and 182b of the Act. The *Energy Reorganization Act* of 1974 transferred the AEC's licensing functions to the U.S. Nuclear Regulatory Commission (NRC), and the Committee has continued serving the same advisory role to the NRC.

The ACRS provides independent reviews of, and advice on, the safety of proposed or existing NRC-licensed reactor facilities and the adequacy of proposed safety standards. The ACRS reviews power reactor and fuel cycle facility license applications for which the NRC is responsible, as well as the safety-significant NRC regulations and guidance related to these facilities. On its own initiative, the ACRS may review certain generic matters or safety-significant nuclear facility items. The Committee also advises the Commission on safety-significant policy issues, and performs other duties as the Commission may request. Upon request from the U.S. Department of Energy (DOE), the ACRS provides advice on U.S. Naval reactor designs and hazards associated with the DOE's nuclear activities and facilities. In addition, upon request, the ACRS provides technical advice to the Defense Nuclear Facilities Safety Board.

ACRS operations are governed by the *Federal Advisory Committee Act* (FACA), which is implemented through NRC regulations at Title 10, Part 7, of the *Code of Federal Regulations* (10 CFR Part 7). ACRS operational practices encourage the public, industry, State and local governments, and other stakeholders to express their views on regulatory matters.

MEMBERS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Dr. George E. Apostolakis, Professor of Nuclear Engineering, Professor of Engineering Systems, Massachusetts Institute of Technology, Cambridge, Massachusetts

Dr. Mario V. Bonaca, Retired Director, Nuclear Engineering Department, Northeast Utilities, Connecticut

Dr. Richard S. Denning, Senior Research Leader, Battelle Memorial Institute, and Adjunct Professor, the Ohio State University, Columbus, Ohio

Dr. Thomas S. Kress, Retired Head of Applied Systems Technology Section, Oak Ridge National Laboratory, Oak Ridge, Tennessee

Dr. Dana A. Powers, Senior Scientist, Sandia National Laboratories, Albuquerque, New Mexico

Dr. Victor H. Ransom, Professor Emeritus, Purdue School of Nuclear Engineering, West Lafayette, Indiana

Dr. William J. Shack,(Vice-Chairman), Associate Director, Energy Technology Division, Argonne National Laboratory, Argonne, Illinois

Mr. John D. Sieber, **(Member-at-Large)**, Retired Senior Vice-President, Nuclear Power Division, Duquesne Light Company, Pittsburgh, Pennsylvania

Dr. Graham B. Wallis, (Chairman), Sherman Fairchild Professor Emeritus, Thayer School of Engineering, Dartmouth College, Hanover, New Hampshire

ACKNOWLEDGMENT

The Committee would like to acknowledge the contributions of Dr. Hossein Nourbakhsh and Mr. Sam Duraiswamy of the ACRS Staff to the development of this assessment.

ABSTRACT

In this report, the Advisory Committee on Reactor Safeguards (ACRS) presents the results of its assessment of the quality of selected research projects sponsored by the Office of Nuclear Regulatory Research(RES) of the NRC. An analytic/deliberative methodology was adopted by the Committee to guide its review of research projects. The methods of multi-attribute utility theory were utilized to structure the objectives of the review and develop numerical scales for rating the project with respect to each objective. The results of the evaluations of the quality of the three research projects are summarized as follows:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program

  - This project was found to be more than satisfactory . The results meet the research objectives.

- Steam Generator Tube Integrity Program at the Argonne National Laboratory

  - This project was found to be satisfactory. The results meet the research objectives.

- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University

  - This project marginally satisfied the research objectives. The Committee identified significant deficiencies.

CONTENTS

FIGURES

TABLES

ACRONYMS

Acronym	Definition
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ANL	Argonne National Laboratory
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
DOE	Department of Energy
EDG	Emergency Diesel Generator
FACA	Federal Advisory Committee Act
FY	Fiscal Year
GPRA	Government Performance and Results Act
LOCA	Loss-of-Coolant Accident
MAUT	Multi-Attribute Utility Theory
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk assessment
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
SBO	Station Blackout
SG	Steam Generator
SPAR	Standardized Plant Analysis Risk

1 INTRODUCTION

The Nuclear Regulatory Commission (NRC) maintains a safety research program to ensure that the agency's regulations have sound technical bases. The research effort is needed to support regulatory activities and agency initiatives while maintaining an infrastructure of expertise, facilities, analytical tools, and data to support regulatory decisions.

The Office of Nuclear Regulatory Research (RES) is required to have an independent evaluation of the effectiveness (quality) and utility of its research programs. This evaluation is required by the NRC Strategic Plan that was developed as mandated by the Government Performance and Results Act (GPRA). The Advisory Committee on Reactor Safeguards (ACRS) has agreed to assist RES by performing independent assessments of the quality of selected research projects. Quality assessment of individual research projects constitutes a new undertaking for the Committee; one that is quite different in scope and depth in comparison to the ACRS biennial review of the overall NRC research activities. During fiscal year (FY) 2004, the ACRS conducted a trial review of the quality of selected research projects [Ref. 1]. Based on the outcome of this trial review , the Committee has established the following review process:

- RES submits to the ACRS a list of candidate research projects for review because they have reached sufficient maturity that meaningful technical review can be conducted.
- The ACRS selects no more than four projects for detailed review during the fiscal year.
- A panel of three ACRS members is established to assess the quality of each research project.
- The panel follows the guidance developed by the ACRS full Committee in conducting the technical review. This guidance is discussed further below.
- Each panel assesses the quality of the assigned research project and presents an oral and a written report to the ACRS full Committee for review. This review is to ensure uniformity in the evaluations by the various panels.
- The Committee revises these reports, as needed, and provides them to the cognizant research manager, as appropriate.
- The Committee submits an annual summary report to the RES Director.

An analytic/deliberative decisionmaking framework was adopted for evaluating the quality of NRC research projects. The definition of quality research adopted by the Committee includes two major characteristics:

- Results meet the objectives
- The results and methods are adequately documented

Within the first characteristic, ACRS considered the following general attributes in evaluating the NRC research projects:

- Soundness of technical approach and results
  - Has execution of the work used available expertise in appropriate disciplines?

- Justification of major assumptions
  - Have assumptions key to the technical approach and the results been tested or otherwise justified?

- Treatment of uncertainties/sensitivities
  - Have significant uncertainties been characterized?
  - Have important sensitivities been identified?

Within the general category of documentation, the projects were evaluated in terms of following measures:

- Clarity of presentation

- Identification of major assumptions

In this report, the ACRS presents the results of its assessment of the quality of the research projects associated with:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program

- Steam Generator Tube Integrity Program at the Argonne National Laboratory

- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag experiments at the Penn State University

These projects were selected from a list of candidate projects suggested by RES.

A fourth research project on reactor containment performance being conducted at Sandia National Laboratories will be evaluated during FY-2006, once a particularly pivotal report on this research becomes available.

The methodology for developing the quantitative metrics (numerical grades) for evaluating the quality of NRC research projects is presented in Section 2 of this report. The results of assessment and ratings for the selected projects are discussed in Section 3.

2 METHODOLOGY FOR EVALUATING THE QUALITY OF RESEARCH PROJECTS

To guide its review of research projects, the ACRS has adopted an analytic/deliberative methodology [Ref. 2 and 3]. The analytical part utilizes methods of multi-attribute utility theory (MAUT) [Ref. 4 and 5] to structure the objectives of the review and develop numerical scales for rating the project with respect to each objective. The objectives were developed in a hierarchical manner (in the form of a "value tree"), and weights reflecting their relative importance were developed. The value tree and the relative weights developed by the full Committee are shown in Figure 1.

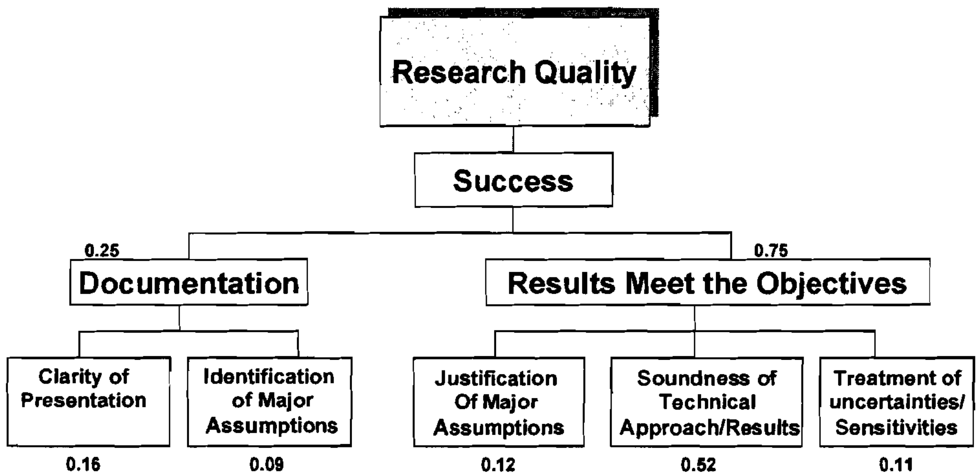

Figure 1 The value tree used for evaluating the quality of research projects

The quality of projects is evaluated in terms of the degree to which the results meet the objectives of the research and of the adequacy of the documentation of the research. It is the consensus of the ACRS that meeting the objectives of the research should have a weight of 0.75 in the overall evaluation of the research project. Adequacy of the documentation was assigned a weight of 0.25. Within these two broad categories, research projects were evaluated in terms of subsidiary "performance measures":

- justification of major assumptions (weight: 0.12)
- soundness of the technical approach and reliability of results (weight: 0.52)
- treatment of uncertainties and characterization of sensitivities (weight: 0.11)

Documentation of the research was evaluated in terms of the following performance measures:

- clarity of presentation (weight: 0.16)
- identification of major assumptions (weight: 0.09)

To evaluate how well the research project performed with respect to each performance measure, constructed scales were developed as shown in Table 1. The starting point is a rating of 5, Satisfactory (professional work that satisfies the research objectives). Often in evaluations of this nature, a grade that is less than excellent is interpreted as pejorative. In this ACRS evaluation, a grade of 5 should be interpreted literally as satisfactory. Although innovation and excellent work are to be encouraged, the ACRS realizes that time and cost place constraints on innovation. Furthermore, research projects are constrained by the work scope that has been agreed upon. The score was, then, increased or decreased according to the attributes shown in the table. The overall score of the project was produced by multiplying each score by the corresponding weight of the performance measure and adding all the weighted scores.

The value tree, weights, and constructed scales were the result of extensive deliberations of the whole ACRS. As discussed in Section 1, a panel of three ACRS members was formed to review each selected research project. Each member of the review panel independently evaluated the project in terms of the performance measures shown in the value tree. The panel deliberated the assigned scores and developed a consensus score, which was not necessarily the arithmetic average of individual scores. The panel's consensus score was discussed by the full Committee and adjusted in response to ACRS members' comments. The final consensus scores were multiplied by the appropriate weights, the weighted scores of all the categories were summed, and an overall score for the project was produced. A set of comments justifying the ratings was also produced.

Table 1. Constructed Scales for the Performance Measures

SCORE	LABEL	INTERPRETATION
10	Outstanding	Creative and uniformly excellent
8	Excellent	Important elements of innovation or insight
5	Satisfactory	Professional work that satisfies research objectives
3	Marginal	Some deficiencies identified; marginally satisfies research objectives
0	Unacceptable	Results do not satisfy the objectives or are not reliable

3. RESULTS OF QUALITY ASSESSMENT

3.1 STATION BLACKOUT RISK EVALUATION FOR NUCLEAR POWER PLANTS PERFORMED AS A PART OF SPAR MODELS DEVELOPMENT PROGRAM

In 1988, the NRC issued the Station Blackout Rule, 10 CFR 50.63, and the associated Regulatory Guide 1.155 establishing requirements and guidance to ensure decay heat removal for the period following loss-of-offsite power. Subsequent Probabilistic risk assessments (PRAs) indicated that compliance with these regulatory documents resulted in appropriately small core damage frequencies for station blackout (SBO) scenarios. On August 14, 2003, a widespread grid-related loss-of-offsite power event resulted in the controlled shut down of nine nuclear power plants. The NRC initiated a program to reevaluate the frequencies and durations of loss-of-offsite power, as well as the SBO risk contribution. The results of this study are documented in Reference 6. This report that the Committee reviewed is an update of previous reports analyzing the risk from loss-of-offsite power and subsequent SBO events in all operating U.S. power plants.

The SPAR models were used to evaluate the core damage frequency from internal events only for each plant during power operation. A number of enhancements to the SPAR models had to be made for this evaluation. The reliability estimates for diesel generators were also updated using recent data. Updated data were also collected for turbine-driven pumps, high-pressure core spray motor-driven pumps, and diesel-driven pumps. For the pressurized water reactors (PWRs), pump-seal failure models were selected based on the most recent developments.

The scope of this quality review is limited to the above report rather than a broader assessment of the quality of the updated SPAR models requested by RES. The Committee judged that it would have been overly ambitious to undertake such an evaluation in a single step and within the time constraints of the present review. The ACRS decided to have its Reliability and Probabilistic Risk Assessment Subcommittee perform a much broader review of the SPAR models during the upcoming year. Thus, in evaluating this report, the Committee has not considered the validity of the SPAR models that form the basis for the study.

<u>GENERAL OBSERVATIONS</u>

This report is an excellent example of the value of the SPAR Models Development Program and of the contribution that RES can make to the understanding of the safety of operating plants. The independent capability to evaluate risk issues across the population of operating plants has great value. By utilizing the same model and assumptions for all types of reactors in the fleet, the staff has been able to reach several conclusions regarding the effects of plant-specific design features on the risk from SBO. The availability of these models allows for periodic reevaluation of issues and trends associated with, for example, the effect of deregulation on grid reliability, and the effect of online maintenance on SBO.

The consensus scores for this project are shown in Table 2. This project was found to be more than satisfactory with a number of elements of excellence present. Comments and conclusions within the evaluation categories are:

Table 2. Summary Results of ACRS Assessment of the Quality of the Project on Station Blackout Risk Evaluation for Nuclear Power Plants

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	7.0	0.16	1.12
Identification of major assumptions	7.0	0.09	0.63
Justification of major assumptions	6.33	0.12	0.76
Soundness of technical approach/results	6.66	0.52	3.46
Treatment of uncertainties/sensitivities	6.0	0.11	0.66
		Overall Score:	**6.63**

Documentation

- Clarity of presentation **(Consensus score = 7.0)**

  The report is clearly written and well organized. It provides a good description of prior work and describes in detail the logic utilized in the selection of databases and assumptions. It presents the results in the context of previous evaluations, provides good explanation of changes, and discusses important trends and insights.

- Identification of major assumptions **(Consensus score = 7.0)**

  Assumptions are clearly stated, and the report does a good job of explaining the logic behind these assumptions.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 6.33)**

  Major assumptions are generally well justified, for example the use of industry-average data rather than plant-specific data for component unreliability, train test and maintenance outage probabilities, and initiating event frequencies.

In some instances, a full explanation is not provided. For example, no argument is provided for not modifying the Babcock & Wilcox (B&W) seal leakage model, except that there is no pending submittal to the NRC. From that statement, the reader is left with no insights regarding the quality of the B&W seal leakage model. Another example is the choice of a factor of two in the emergency diesel generator (EDG) performance sensitivity study. It is not clear why a factor of two was chosen.

- Soundness of technical approach and results **(Consensus score = 6.66)**

  There is nothing novel about the approach (this is not a criticism). The event trees are borrowed from those that had been developed previously.

  The use of industry-wide data to place all nuclear power plants on a common basis helped in determining the relative effectiveness of general features of electric power systems and backup safe shutdown modes in reducing the risk from SBO.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 6.0)**

  The report includes the results of an uncertainty analysis and of a sensitivity study.

  The sensitivity results are point estimates, i.e., no uncertainty analyses were performed for the sensitivity cases. It is this last point that generated discussion among the panel members. What does a "sensitivity analysis" mean in the probabilistic world? In traditional engineering analysis where all the calculations were done on a "point estimate" basis, a sensitivity study usually means to vary, more or less arbitrarily, various parameters and evaluate their impact on the final answer. In probabilistic analyses, this approach must be reconsidered. Possible variability in parameter values should be included in the uncertainty distributions of these parameters. The focus should be on the assumptions and parameters that drive the results. An example is the use of the risk achievement worth to identify events that may have a significant impact on the core damage frequency calculated in a PRA. The ACRS acknowledges that this issue should be discussed in a broader context with the staff and that, perhaps, it would be unfair to judge the authors of this report harshly on an issue that has not been widely debated.

3.2 STEAM GENERATOR TUBE INTEGRITY PROGRAM AT THE ARGONNE NATIONAL LABORATORY[1]

The overall objective of the steam generator tube integrity research program is to provide experimental data and predictive correlations and models needed to permit the NRC staff to independently evaluate the integrity of steam generator tubes as plants age and degradation proceeds, new forms of degradation appear, and as new defect-specific management schemes are implemented. This program builds upon the results of NRC steam generator tube integrity and inspection research conducted since 1977.

The objectives of the specific project (task 3, Research on Tube Integrity and Integrity Predictions) selected for quality assessment were to:

- Determine if the flow stress of MA Nickel Alloy 600 tube material exhibits dependence on the stress rate or the strain rate (i.e.: the rate of internal pressurization).

- Determine the relationship between crack or ligament size (width, depth, and length), orientation, geometry, morphology, and number of ligaments and the tube leak rate and burst pressure.

- Confirm the validation of the tube leak rate correlation model and its relevance to choked two-phase flow expected at operating temperatures and pressures, including the relative uncertainties involved under various conditions.

- Compare laboratory leak rate and burst pressure models with the results of tests of samples of defective steam generator tubes removed from a decommissioned steam generator from McGuire Nuclear Plant.

These studies were conducted at the Argonne National Laboratory. The results of studies that the ACRS reviewed were documented in References 7 and 8.

The consensus scores for this project are shown in Table 3. This project was found to be satisfactory. The results meet the research objectives. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 4.7)**.

  The manuscripts documenting the results of this project [Ref. 7 and 8] are exceptionally informal. These documents read like laboratory reports prepared by technicians and sent to professional staff to be used in the preparation of a more formal report. Both manuscripts are rather more summary in nature. This terse informality of documentation makes the reports more readable though incomplete.

Table 3 Summary Results of the ACRS Assessment of the Quality of the Project on Steam Generator Tube Integrity

[1] Dr. William J. Shack, ACRS member, did not participate in the Committee's deliberations regarding this matter.

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	4.7	0.16	0.75
Identification of major assumptions	4.7	0.09	0.42
Justification of major assumptions	4.7	0.12	0.56
Soundness of technical approach/results	5.0	0.52	2.6
Treatment of uncertainties/sensitivities	4.3	0.11	0.47
		Overall Score:	**4.8**

The reports are inadequate for the archival documentation of expensive tests. Experimental methods are mentioned in casual ways with no effort, even by reference, to show that these methods are adequate or produce reliable, reproducible results.

Calibration and qualification of instruments are not discussed at all.

Theoretical models and even data analysis methods are mentioned without reference.

Figures showing data and correlations are exceptionally difficult to interpret since minimal legends and labeling are employed despite the figures being quite "busy." For the leak rate studies (page 34 of Ref. 7),except for specimen SLG900, no results are provided. The discussion on page 44 is not clear when correlating L/D ratios and choked flow.

A reader who does not routinely examine reports from this laboratory and is not intimately familiar with the equipment and methods of the laboratory will have difficulty in understanding the documentation. (Only after reading Ref 8 did one come to understand that the unlabeled scale in some photos in Ref. 7 was an inch scale and not a centimeter scale despite all the text on lengths referring to millimeters!) In the end, one can understand the points the authors are trying to make in Ref. 7, but with difficulty. Clarity of presentation is not of high quality, but adequate to understood the work.

It is dubious that the experimental results could ever be used directly in a regulatory process involving licensees. The qualification of methods and calibration of instruments simply will not be acceptable for such direct use.

- Identification of major assumptions **(Consensus score = 4.7)**

The major assumptions employed are not separately and explicitly stated but some of these assumptions are embedded in the text. In a complex report such as this, it is an acceptable and appropriate practice to state assumptions in the context of the issues where they are used or evaluated and rejected.

As noted above, identification and justification of assumptions are difficult to evaluate. There is not a coherent effort to do this in the document largely because it is not evident that results have any applicability. It is not evident that the results for the notched specimens discussed in the document will be used to infer the behavior of real cracks in tubes under accident conditions.

The investigators have done a better job in identifying factors that will affect the experimental results and including their sensitivities in test programs.

The documentation does not provide adequate justification for sensitivities that are included nor does it include discussions concerning the sensitivities of other factors that has not been considered.

The document fails completely to address uncertainties in measurements or to provide adequate descriptions of parametric uncertainties in reporting results of fitting the data to correlations. Presumably, if needed, these uncertainties as well as uncertainties in measurements could be extracted. Therefore, only a modest reduction in the score has been imposed.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 4.7)**

Certain assumptions are implicit in the statement of scope. However, the work plan and scope were designed so that the major assumptions would be tested experimentally to verify the validity of these assumptions. An example was the assumption that flow stress is virtually independent of the rate at which stress and strain are applied to the specimen. This assumption had its origins in earlier test work performed by others prior to the in-depth study undertaken in this project. ANL could not confirm the validity of this assumption and undertook an effort to determine why a rate effect was observed in their tests and not in the earlier tests. Other examples of implicit assumptions involved issues such as ligament linkage and its relationship to both leakage and burst pressure, the quantification of choke flow leakage through cracks with two-phase flow, and the existence of a correlation between leakage and crack growth. The investigators did not make an explicit effort to identify and justify these assumptions.

In connection with the development of failure 'maps', it is asserted that the complex ligament geometries of real cracks can be idealized as either solely axial, solely circumferential, or radial. The report does not include any discussion on how close those assumptions are to reality. As noted above, an assumption about application of correlation developed for two cracks being applicable to configurations with four and six cracks is neither articulated nor justified.

In some cases, the assumed level of familiarity with previous work limits the discussion to the extent that the bases for assumptions are not clear. For example, in the predictions of ligament rupture against the McGuire tests, the ligament rupture pressure of each test was predicted by the equivalent rectangular crack methods. There is no explanation of why this is the appropriate model. An explanation would be worthwhile given that the benchmark is only partially successful. The abstract states that this is the "latest correlation." But some additional explanation would have contributed to a better understanding.

Much of the work on main steamline break effects on damaged tubes (Ref.8) relies on analytical simulation with TRAC-M and RELAP-5 codes. The ability of these codes to model appropriately pressure drops in complex geometries such as those of steam generator tube bundles and tube support plates has been questioned. The report does not discuss this issue. There are good comparisons of results from the two codes and finite-element analysis results, but applicability of these models is an important issue that deserves some discussion.

- Soundness of technical approach and results **(Consensus score = 5.0)**

The scope of work was thorough in identifying the major steps and the technical approach to be used by the investigators. The investigators used sound scientific and engineering methods to conduct these investigations. In addition, it is clear that the investigators followed up on anomalies and results that differed from prior assumptions to gain insights into the phenomenon that they were investigating. These new insights were factored into the analytical models under development to the extent that they could be, and uncertainties were estimated for data that had a range of numerical results. The investigators stated that the models provided conservative predictions.

Though quibbles abound in the review of the technical approach, no flaws were identified that would detract from the value of the results in any major way. On the other hand, the technical approaches adopted in the following four efforts were not inspired, so no bases for higher scores were identified either.

Pressurization rate effects

The first reported task was the confirmation of claims that rupture of flawed tubes is dependent on the rate of pressurization. The approach undertaken was to test a variety of flawed tubes similar to those used by investigators making the claim of a pressurization rate effect. The testing was, however, done in a consistent fashion unlike the testing done by those making the claims.

Testing was done at pressurization rates that varied from quasi-static to greater than 69 MPa/s. This range included, apparently, the pressurization rate used by those making the claims of a pressurization rate effect. Whether it includes prototypic pressurization rates is not stated, but it appears likely that it did. Tests were done at enough pressurization rates that it should be possible to infer by interpolation results for any pressurization rate likely to be of practical interest. This appears to be a technically sound and defensible approach.

In addition, tests are planned on cracks that were formed by a stress corrosion cracking process. The results of these tests will be presumably used to relate the results of tests with machined flaws to more realistic cracks. Again, this seems a prudent and reasonable approach.

o Development of failure maps

To prepare failure maps, the authors have correlated data on the ligament ruptures of two types of flaws in tubes. A simple polynomial model has been used for correlation and it does not seem to have been selected based on some theoretical considerations. Details of the procedure for fitting the data to correlations are not spelled out to any extent. It is apparent that the polynomial is a very approximate description of the data and the parametric values must be changed for different crack lengths. Fitting apparently neglected the uncertainties in the data. Had these uncertainties been recognized, it might have been possible to use simpler correlation expressions. A similar polynomial correlation was developed for rupture pressure for the case of two cracks separated by a circumferential ligament. It appears that the data used for correlation may have come from room temperature tests, but documentation is not definitive on this point, and salient references have not yet been retrieved.

The correlations were then used to develop maps of crack length versus ligament width showing behavior for various pressure differences and crack geometries assuming 80 and 90% through-wall cracks. This approach is common and technically sound for maps involving two cracks separated by an axial or a radial ligament, provided that the correlations developed from test data are applicable at the assumed 300°C.

Maps were also prepared for cases with four and six cracks. There seems to be no demonstration that the correlations of ligament rupture and tube rupture obtained for two cracks are applicable to cases with four or more cracks. To be sure, there is an extrapolation taking place here that is not especially well highlighted in the documentation. Nevertheless, one must concede that if this extrapolation is palatable, the approach adopted in preparing the maps is a widely accepted one. Use of the maps, on the other hand, would demand a great deal more than is attempted in this limited effort. A reader would benefit from some comparison of the map predictions to data for the multiple crack cases.

Leak Rate Studies

The leak rate studies were undertaken to determine the limits of applicability with respect to the through-wall crack length and crack tightness of the simple orifice model for predicting leak rates of cracked tubes. The effort undertaken focused on conditions that will lead to "flashing" of the coolant within the crack. Crack length divided by the hydraulic diameter of the crack was used as the metric for cracks in tubes used in the tests. This is acceptable because realistic cracks are used in the test program. Analysis of the results was supplemented by data from the literature concerning flow through better instrumented slits in plates. The technical approach appears to be adequate to the task.

Results obtained in the effort only address conditions for subcooling in the range of 50-60°C . Such a subcooling range corresponds to cold leg conditions. A plausibility argument is advanced that "conservative" results will be predicted for hot leg conditions that are more appropriate for issues associated with steam generator tube leakage. Thus, results only marginally meet the objective if the objective is to find limits of applicability of the orifice model for conditions where it is likely to be of interest to apply.

Rupture and Leak Rate Predictions for McGuire Steam Generator Tubes

The technical approach for this effort involved acquisition of flawed tubes form the McGuire plant and characterization of the flaws first by nondestructive examianation methods and later by fractography. The tubes were then tested for leakage in a facility that is presumably well established and well described in some other publications. Unfortunately, no references were provided to validate this presumption. No description of the method for measuring leak rates was provided. Presumably, a well established method exists and the authors could have informed the reader about this method by means of a reference. Though poorly documented, the technical approach appears sound.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 4.3)**

The comparison of predictive models of leak rate and rupture as applied to actual tubes removed from a retired McGuire steam generator with leakage and burst test data of these tubes showed reasonable agreement. In the discussion, explanations were provided as to why the predictive models differed from the actual test results. A range of uncertainty and the degree of conservatism between the models and observed results were estimated, in order to establish the degree of usefulness of the correlations developed. Because of the complex nature of stress corrosion cracks, predictive uncertainty exists and has been estimated and factored into the resulting conclusions.

The investigators do a rather good job in developing their experimental projects in considering sensitivities such as sensitivity to the number of cracks, ligament sizes, crack orientation and the like. The investigators have not estimated uncertainties associated with any measured value that they report. Where they have fit data to a parametric correlation, they have failed to cite any uncertainties in the parametric values and certainly have not reported covariance matrices for models involving more than two parameters. They do not report on the uncertainties of predictions derived from correlations. Episodically, the authors report linear correlation coefficients that are essentially useless in the interpretation of the quality of a fit of a parameterized equation to data without a great deal more information about the fitting results.

The adequacy of the investigators' treatments of sensitivities in the development of their research efforts is acknowledged. Neglect of uncertainties in reports of measurements is the basis for reduction of the score in this category.

3.3 ANALYSIS OF ROD BUNDLE HEAT TRANSFER FACILITY TWO-PHASE INTERFACE DRAG EXPERIMENTS AT THE PENN STATE UNIVERSITY

The objective of a task at the Penn State University was to analyze data that had been collected in the Rod Bundle Heat Transfer Facility in order to gain insights to be used in the development and validation of the TRACE computer code. The specific set of data was collected to examine level swell under reflood conditions. The rod bundle in the experimental setup simulates a PWR fuel assembly with spacer grids, as in the standard 17x17 Westinghouse array. The experimental bundle involves a 7x7 array of full length, electrically heated fuel pins. The principal data collected in the experiments were the pressure drop along the length of the pins with varied reflood flow rate, power level, and inlet subcooling. Other properties of the flow, such as void fraction, interfacial drag force, and the product of interfacial area and friction factor, were determined by inference from a simplified model of energy conservation.

The data are said to be "more detailed" than previous data, but no comparisons are made to illustrate why, or to show consistency (or otherwise) with previous work.

The review is based on the only report [Ref. 9] that was provided to the Committee of results from the test program. It is entitled "Analysis of Rod Bundle Heat Transfer Test Facility Two-Phase Interfacial Drag Experiments." It has no number and is believed to be a draft. The title of the report is somewhat misleading, since there were no measurements of interfacial drag. The only parameter measured, apart from those defining the boundary conditions of the experiment, such as flow rate, power supplied etc., was the pressure drop over several lengths of a rod bundle.

The broader experimental program, which represents a substantial undertaking, with extensive measurement of parameters such as temperature, droplet size, and velocity, was not part of this review.

The Committee also had the benefit of an earlier report describing the test facility and of the RES Thermal-hydraulics Research Plan, dated March 1, 2005. RES provided a memo dated June 6, 2005 entitled, "Usage of Data from the Rod Bundle Heat Transfer Test".

The consensus scores for the project are shown in Table 4. This project marginally satisfied the research objectives. The Committee identified important deficiencies. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 4.33)**

  The report is readable and it is reasonably clear on what was done. However, the objectives of the work are not clearly stated.

Table 4 Summary Results of the ACRS Assessment of the Quality of the Project on Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	4.33	0.16	0.69
Identification of major assumptions	4.0	0.09	0.36
Justification of major assumptions	3.33	0.12	0.40
Soundness of technical approach/results	3.33	0.52	1.73
Treatment of uncertainties/sensitivities	0.66	0.11	0.07
		Overall Score:	**3.25**

Figures are mostly clear but some lack essential details. Descriptions of the location of pressure taps are inconsistent.

The report requires substantial manipulation of pressure drop data to infer void fraction, interfacial drag force, and the product of interfacial area and friction factor but the main report does not explain how these properties are obtained. The reader has to study the appendices to determine the assumptions and theory applied.

- Identification of major assumptions **(Consensus score = 4.0)**

  "Correction" of data is described but insufficiently to provide understanding of how spacers were treated, or why certain flow regimes were used to predict terms needed to convert from pressure drop to void fraction. These assumptions prejudice the eventual use for TRACE development, since they are in parallel to the comparisons with TRACE. It would be better to have TRACE predict the raw data.

  The assumption that the pressure drop does not influence fluid properties appears to be used but is not identified.

  The assumption that the only source of vapor generation is the addition of heat ignores the significant effect of flashing that is not identified.

The assumption that "the total pressure drop is small" is incorrect. Since the pressure drop along the bundle can be substantial (almost 6psi), specification of a single "pressure" (e.g. 20psia) for each experiment is inadequate without identifying clearly where it is measured.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 3.33)**

  Several inappropriate flow regimes are used.

  The energy balance is erroneous, omitting an important "flashing" term, leading to inaccurate prediction of quality.

  Property changes along the bundle due to pressure drop are ignored, though they are influenced by pressure and temperature changes.

  The effect of spacers on the flow pattern, pressure drop, and void fraction is not explained. In "correcting" the pressure drop measurements to compute a void fraction, some justification is provided for the friction pressure drop correction, but none for the acceleration pressure drop correction.

- Soundness of technical approach and results **(Consensus score = 3.33)**

  It is doubtful if the results are useful for TRACE development. There is no discussion of models currently in TRACE or direct comparison with these models.

  The presentation and reduction of data contain errors and there is no investigation of the effects of assumptions.

  Several of the comparisons with theory are inappropriate. There is no critical examination of features of the data, such as large fluctuations in the pressure drop data and the apparent lack of steady state in some tests.

  Since the intent of the report is to derive interfacial drag, there should be more information on how this was done, the sources of error, the effect of parameters, the effect of spacers, etc. Only one example is given, and it appears to have a basic flaw, since the large spikes of extreme values that are predicted indicate the flow to be close to homogeneous, which is inconsistent with evidence provided by the void fraction results.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 0.66)**

  There is no treatment or discussion of uncertainties.

4. REFERENCES

1. Letter Dated November 18, 2004, from Mario V. Bonaca, Chairman, ACRS, to Carl J. Paperiello, Director, Office of Nuclear Regulatory Research, NRC, Subject: ACRS Assessment of the Quality of Selected NRC Research Projects.

2. National Research Council, *Understanding Risk: Informing Decisions in a Democratic Society.* National Academy Press, Washington, DC, 1996.

3. Apostolakis, G.E., and Pickett, S.E., "Deliberation: Integrating Analytical Results into Environmental Decisions Involving Multiple Stakeholders," *Risk Analysis*, 18:621-634, 1998.

4. Clemen, R., *Making Hard Decisions*, 2nd Edition, Duxbury Press, Belmont, CA, 1995.

5. Keeney, R.L., and H. Raiffa, *Decisions with Multiple Objectives: Preferences and Value Tradeoffs*, Wiley, New York, 1976.

6. Eide, S. A., T.E. Wierman, and D.M. Rasmuson, " Station Blackout Risk evaluation for Nuclear Power Plants," Draft, NUREG/CR, INEEL/EXT-04-02525, Idaho National Engineering and Environmental Laboratory, January 2005.

7. Majumdar, S., K. Kasza, S. Bakhtiari, J. Oras, J. Franklin, and C. Vulyak, Jr., "Pressurization Rate Effect on Flawed Tube Rupture, Failure Maps for Complex Multiple Cracks, Validation of Leak Rate Correlation Model and Rupture and Leak Rate Predictions for McGuire Steam Generator Tubes," Draft, NUREG/CR-xxxx, Argonne National Laboratory, April 2004.

8. Majumdar, S., K. Kasza, J. Oras, J. Franklin and C. Vulyak, Jr., "Sensitivity Studies of Failure of Steam Generator Tubes during Main Steam Line Break and Other Secondary Side Depressurization Events," Draft, NUREG/CR-xxxx, Argonne National Laboratory , April 2004.

9. Hochreiter, L. E., F. B. Cheung, T. F. Lin, and D.J. Miller, "Analysis of Rod Bundle Heat Transfer Test Facility Two-Phase Interfacial Drag Experiments," The Pennsylvania State University, June 2005.

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

January 17, 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 POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

Attached is a copy of the EDO's December 21, 2005 response to the ACRS's November 18, 2005 report on the safety aspects of the license renewal application for the Point Beach Nuclear Plant (PBNP) Units 1 and 2. A copy of the Committee's report is also attached.

COMMITTEE REPORT

The Committee recommended that the license renewal application for PBNP Units 1 and 2 be approved under the condition that the staff perform additional actions to ensure that the requirements of the license renewal rule have been met. These additional actions are (1) expanding the scope of its post-approval site inspection to verify that all license renewal programs have been implemented and commitments have been met and (2) performing a review of the effectiveness of the PBNP Corrective Action Program (CAP) before the plant enters the period of extended operation.

EDO RESPONSE

The staff believes the Committee's concerns will be adequately addressed with Inspection Procedures (IP) 71003 "Post-Approval Site Inspection for License Renewal" and 71152 "Identification and Resolution of Problems."

The EDO response stated that the region has considerable latitude in determining the scope of IP 71003, and Region III will consider expanding the scope of its inspection of license renewal commitments at PBNP. Region III is also considering inspecting some of the aging management programs through the normal baseline inspection process.

The EDO response also stated that in accordance with IP 71152 the NRC evaluates CAP effectiveness during Problem Identification and Resolution (PI&R) baseline inspections. These inspections will ensure that issues with the CAP will not prevent PBNP from implementing license renewal programs and meeting commitments. Region III plans to perform at least two PI&R inspections before PBNP Unit 1 enters the period of extended operation and additional PI&R inspections before Unit 2 enters the period of extended operation. Region III also plans to spend at least 100 hours of inspection on special reviews of the PBNP CAP after the original red findings are closed out.

ANALYSIS

The EDO response is partially satisfactory. The Committee's conditions for approving the PBNP license renewal application were that the staff expand the scope of its post-approval site inspection and perform a review of the effectiveness of the PBNP CAP. Region III is only considering expanding the scope of its inspection of PBNP license renewal commitments but will evaluate CAP effectiveness by performing PI&R inspections before PBNP enters the period of extended operation. The NRC renewed the licenses for PBNP Units 1 and 2 on December 22, 2005.

Attachments: As stated

cc: w/Attachments : ACRS Members
J. Larkins
A. Thadani
M. Scott
M. Snodderly
J. Lamb
S. Duraiswamy

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

December 21, 2005

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

Dear Dr. Wallis:

In your letter to Chairman Diaz dated November 18, 2005, you summarized the results of the final review by the Advisory Committee on Reactor Safeguards (ACRS) of the Nuclear Management Company's license renewal application (LRA) for Point Beach Nuclear Plant (PBNP), Units 1 and 2, and the associated final safety evaluation report prepared by the U.S. Nuclear Regulatory Commission (NRC) staff.

On the basis of its review, the ACRS concluded that all open and confirmatory items had been resolved, and agreed that the applicant's proposed aging management programs (AMPs) are adequate. The ACRS recommended renewing PBNP's license with the inclusion of one condition. The Committee recommends expanding the scope of the license renewal post-approval site inspection to verify that all license renewal programs have been implemented and commitments met. In addition, the Committee recommended that the staff review the effectiveness of the PBNP corrective action program (CAP) before entering the period of extended operation. The staff understands the Committee's concern about PBNP's ability to implement its AMPs and LRA commitments.

The Region III staff will consider expanding the scope of its inspection of license renewal commitments at PBNP just as it would at any other plant whose performance at the time of extended operation warrants increased inspection effort. The license renewal is conducted in accordance with Inspection Procedure (IP) 71003, "Post-Approval Site Inspection for License Renewal," which gives the regional staff considerable latitude in broadening the scope of the inspection. In addition, the regional administrator is authorized to determine if further inspections are needed. Outstanding commitments will be discussed with the NRC headquarters technical staff and appropriate actions, including enforcement, will be taken as needed. The regional staff is also considering inspecting some AMPs through the normal baseline inspection process.

The Committee recommended that the staff review the effectiveness of the CAP before PBNP enters the period of extended operation. The staff agrees that an adequate CAP is a key element in the successful implementation of the AMPs and that the staff should assess its effectiveness. In accordance with IP 71152, "Identification and Resolution of Problems," the NRC evaluates CAP effectiveness during biennial problem identification and resolution (PI&R) baseline inspections. IP 71152 also requires corrective action followup on three to six issues a year. The Region III staff will perform at least two biennial PI&R inspections at PBNP before

Unit 1 enters the period of extended operation. Additional PI&R inspections will be conducted before Unit 2 enters the period of extended operation.

During these PI&R inspections, the staff will evaluate the licensee's ability to identify and correct problems, including problems related to license renewal. These PI&R inspections will ensure that CAP issues will not prevent the licensee from implementing its license renewal commitments and AMPs. In addition, as stated in our letter dated July 15, 2005, and consistent with Inspection Manual Chapter 0305, "Operating Reactor Assessment Program," the Region III staff plans to spend at least 100 hours of inspection on special reviews of the licensee's CAP, after the original red findings have been closed out. This will increase confidence that the licensee is appropriately detecting and correcting problems before PBNP enters the period of extended operation.

During the past year and a half, the staff performed several onsite audits and inspections to ensure that the applicant addressed all items required by the license renewal rule. The staff believes that the combination of the IP 71003 and the PI&R inspections will adequately address the Committee's concern. The staff is confident that the NRC Reactor Oversight Process will ensure that PBNP is operated in accordance with its current licensing basis during the period of extended operation without undue risk to the health and safety of the public.

Once again, the staff recognizes the ACRS's commitment to safety and appreciates the Committee's continued efforts in 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-2165

November 18, 2005

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 POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

Dear Chairman Diaz:

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, we completed our review of the license renewal application for the Point Beach Nuclear Plant (PBNP) Units 1 and 2, and the final Safety Evaluation Report (SER) prepared by the NRC staff. We issued an interim report on the safety aspects of this application and the draft SER on June 9, 2005. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on May 31, 2005. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Management Company, LLC (NMC). 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.

RECOMMENDATIONS

1. With the inclusion of the conditions in Recommendation 2, the NMC application for license renewal of PBNP Units 1 and 2 should be approved.

2. The staff should expand the scope of its post-approval site inspection to verify that all license renewal programs have been implemented and commitments have been met. In addition, the staff should review the effectiveness of the PBNP corrective action program (CAP) before PBNP enters the period of extended operation.

BACKGROUND AND DISCUSSION

The PBNP Units 1 and 2 are two-loop Westinghouse pressurized water reactors housed in dry ambient containments. Originally, each unit was licensed at a power level of 1519 MWt. Each unit has undergone a low-pressure turbine modification and a measurement uncertainty recapture power uprate to increase the power level to 1540 MWt. NMC has requested renewal of the operating licenses of Units 1 and 2 for 20 years beyond their current license terms, which expire on October 5, 2010, and March 8, 2013, respectively.

In the final SER, the staff documents its review of the license renewal application and other information submitted by the applicant and obtained through the audits and inspections 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; and the identification and assessment of time-limited aging analyses (TLAAs).

The PBNP application demonstrates consistency with, or documents deviations from, the approaches specified in the Generic Aging Lessons Learned Report. The staff questioned the applicant's approach to identifying nonsafety-related components whose failure could affect safety-related components. The applicant modified its scoping methodology to address the staff's questions. An inspection completed on August 17, 2005 confirmed that this methodology has been appropriately implemented. In the final SER, the staff concludes that the scoping and screening processes implemented by the applicant have successfully identified SSCs within the scope of license renewal and subject to an aging management review. We agree with this conclusion.

The applicant performed a comprehensive aging management review of all SSCs within the scope of license renewal. In the application, the applicant describes 26 aging management programs for license renewal, including existing, enhanced, and new programs. The draft SER identified 5 open items and 15 confirmatory items. The final SER describes the resolution of these items. We agree with the resolution of these items and with the staff's conclusion that the applicant's proposed aging management programs are adequate.

One of the open items relates to plant-specific operating experience of the two units. Containment liner corrosion due to borated water leakage has been identified in both units. The applicant has committed to performing augmented inspections in accordance with ASME Section XI Subsection IWE to monitor the extent of corrosion. The Boric Acid Corrosion Program is also credited with assessing and managing loss of material in the containment liner. The augmented inspection program does not include specific criteria for evaluation, repair, or replacement. At the staff's request, the applicant has agreed to include in the acceptance criteria element of the aging management program, "ASME Section XI, Subsections IWE and IWL Inservice Inspection Program," an appropriate discussion of the evaluation, repair or replacement criteria, and reexamination requirements necessary to ensure leak-tightness and structural integrity of the liner.

The applicant identified and reevaluated systems and components requiring TLAAs for 20 more years of operation. The upper shelf energy for both vessels and the reference temperature for pressurized thermal shock (PTS) for the Unit 2 vessel failed to meet the screening criteria.

To address the low upper shelf energy, the applicant performed equivalent margin analyses allowed by 10 CFR Part 50, Appendix G. These analyses yielded acceptable results through the end of the period of extended operation. The staff performed independent analyses to confirm the applicant's conclusion.

The intermediate-to-lower shell circumferential weld of the Unit 2 vessel is projected to exceed the PTS screening criterion in 2017. Consistent with the requirements of 10 CFR 54.21(c)(1)(iii), the applicant has chosen to manage the effects of aging of this weld during the period of extended operation. The applicant's commitments for PTS include implementing a low-low leakage fuel management pattern, using hafnium absorber assemblies, and documenting a flux reduction plan. This documentation will include any required safety analyses supporting continued operation. Other options the applicant may pursue include a more refined analysis of PTS or thermal annealing of the reactor pressure vessel.

In our June 9, 2005 interim report on the PBNP application, we expressed concern with the effectiveness of the PBNP CAP and the applicant's ability to effectively implement license renewal programs and meet commitments. We were concerned that the resources needed to address the staff's April 21, 2004 Confirmatory Action Letter to PBNP would compete with the effective development, tracking, and implementation of license renewal programs and commitments. We recommended that, prior to the units entering the period of extended operation, the staff take additional actions to increase confidence that the requirements of the license renewal rule have been met. We suggested, for example, an expanded inspection of license renewal commitments and a focused review of the effectiveness of the CAP. The PBNP remains in the Multiple/Repetitive Degraded Cornerstone column of the Reactor Oversight Process Action Matrix, and there are still weaknesses in the CAP.

In its July 15, 2005 response to the Committee, the staff described the inspections being conducted at PBNP to verify that license renewal programs and commitments are appropriate and consistent with the rule. However, detailed development and implementation of many of these programs and commitments will occur after the license is renewed and prior to the license renewal period. The staff plans to perform a post-approval site inspection in accordance with Inspection Procedure 71003 before the period of extended operation begins.

Inspection Procedure 71003 is the standard inspection that the staff performs prior to the period of extended operation. This inspection evaluates only a sample of the license renewal commitments and programs. In light of the applicant's weakness in managing commitments, as discussed in our interim report, the staff should expand the scope of the post-approval site inspection to verify that all license renewal programs have been implemented and commitments have been met. In addition, before PBNP enters the period of extended operation, the staff should review the effectiveness of the CAP. These actions are necessary to ensure that there is reasonable assurance that aging degradation can be adequately managed.

With a commitment to perform the expanded inspections described above, the application for renewal of the operating licenses of the PBNP Units 1 and 2 should be approved.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. Nuclear Management Company, LLC, "Application for Renewed Operating Licenses Point Beach Nuclear Plant Units 1 & 2," February 2004.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2," May 2005.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2," October 2005.
4. Letter from Graham B. Wallis, Chairman, ACRS, to Luis A. Reyes, Executive Director for Operations, NRC, "Interim Report on the Safety Aspects of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2," June 9, 2005.
5. Pacific Northwest National Laboratory, "Audit and Review Report for Plant Aging Management Reviews and Programs, Point Beach Nuclear Plant Units 1 and 2," April 11, 2005.
6. U.S. Nuclear Regulatory Commission, "Point Beach Nuclear Plant, Units 1 and 2 NRC License Renewal Scoping, Screening, and Aging Management Inspection Report 05000266/2005005 (DRS); 05000301/2005005 (DRS)," May 2, 2005.
7. U.S. Nuclear Regulatory Commission, "Point Beach Nuclear Plant, Units 1 and 2 NRC License Renewal Followup Inspection Report 05000266/2005015 (DRS); 05000301/2005015 (DRS)," September 9, 2005.
8. U.S. Nuclear Regulatory Commission, "Point Beach Nuclear Plant, Units 1 and 2 NRC Special Inspection Report 05000266/2005011; 05000301/2005011," September 23, 2005.
9. U.S. Nuclear Regulatory Commission, "Point Beach Nuclear Plant, Units 1 and 2 NRC Special Emergency Preparedness Inspection Report 05000266/2005009 (DRS); 05000301/2005009 (DRS)," August 2, 2005.

10. Letter from J. Dyer, Regional Administrator, to M. Warner, Site Vice President, Kewaunee and Point Beach Nuclear Plants, Nuclear Management Company, LLC, "Point Beach Special Inspection - NRC Inspection Report 50-266/01-17(DRS); 50-301/01-17(DRS), Preliminary Red Finding," April 3, 2002.
11. Letter from J. Dyer, Regional Administrator, to M. Warner, Site Vice President, Kewaunee and Point Beach Nuclear Plants, Nuclear Management Company, LLC, "Point Beach Nuclear Plant Final Significance Determination for a Red Finding and Notice of Violation NRC Special Inspection Report No. 50-266/01-17(DRS; 50-301/01-17(DRS)," July 12, 2002.
12. Letter from J. Dyer, Regional Administrator, to A. Cayia, Site Vice President, Point Beach Nuclear Power Plant, Nuclear Management Company, LLC, "Point Beach Nuclear Plant Special Inspections: Resolution of Auxiliary Feedwater Old Design Issue and Preliminary Red Finding - Auxiliary Feedwater Orifice Plugging Issue; NRC Inspection Report 50-266/02-15(DRP); 50-301/02-15(DRP)," April 2, 2003.
13. Letter from J. Caldwell, Regional Administrator, to A. Cayia, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, "Point Beach Nuclear Plant, Units 1 and 2 Final Significance Determination for a Red Finding and Notice of Violation (NRC Inspection Report No. 50-266/02-15(DRP); 50-301/02-15(DRP))," December 11, 2003.
14. Letter from G. Van Middlesworth, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, to U.S. Nuclear Regulatory Commission Document Control Desk, "Commitments in Response to 95003 Supplemental Inspection," March 22, 2004.
15. Letter from J. Caldwell, Regional Administrator, to G. Van Middlesworth, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, "Confirmatory Action Letter," April 21, 2004.
16. Letter from J. Caldwell, Regional Administrator, to D. Koehl, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, "Annual Assessment Letter - Point Beach Nuclear Plant (Report 05000266/200501; 05000301/200501)," March 2, 2005.
17. Letter from D. Koehl, Site Vice-President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, to U.S. Nuclear Regulatory Commission Document Control Desk, "License Renewal Application Revised Information," September 10, 2004.
18. Memorandum from L. Reyes, EDO, to Chairman Diaz, Commissioner McGaffican, and Commissioner Merrifield, "Pressurized Thermal Shock Analyses for Renewal of Certain Nuclear Power Plant Operating Licenses," May 27, 2004.

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

February 3, 2006

MEMORANDUM TO: Rich Denning, Chairman
Fire Protection Subcommittee

FROM: John G. Lamb, Senior Staff Engineer
Advisory Committee on Reactor Safeguards Staff

John G. Lamb

SUBJECT: ANALYSIS OF EDO RESPONSE TO THE ACRS LETTER, DATED NOVEMBER 18, 2005, CONCERNING THE STAFF RECOMMENDATION TO WITHDRAW THE PROPOSED RULE ON POST-FIRE OPERATOR MANUAL ACTIONS

Attachment 1 contains a copy of the Executive Director for Operations (EDO) December 23, 2005 response (ADAMS Accession No. ML053340063) to the Advisory Committee on Reactor Safeguards (ACRS) November 18, 2005 letter regarding the staff recommendation to withdraw the proposed rule on post-fire operator manual actions. Attachment 2 contains a copy of the Committee letter.

The conclusions and recommendations in the ACRS letter are as follows:

- The proposed rule on post-fire operator manual actions would not satisfy the objective of significantly reducing the number of future exemption requests.

- We concur with the staff's decision to withdraw the proposed rule.

EDO RESPONSE

Since the staff position on this topic is the same as that of the ACRS, the EDO has no comments on the ACRS conclusions.

ANALYSIS

The EDO response is satisfactory.

Attachments:
1. Letter from the EDO to G. Wallis, ACRS, dated December 23, 2005, Subject: Staff Recommendation to Withdraw the Proposed Rule on Post-Fire Operator Manual Actions (ADAMS Accession No. ML053340063)
2. Letter from G. Wallis, ACRS, to the EDO, dated November 18, 2005, Subject: Staff Recommendation to Withdraw the Proposed Rule on Post-Fire Operator Manual Actions (ADAMS Accession No. ML053250543)

cc: ACRS Members
J. Larkins
A. Thadani
M. Scott
M. Snodderly
S. Duraiswamy

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

December 23, 2005

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

SUBJECT: STAFF RECOMMENDATION TO WITHDRAW THE PROPOSED RULE ON POST-FIRE OPERATOR MANUAL ACTIONS

Dear Dr. Wallis:

I am responding to your November 18, 2005, letter on the staff recommendation to withdraw the proposed rule on post-fire operator manual actions. The Advisory Committee on Reactor Safeguards (ACRS) concluded that the proposed rule on post-fire operator manual actions would not satisfy the objective of significantly reducing the number of future exemption requests and agreed with the staff decision to withdraw the proposed rule.

Since the Nuclear Regulatory Commission staff position on this topic is the same as that of the ACRS, we have no comments on the ACRS conclusions. We appreciate the time and effort the Committee has devoted to this subject. We will continue to work closely with the ACRS on future fire protection issues.

Sincerely,

Jacqueline E. Silber 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, DC 20555 - 0001

ACRSR-2166

November 18, 2005

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

SUBJECT: STAFF RECOMMENDATION TO WITHDRAW THE PROPOSED RULE ON POST-FIRE OPERATOR MANUAL ACTIONS

Dear Chairman Diaz:

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, we discussed the staff's recommendation to withdraw the proposed rule on post-fire operator manual actions. During our review, we had the benefit of discussions with representatives of the staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

- The proposed rule on post-fire operator manual actions would not satisfy the objective of significantly reducing the number of future exemption requests.
- We concur with the staff's decision to withdraw the proposed rule.

DISCUSSION

A proposed rule that would modify Appendix R of 10 CFR 50 to include the regulation of post-fire operator manual actions was issued for public comment on March 7, 2005. After evaluating the public comments, the staff concluded that the final rule would not achieve the objective of reducing the number of exemption evaluations required and that it should be withdrawn.

Section III.G of Appendix R provides requirements that assure the protection of at least one path of achieving safe shutdown during a fire at any location in the plant. Plants that received their licenses after 1979 are not subject to Appendix R, but comply with similar requirements. Because the plants to which Appendix R applies were constructed in the absence of standards addressing separation and protection, some fire areas contain equipment from more than one safe shutdown train. Section III.G.2 of Appendix R identifies three alternative means of protecting at least one train of safe shutdown equipment within a fire area:

- A 3-hour rated fire barrier (for fire areas outside containment)
- Separation by at least 20 feet with no intervening material, in combination with fire detection and automatic fire suppression equipment
- Enclosure of one train of equipment with a 1-hour rated fire barrier, in combination with detection and automatic fire suppression equipment.

Some plants have had difficulty in complying with Section III.G.2 and have sought exemptions in which operator manual actions compensate for an inability to satisfy one of the alternatives. Some plants relied on compensatory operator manual actions without receiving regulatory approval. To achieve compliance, either plants can obtain exemption from Section III.G.2 requirements or the requirements can be modified by rulemaking to cover those conditions for which manual actions represent an acceptable alternative. The staff developed the proposed rule for this purpose.

In our letter dated November 19, 2004, we recommended that the draft rule be published for public comment. In approving publication of the proposed rule, the Commission directed the staff to "engage stakeholders to get a clear understanding of the likelihood that the proposed rule would achieve its underlying purpose, including the number of plants for which the proposed rule would address the operator manual actions issue. This information should be considered in deciding whether to proceed to final rulemaking."

Comments were received from the public, licensees, and the Nuclear Energy Institute. Based on its evaluation of the comments, the staff has concluded that the proposed rule would not lead to a significant reduction in the number of exemption requests. We concur with the staff's recommendation to withdraw the proposed rule.

In the absence of the final rule, the staff will proceed with enforcement of the existing regulations and the case-by-case resolution of exemption requests. An alternative available to licensees is to transition to a risk-informed fire protection program under 10 CFR 50.48(c). Appendix R sets forth an established deterministic approach for assuring the ability to safely shut down a nuclear plant during a fire. However, when a licensee seeks an exemption from Appendix R, risk insights may be useful to determine that adequate safety is preserved.

Sincerely,

/RA/

Graham B. Wallis
Chairman

<u>References</u>:

1. Memorandum from J. Lyons, NRR, to J. Larkins, ACRS, dated October 28, 2005, "Proposed Withdrawal of Rulemaking Allowing Use of Post-Fire Operator Manual Actions," (ADAMS Accession No. ML052970102).

2. Letter from M. Bonaca, ACRS, to N. Diaz, Chairman, dated November 19, 2004, "Draft Proposed Rule on Post-Fire Operator Manual Actions," (ADAMS Accession No. ML043240215).

3. Memorandum from E. Merchoff acting for EDO, to M. Bonaca, ACRS, dated December 22, 2004, "Draft Proposed Rule on Post-Fire Operator Manual Actions," (ADAMS Accession No. ML043380177).

4. Staff Requirements, SECY-04-0233 - Proposed Rulemaking - Post-Fire Operator Manual Actions, January 18, 2005.

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

February 3, 2006

MEMORANDUM TO: John D. Sieber, Chairman
Plant Operations Subcommittee

FROM: John G. Lamb, Senior Staff Engineer
Advisory Committee on Reactor Safeguards Staff

SUBJECT: ANALYSIS OF EDO RESPONSE TO THE ACRS LETTER, DATED NOVEMBER 18, 2005, CONCERNING DRAFT FINAL GENERIC LETTER 2005-XX, "GRID RELIABILITY AND THE IMPACT ON PLANT RISK AND THE OPERABILITY OF OFFSITE POWER"

Attachment 1 contains a copy of the Executive Director for Operations (EDO) December 23, 2005 response (ADAMS Accession No. ML053480114) to the Advisory Committee on Reactor Safeguards (ACRS) November 18, 2005 letter regarding draft final Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power." Attachment 2 contains a copy of the Committee letter.

The recommendation in the ACRS letter is as follows:

- Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," should be issued.

The ACRS letter stated: "The staff may need to explore these same questions [in the Generic Letter] with licensees after the Electric Reliability Organization is established and functioning, and the electric reliability standards are approved and in full force and effect. Also, the staff should continue to interact with [Federal Energy Regulatory Commission] FERC and [North American Electric Reliability Council] NERC on grid reliability issues. We would like to hear a briefing from the staff after it has evaluated the information submitted by the licensees in response to this Generic Letter."

EDO RESPONSE

The responses in the EDO letter are as follows:

- The staff will consider exploring the grid reliability issues stated in the Generic Letter with the licensees after the electric reliability standards are approved and in effect.
- The staff will continue to work with FERC and NERC on grid reliability matters as suggested in your letter to ensure a reliable offsite power system for the nuclear power plants.
- We will brief the ACRS after the staff has evaluated the information submitted by the licensees in response to the subject Generic Letter.

ANALYSIS

The EDO response is satisfactory. The EDO agrees with the ACRS recommendations.

Attachments:
1. Letter from the EDO to G. Wallis, ACRS, dated December 23, 2005, Subject: Draft Final Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power" (ADAMS Accession No. ML053480114)
2. Letter from G. Wallis, ACRS, to the EDO, dated November 18, 2005, Subject: Draft Final Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power" (ADAMS Accession No. ML053250539)

cc: ACRS Members
J. Larkins
A. Thadani
M. Scott
M. Snodderly
S. Duraiswamy

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

December 23, 2005

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

SUBJECT: DRAFT FINAL GENERIC LETTER 2005-XX, "GRID RELIABILITY AND THE IMPACT ON PLANT RISK AND THE OPERABILITY OF OFFSITE POWER"

Dear Dr. Wallis:

This is in response to your letter dated November 18, 2005, which summarized the results of the Advisory Committee on Reactor Safeguards' review of the subject draft final generic letter (GL). We appreciate the Committee's comments and acknowledge your suggestions.

In response to your recommendation, the staff will consider exploring the grid reliability issues stated in the GL with the licensees after the electric reliability standards are approved and in effect. Also, the staff will continue to work with the Federal Energy Regulatory Commission (FERC) and the North American Electric Reliability Council (NERC) on grid reliability matters as suggested in your letter to ensure a reliable offsite power system for the nuclear power plants.

We will brief the Advisory Committee on Reactor Safeguards after the staff has evaluated the information submitted by the licensees in response to the subject GL.

Sincerely,

Jacqueline E. Silber 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, DC 20555 - 0001

ACRSR-2167

November 18, 2005

Luis A. Reyes
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL GENERIC LETTER 2005-XX, "GRID RELIABILITY AND THE IMPACT ON PLANT RISK AND THE OPERABILITY OF OFFSITE POWER"

Dear Mr. Reyes:

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, we reviewed the draft final Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power." During our review, we had the benefit of the document referenced and discussions with representatives of the staff and the Nuclear Energy Institute.

RECOMMENDATION

Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," should be issued.

BACKGROUND AND DISCUSSION

The blackout of much of the Northeastern United States and parts of Canada on August 14, 2003, highlighted the extent to which changed conditions in the electric utility infrastructure could affect the probability of a station blackout event at nuclear power plants (NPPs). During the August 14 2003, event, nine NPPs lost all offsite power for periods ranging from 1 hour to 6.5 hours. Emergency diesel generators at these plants started and operated to supply emergency power, as designed. Adequate core cooling was maintained at all plants. Nonetheless, this event was significant because of the number of plants affected and the duration of the power outage. The severity and duration of this event called into question the bases for determining the risk impacts to the fleet of NPPs due to grid reliability issues.

Concerns about the reliability of the Nation's electrical grid prompted the U.S. Congress to enact the Electricity Modernization Act of 2005, which was signed on August 8, 2005. This Law added Section 215 to the Federal Power Act (FPA). Section 215 requires the Federal Energy Regulatory Commission (FERC) to enact regulations to improve and enforce the reliability of the electric power transmission infrastructure. FERC is currently amending its regulations to implement the requirements of the amended FPA. Among the changes under the amended FPA, FERC is charged with approving enforceable reliability standards. The North American Electric Reliability Council (NERC) is currently developing these reliability standards. The establishment of a national Electric Reliability Organization and the implementation of enforceable grid reliability standards are expected to be completed by December 31, 2006. The NRC has entered into a Memorandum of Agreement with both FERC and NERC which allows the staff to observe and participate in this important ongoing work. The continued cooperation between the staff and FERC and NERC is important in achieving the objectives of enhanced grid reliability without duplication of effort or conflicting goals, rules, or strategies. This cooperation should continue until satisfactory resolution of the grid reliability issue is achieved.

Even though FERC is taking important steps to improve grid reliability, the staff is rightly concerned as to how licensees are operating their NPPs in compliance with the rules and technical specifications relevant to grid operability. The NRC staff has developed Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," to obtain information needed to assess whether licensees are in compliance with technical specifications, license conditions, and regulations regarding the operability and reliability of offsite power sources. Specifically, the Generic Letter requests licensees to provide detailed information, under oath or affirmation, regarding the details of their compliance with the following regulations:

- 10 CFR 50.63 (Station Blackout Rule)
- 10 CFR 50.65 (Maintenance Rule)
- 10 CFR Part 50, Appendix A, General Design Criterion 17 (Electric Power Systems)
- Technical Specification 3.8.1 (Operability of Offsite Power Systems)

The questions posed in the Generic Letter are appropriate and the staff should issue the Generic Letter to the licensees. The staff may need to explore these same questions with licensees after the Electric Reliability Organization is established and functioning, and the electric reliability standards are approved and in full force and effect. Also, the staff should continue to interact with FERC and NERC on grid reliability issues. We would like to hear a briefing from the staff after it has evaluated the information submitted by the licensees in response to this Generic Letter.

Sincerely,

/RA/

Graham B. Wallis
Chairman

Reference:
Memorandum from M. Mayfield, NRR, to J. Larkins, ACRS, dated October 6, 2005, Subject: Request for Review and Endorsement by the Advisory Committee on Reactor Safeguards (ACRS) Regarding the Proposed Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," (ADAMS Accession No. ML052790683).

Tab 11

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

January 17, 2006

MEMORANDUM TO: George E. Apostolakis, Chair
ACRS Digital Instrumentation & Control Systems Subcommittee

FROM: E. Thornsbury, Senior Staff Engineer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON THE DRAFT NRC DIGITAL SYSTEM RESEARCH PLAN FOR FY 2005 - FY 2009

Attached is a copy of the EDO's December 23, 2005 letter of response to the ACRS's November 21, 2005 report on the Committee's review of the Draft NRC Digital System Research Plan for FY 2005 - FY 2009. A copy of the Committee's November 21, 2005 letter is also attached.

Committee Letter

In its letter, the Committee concluded that the plan is well directed toward meeting agency needs. The Committee recommended refinements to the plan through addition of a research project to develop an inventory and classification of digital systems likely to be used in nuclear power plants, inclusion of a more detailed identification of regulatory needs and benefits of the research, more consideration of the "system-centric" aspects of software safety, and higher priority on advanced nuclear power plant digital systems.

EDO Response

In the EDO's response letter, the staff agrees with all of the Committee's recommendations.. The staff plans to expand the research project in Section 3.3.1 of the plan to include development of an inventory and classification system as recommended. The staff plans to better identify regulatory needs and anticipated benefits across all research areas. The staff believes the research gives equal weight to the two aspects of software safety, and plans to ensure that the system-centric approach is more apparent in the plan. Finally, the staff plans to conduct research related to advanced nuclear power plant digital systems with a high priority once the design information becomes available.

Analysis

The EDO's response is satisfactory. The staff plans to incorporate the Committee's recommendations into the plan prior to its issuance. The staff plans to continue to interact with the Committee as work progresses under the research plan.

cc: ACRS Members
SDuraiswamy
MSnodderly
MScott

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

December 23, 2005

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

SUBJECT: RESPONSE TO ACRS LETTER, DATED NOVEMBER 21, 2005, ON THE DRAFT NRC DIGITAL SYSTEM RESEARCH PLAN FOR FY 2005 – FY 2009

Dear Dr. Wallis:

Thank you for your letter, dated November 21, 2005, in which the Advisory Committee on Reactor Safeguards (ACRS or the Committee) conveyed its views and recommendations regarding the "Draft NRC Digital System Research Plan for FY 2005 – FY 2009." As noted in your letter, the U.S. Nuclear Regulatory Commission (NRC) faces a number of challenges in licensing digital technology for safety applications in nuclear facilities. The programs outlined within this Research Plan should help to address these challenges by providing important inputs to the agency's regulatory process.

In response to the Committee's recommendations regarding the Research Plan, the NRC staff provides the following comments:

Recommendation 1:

The plan should include a research project to develop an inventory and classification, e.g., by function, of these various types of digital systems that are used and are likely to be used in nuclear power plants in the future.

The staff agrees with this recommendation. The Draft NRC Digital System Research Plan includes ongoing projects to identify and investigate digital systems that are likely to be used in future applications (Section 3.5, "Emerging Digital Technology and Applications"), as well as projects to collect and review failure data associated with digital systems (Section 3.3.1, "Development and Analysis of Digital System Failure Data"). In addition, the staff will expand the project in Section 3.3.1, to include research to develop an inventory and classification of the various types of digital systems that are used or likely to be used in nuclear power plants in the future. The staff will then use this classification, along with a concurrent examination of the failures that have occurred in digital systems, as appropriate, to provide information regarding the types of tools that may be best-suited for different digital systems assessments.

Recommendation 2:

The research plan should include a more detailed identification of current and future regulatory needs and possible benefits of the planned research to the regulatory system.

requires the agency to update its knowledge base frequently, and new methods and acceptance criteria are needed to assess the safety and security of the systems.

The Office of Nuclear Regulatory Research has developed a plan for digital instrumentation and control systems research for Fiscal Years FY 2005 - FY 2009. This plan updates the previous plan for Fiscal Years FY 2000 - FY 2004. The plan has been reviewed by the Office of Nuclear Reactor Regulation, the Office of Nuclear Material Safety and Safeguards, and the Office of Nuclear Security and Incident Response.

DISCUSSION

The draft plan divides the research into six areas:

- System aspects of digital technology
- Software quality assurance
- Risk assessment of digital systems
- Security aspects of digital systems
- Emerging digital technology and applications
- Advanced nuclear power plant digital systems

The proposed research areas are comprehensive.

The applicability of the methods being investigated can vary greatly across the spectrum of possible systems. There is, therefore, a need for an inventory and classification, e.g., by function, of the various types of digital systems that are used or likely to be used in nuclear power plants in the future. Such a classification, along with a concurrent examination of the failures that have occurred in digital systems, should provide information on what types of tools may be best suited for different assessments. This classification could be the key to understanding the limitations of current methods of assessment and to guiding future efforts. For example, the analytical tools required to evaluate the performance of systems with simple actuation software are expected to be simpler than those required to evaluate systems with feedback and control software.

The plan discusses the shortcomings of the current regulations and the potential improvements that the proposed research is expected to produce. The plan would benefit by better identifying regulatory needs and anticipated benefits across all research areas. During our meetings, it was evident that the staff had thought through most of these issues, but its thinking was not well documented in the plan. Such documentation should be included.

As stated in the additional comments to our June 9, 2004, letter, the literature on digital software indicates that there are two main approaches to software reliability. The first approach views "failure" as a property of the software itself, just as the failure modes of hardware are considered properties of the components. This first approach is "software-centric." The second approach is "system-centric," in that the software is considered part of the system and the focus is on system failures.

Although the staff is aware of the two approaches to digital system reliability, the plan appears to be heavily focused on the software-centric view. For example, one objective of the research project described in Section 3.3.3, "Investigation of Digital System Characteristics Important to Risk," is said to be the calculation of the risk-importance of generic digital systems. This project seems to focus on the software more than the overall system. Although such a calculation may be meaningful for software in actuation systems such as the reactor protection system, it is unclear whether this can be done in more complex cases. Similarly, the term "digital system reliability" is used repeatedly in Section 3.3.4, "Investigation of Digital System Reliability Assessment Methods." A system-centric analysis focuses on the reliability of the broader system, not just the digital part. Such an approach to reliability should receive more consideration in the plan. The digital system classification in Recommendation 1 will assist the staff in determining when each approach is appropriate.

The research plan includes a program to investigate advanced nuclear power plant digital systems (Section 3.6), but this work has not begun. Due to the rapidly increasing interest in new reactors and the anticipated regulatory needs, this research should be given higher priority than it currently has.

In conclusion, we found the Digital System Research Plan for FY 2005 - FY 2009 to be well developed. The planned research programs should provide important inputs to the regulatory process. We look forward to continuing discussions with the staff on these programs as work progresses.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. Memorandum from Michelle G. Evans, Chief, Engineering Research Applications Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "Transmittal of Material to Support the November 3 and 4, 2005, ACRS Meeting," September 29, 2005. (Pre-decisional).
2. Letter dated June 9, 2004 from Mario V. Bonaca, Chairman, Advisory Committee on Reactor Safeguards, to Luis A. Reyes, Executive Director for Operations, Nuclear Regulatory Commission, Subject: Digital Instrumentation and Control Research Program.

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

February 3, 2006

MEMORANDUM TO: Rich Denning, Chairman
Fire Protection Subcommittee

FROM: John G. Lamb, Senior Staff Engineer
Advisory Committee on Reactor Safeguards Staff

SUBJECT: ANALYSIS OF EDO RESPONSE TO THE ACRS LETTER, DATED DECEMBER 21, 2005, CONCERNING THE DRAFT FINAL GENERIC LETTER 2005-XX, "IMPACT OF POTENTIALLY DEGRADED HEMYC/MT FIRE BARRIER MATERIALS ON COMPLIANCE WITH APPROVED FIRE PROTECTION PROGRAMS"

Attachment 1 contains a copy of the Executive Director for Operations (EDO) January 19, 2006 response (ADAMS Accession No. ML060040050) to the Advisory Committee on Reactor Safeguards (ACRS) December 21, 2005 letter regarding the draft final Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs". Attachment 2 contains a copy of the Committee letter.

The conclusions and recommendations in the ACRS letter are as follows:

- The Generic Letter should be issued.
- We look forward to a briefing by the staff after they have reviewed the responses submitted by the licensees.

EDO RESPONSE

Since the staff position on this topic is the same as that of the ACRS, the EDO has no comments on the ACRS conclusions.

ANALYSIS

The EDO response is satisfactory.

Attachments:
1. Letter from the EDO to G. Wallis, ACRS, dated January 19, 2006, Subject: Draft Final Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs" (ADAMS Accession No. ML060040050)
2. Letter from G. Wallis, ACRS, to the EDO, dated December 21, 2005, Subject: Draft Final Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs" (ADAMS Accession No. ML053620425)

cc: ACRS Members
J. Larkins M. Snodderly
A. Thadani S. Duraiswamy

Similar issues with fire barrier materials have been identified in the past. The 1989 test results of the Thermo-lag fire barrier system indicated that it could not satisfy testing standards. The NRC issued a number of generic communications on this subject. These documents provide a precedent for the activities that are currently being undertaken to understand the scope of the Hemyc and MT issue and ensure that appropriate corrective actions are undertaken.

As a consequence of the test results, the staff developed a generic letter that requests the licensees to report whether Hemyc or MT fire barrier materials are installed and relied on for safe shutdown purposes. Also, the generic letter asks licensees to provide a description of existing programmatic controls to ensure that other types of fire barriers will be assessed for potential degradation and resulting adverse effects. Licensees that have installed Hemyc or MT fire barrier materials must describe the extent of installation, compliance with 10 CFR 50.48 in light of recent test findings, compensatory measures that have been implemented, and a general description of the plan and schedule for corrective actions.

The generic letter also states that affected licensees should provide confirmation by December 1, 2007 that their fire protection programs are in compliance with applicable regulatory requirements. They should also provide a summary of the evaluation used to determine the adequacy of the fire protection program in the presence of potentially degraded Hemyc or MT fire barriers, including the results of any supporting tests performed.

The results of the NRC's independent, confirmatory tests indicate that Hemyc and MT fire barrier systems may not provide adequate protection. The generic letter should be issued to seek information from the licensees. We look forward to a briefing by the staff after they have reviewed the responses submitted by the licensees.

Sincerely,

/RA/

Graham B. Wallis
Chairman

<u>References</u>:

1. Memorandum from J. Lyons, NRR, to J. Larkins, ACRS, dated November 8, 2005, Subject: Proposed Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs" (ADAMS Accession No. ML053110527)
2. Memorandum from J. Larkins, ACRS, to L. Reyes, EDO, dated July 7, 2005, Subject: Proposed Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs" (ADAMS Accession No. ML051940496)
3. Memorandum from D. Lew, RES, to J. Hannon, NRR, dated March 28, 2005, Subject: Preliminary Pass/Fail Test Results for Hemyc 1-Hour Rated Electrical Raceway Fire Barrier Systems (ADAMS Accession No. ML050880176)
4. Information Notice 2005-07, "Results of Hemyc Electrical Raceway Fire Barrier System Full Scale Testing," dated April 1, 2005 (ADAMS Accession No. ML050890089)
5. Regulatory Issue Summary 2005-07, "Compensatory Measures to Satisfy the Fire Protection Program Requirements," dated April 19, 2005 (ADAMS Accession No. ML042360547)

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

February 8, 2006

MEMORANDUM TO: Dana A. Powers, Chairman
Early Site Permits Subcommittee

FROM: M. Snodderly, Acting Chief,
Technical Support Branch, ACRS/ACNW

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON EARLY SITE PERMIT APPLICATION FOR THE GRAND GULF SITE AND THE ASSOCIATED FINAL SAFETY EVALUATION REPORT

Attached is a copy of the EDO's February 1, 2006 letter of response to the ACRS' December 23, 2005 letter on the Committee's review of the final Safety Evaluation Report of the System Energy Resources, Inc., application for the Grand Gulf early site permit.

Committee Letter

In its letter, the Committee concluded that the NRC staff has written a very readable and comprehensive Safety Evaluation Report. The Committee also concluded that the three permit conditions proposed by the staff for the early site permit and the 26 action items for the combined license phase are appropriate. The Committee recommended that the Safety Evaluation Report should be issued once the staff has made more explicit its analyses of the hazards posed to the proposed site by explosions in transportation accidents on the Mississippi River. The Committee also recommended that the NRC staff provide additional guidance to applicants concerning the discussion in an application of "Major Features" of the emergency planning for a proposed site.

EDO Response

The EDO's response states that the NRC staff has noted the ACRS concern with transportation accidents on the Mississippi River and has asked the applicant to provide additional information to demonstrate how it meets Regulatory Guide (RG) 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants." The NRC staff's evaluation of this information will be documented in an upcoming NUREG. Prior to issuance of the NUREG, the staff plans to inform the ACRS of the proposed changes.

The NRC staff agrees with the ACRS recommendation that the NRC staff provide additional guidance to applicants concerning "Major Features" of emergency planning for a proposed site and is working to establish additional guidance. This guidance will be included in a revision of Supplement 2 to NUREG-0654/FEMA-REP-1. It is the NRC staff's understanding that industry does not plan to submit a "Major Features" ESP application in the near future and therefore the priority for this work is considered low. The NRC staff's focus is on activities related to updating the emergency planning sections of the standard review plan and creation of guidance for future combined license applicants.

Recommendation

The EDO's response is satisfactory. The Committee should plan to review guidance being developed for future combined license applicants. In addition, the Committee stated in its July 18, 2005 report to the Commission that, "This first use of the early site permit process has revealed several areas where the process can be refined and streamlined. We look forward to working with the staff to improve the early site permit process." The NRC staff stated in its September 1, 2005 response that, "While some issues may be resolved in the ongoing Part 52 proposed rulemaking, the staff will work with the ACRS to develop additional recommendations as needed." The NRC staff did not mention this commitment in its response to this letter. ACRS staff will coordinate with NRC staff to schedule a joint meeting of the appropriate ACRS Subcommittees to discuss ESP lessons learned, proposed revisions to 10 CFR Part 52, and guidance for future COL applicants.

cc: ACRS Members
J. Larkins
S. Duraiswamy

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

February 1, 2006

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

SUBJECT: EARLY SITE PERMIT APPLICATION FOR THE GRAND GULF SITE AND THE ASSOCIATED FINAL SAFETY EVALUATION REPORT

Dear Chairman Wallis:

Thank you for your letter dated December 23, 2005, regarding the final safety evaluation report (FSER) of the System Energy Resources, Inc. (SERI), application for the Grand Gulf early site permit (ESP). The staff of the U.S. Nuclear Regulatory Commission (NRC) will reproduce your letter as Appendix E to the FSER for the Grand Gulf ESP which will be issued as a final NRC technical report in an upcoming NUREG. In your letter, the Advisory Committee on Reactor Safeguards (ACRS) agreed with the staff's proposed permit conditions, but expressed concern over some of the staff's conclusions associated with the nature of the proposed site.

Specifically, your letter stated that the technical basis for the staff's conclusion on its analyses of the hazards posed to the proposed site by explosions in transportation accidents on the Mississippi River needed to be more explicit. The staff has noted the ACRS concern and has asked the applicant to provide additional information to demonstrate how it meets Regulatory Guide (RG) 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants." The staff's evaluation of this information will be documented in the NUREG. Prior to issuance of the NUREG, the staff plans to inform the ACRS of the proposed changes.

Lastly, ACRS recommended that the staff provide additional guidance to applicants concerning "Major Features" of emergency planning for a proposed site. The staff agrees with the ACRS recommendation and is working to establish additional guidance, which will be included in a revision of Supplement 2 to NUREG-0654/FEMA-REP-1. It is the staff's understanding that industry does not plan to submit a "Major Features" ESP application in the near future and therefore the priority for this work is considered low. Currently, the staff's focus is on activities related to updating the emergency planning sections of the standard review plan and creation of guidance for future combined license applicants.

The NRC staff appreciates the insights that the ACRS has provided concerning the safety review of the Grand Gulf 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

December 23, 2005

Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: EARLY SITE PERMIT APPLICATION FOR THE GRAND GULF SITE AND THE ASSOCIATED FINAL SAFETY EVALUATION REPORT

Dear Mr. Reyes:

During the 528th meeting of the Advisory Committee on Reactor Safeguards, December 7-10, 2005, 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 application and the NRC staff's final Safety Evaluation Report (FSER). We provided an interim report on this application and the draft Safety Evaluation Report on June 14, 2005. We 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.

CONCLUSIONS AND RECOMMENDATIONS

- The NRC staff has written a very readable and comprehensive Safety Evaluation Report. The three permit conditions the staff proposes for the early site permit and the 26 action items for the combined license phase are appropriate.
- This Safety Evaluation Report should be issued once the staff has made more explicit its analyses of the hazards posed to the proposed site by explosions in transportation accidents on the Mississippi River.
- The staff needs to provide additional guidance to applicants concerning the discussion in an application of "Major Features" of the emergency planning for a proposed site.

DISCUSSION

SERI seeks an early site permit for a reactor or a set of reactor modules of total power up to 4300 MW_{th} on a site adjacent to the current Grand Gulf Nuclear Power Station, a BWR/6 with a Mark III containment. With the additional unit or modules, the total nuclear generating capacity at the Grand Gulf site could be as high as 8600 MW_{th}. The Grand Gulf site had previously been approved for two units, but the second unit was never completed.

The SERI application for an early site permit does not specify a particular power plant technology for the new reactor or reactor modules to be placed on the site. The early site permit application, instead, uses a "plant parameter envelope" of power plant characteristics that is intended to bound the reactor technology that could eventually be selected.

- Nature of the Proposed Site

The proposed site is located on the eastern side of the Mississippi River about 25 miles south of Vicksburg, Mississippi. The site is rural in nature. There is little industrial activity and no military base near the site. There is a natural gas pipeline somewhat more than 4 miles from the site.

The nearest major airport is at Jackson, Mississippi, about 65 miles from the proposed site. The staff has determined that the air traffic corridors near the site pose no undue risk. There is a highway 4½ miles from the site. The principal ground transportation hazard is thought to be the delivery of hydrogen to the site for use in the currently operating boiling water reactor. The staff has found that the delivery and storage of this hydrogen would pose no undue risk to the proposed new power plant site.

The most important transportation route near the site is the Mississippi River. The nearest bank of this river is about 1.1 miles from the proposed site. Explosions and releases of toxic gases and vapors could pose threats to the proposed site. The staff and the applicant have agreed to defer consideration of the threats posed by the accidental releases of toxic vapors and gases until a specific plant for the site has been chosen and the habitability of the control room can be evaluated.

The staff has concluded that the detonation of 5000 tons TNT-equivalent bounds the explosion threat to the proposed site. According to staff-approved methods of analysis, such a detonation would require a standoff distance of about 2.1 miles from the facility. The staff concludes, however, that because the site is located behind a 65-foot bluff, the 1.1 mile standoff is adequate. The technical basis for this conclusion needs to be made clear in the Safety Evaluation Report prior to its issuance. This clarification should include a description of the reliability of the calculational method adopted by the staff.

The staff has concluded also that the detonation bounds the explosive hazard posed by vapor explosions such as might occur in the release of liquefied natural gas during a transportation accident on the river. The technical basis for this conclusion should also be made clear in the Safety Evaluation Report. The clarification should include a discussion of whether the staff used the TNT-equivalent method to analyze vapor explosions and the conservatisms associated with such an approximation if it was adopted.

- Population in the Vicinity of the Site

The permanent population around the site is low. The nearest town, Port Gibson, Mississippi, is about 6 miles from the proposed site and has a population of about 1750. The nearest population center, Vicksburg, Mississippi, is 25 miles to the north and has a current population of about 27,000. The projected population growth in the area to the year 2070 is expected to be small, perhaps less than 20%.

- Geology and Seismicity of the Site

The proposed site is located on consolidated river sediments. Geological investigations show no evidence of significant ground deformation for at least the last 500,000 years and perhaps for the last 5 million years. Salt domes in the area are 6 and 8 miles from the proposed site.

The site is in an area of little seismic activity. The nearest historical seismic event occurred more than 25 miles away. The limiting earthquake source is the New Madrid seismic zone over 200 miles away. SERI has performed a probabilistic seismic hazard analysis that takes into account recent revisions made by the U.S. Geological Survey to the frequencies and intensities of events in the New Madrid seismic center. The analysis also considers the possibility of seismic activity along the suspected faults on the Saline River which may not be capable faults. The proposed site is a deep soil site (bedrock is at a depth of about 10,000 feet). SERI has done sufficient characterization of the site to produce analyses of the soil amplification factors. The probabilistic seismic hazard curve developed for the site is bounded by the design safe shutdown earthquake curves adopted in the plant parameter envelope.

- Meteorology

Vigorous storms such as hurricanes and tornados are the principal weather threats to a reactor located on the proposed site. SERI and the staff have used historical information to characterize these and other weather features of the site. In our review of the Safety Evaluation Report, we examined the applicability of hurricane frequency data on the prediction of future storm activity. There is evidence that storm activity is increasing in the Gulf of Mexico due to known weather cycles. The staff and the applicant have used historical data over a sufficient period to capture data from previous weather cycles. We find no definitive evidence that storm intensities in excess of the bounds established by the applicant and accepted by the staff will develop. These bounds may not be especially conservative. Representatives of SERI informed us that inland wind gusts produced by the recent hurricane Katrina at the latitude of the proposed site were somewhat less than 92 mph which can be compared to the 96 mph maximum three-second wind gust adopted for the site characterization. The staff has stated that should future weather evidence indicate site characteristics accepted in the Safety Evaluation Report are not adequate, these characteristics will be amended as needed.

The proposed site is located on a bluff about 65 feet above the normal river level. Land on the opposite bank of the river is more easily flooded than the proposed site. Consequently, major river flooding is not a threat to the site. Local, onsite flooding will have to be addressed if the permit is granted and a decision is made to construct a power plant on the site.

- Emergency Plans

The applicant has elected to submit for review just the "major features" of emergency planning for the proposed site, as is allowed by the regulations. The staff has concluded that these major features are largely adequate. The applicant has stated that the remaining information would be submitted with a combined license application. The applicant and the staff encountered challenges in defining the limitations that should exist on descriptions of major features of emergency planning, especially for a site where reactors currently exist. These challenges could be avoided in the future by providing additional guidance to the applicants.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. 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.

2. System Energy Resources, Inc., Grand Gulf Early Site Permit Application, Revision 0, October 2003.

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.