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

April 4, 2006

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

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 530th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, MARCH 9-11, 2006, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 530th meeting, March 9-11, 2006, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letter, and memoranda:

REPORTS:

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

- Review and Evaluation of the NRC Safety Research Program, dated March 15, 2006
- Final Review of the Exelon Generation Company, LLC, Application for Early Site Permit and the Associated NRC Staff's Final Safety Evaluation Report, dated March 24, 2006
- Report on the Safety Aspects of the License Renewal Application for the Browns Ferry Nuclear Plant Units 1, 2, and 3, dated March 23, 2006
- Generic Safety Issue 191 — Assessment of Debris Accumulation on PWR sump Performance, dated March 24, 2006

LETTER:

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

- Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," dated March 28, 2006

MEMORANDUM:

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

- Resolution of Generic Safety Issue 188, "Steam Generator Tube Leaks or Ruptures Concurrent with Containment Bypass from Main Steam Line or Feedwater Line Breaches," dated March 17, 2006

HIGHLIGHTS OF KEY ISSUES

1. Final Review of the Clinton Early Site Permit Application

The Committee heard presentations by and held discussions with representatives of Exelon Generation Company, LLC (Exelon) and the NRC staff regarding the Early Site Permit (ESP) application for the Clinton site and the associated NRC staff's final Safety Evaluation Report (SER). The Committee had previously met with the NRC staff and applicant during the September 2005 ACRS Full Committee meeting and prepared an interim letter on this application and the associated draft SER on September 22, 2006. This ACRS meeting focused on the geologic and seismic aspects of the Clinton ESP application.

Exelon's ESP application is based on the now familiar "plant parameter envelope" approach since the applicant has not identified the particular reactor technology that will be adopted. Exelon noted that the staff has accepted it's proposed alternative, performance-based, method for the determination of the Safe Shutdown Earthquake (SSE) ground motion spectrum. The geotechnical approach used, and the seismic evaluation conducted, to determine the SSE ground motion were summarized for the Committee. The performance-based approach uses a target mean frequency of 1E-5 per year for seismically induced onset of significant inelastic deformation. This is in contrast to the Regulatory Guide 1.165 approach which uses a reference probability based on not exceeding the median seismically-induced core damage frequency from 29 Individual Plant Examinations for External Events.

The NRC staff provided the Full Committee with a more detailed discussion of the geologic and seismologic review of the Clinton ESP application. The NRC staff concluded that the performance-based approach used by Exelon was technically sound, that the seismic design using the performance-based SSE achieves a safety level generally higher than currently operating plants, and that the performance-based SSE adequately reflects the local ground motion hazard. The NRC staff explained its basis for reaching each of these conclusions. Overall, the NRC staff concluded that the site is acceptable from a geologic and seismologic standpoint and meets the requirements of 10 CFR 100.23.

The final SER documents the staff's technical review of the applicant's site safety analysis report and emergency planning information. Overall, the staff concluded that the level of safety and emergency planning associated with the Clinton ESP is acceptable and meets the regulations.

Committee Action

The Committee issued a report to Chairman Diaz dated March 24, 2006, concluding that the ESP application and final SER show that the site adjacent to the existing Clinton Nuclear Power Station is an acceptable site for nuclear power plants that meet the plant parameter envelope proposed by the applicant. The Committee also concluded that the staff has thoroughly reviewed a performance-based method proposed by the applicant for determining SSE ground motion and recommended that the staff consider the development of a regulatory guide dealing with the alternative, performance-based method for assessing the seismic hazard of a site.

2. 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" and Results of the Chemical Effects Tests Associated with PWR Sump Performance

The staff discussed licensee responses to Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," and presented the results of efforts by the Office of Nuclear Regulatory Research to understand several phenomenological issues that have arisen as part of the GSI-191 effort, including chemical effects, downstream effects, and head loss correlations through debris beds. The responses to the GL and the results of the recent research have raised new questions. Present plans by licensees to increase the size of their sump screens will reduce the head loss across these screens, but the staff's ability to assess the adequacy of the reduction may be limited by uncertainties in the available knowledge base. In addition downstream effects may be exacerbated by some screen designs and configurations.

The staff reported that exploratory chemical effects tests have revealed that some chemical species can be produced under certain conditions that can have a substantial effect on screen pressure drop. The staff has concluded that plant-specific evaluations of the response to this phenomenon are required. Additional experiments to reproduce previous screen head loss data have produced significantly different results, and these results indicate that the structure of the debris bed and the way in which it is formed can have a huge influence on the head loss. Unless the assumption of a homogeneous bed can be justified, it will be necessary to develop an adequate model for these effects or find a way to scale them in the proof tests now planned by industry.

With regard to debris that passes through the screens into the reactor coolant system, the staff and industry representatives stated that they thought that the core would be adequately cooled in a number of scenarios, however, they presented no physical models or analytical predictions to show a validated, quantitative basis for these conclusions. The Committee believes that additional research is needed to develop an adequate understanding of the effects of the various debris species which enter the reactor vessel and reach the core.

Committee Action

The Committee issued a report to Chairman Diaz dated March 24, 2006, recommending that additional work is required to provide the technical basis by which the staff can assess the adequacy of the planned modifications to PWR sump screens. Improved predictive methods and guidance should be developed for particle/fiber mixtures and chemical reaction products that are deposited on sump screens. Methods for predicting the quantity and properties of debris that bypasses the sump screens should be developed, and their potential adverse effects on downstream components should be evaluated. Equilibrium chemistry models should be validated further and guidance should be developed for their use. The results of tests of coating debris formation and transport should be included in the assessment of core coolability as they become available, which should include the development of adequate predictive capability for the effects of coating debris on screen pressure drop and bypass.

3. Final Review of the License Renewal Application for Browns Ferry Units 1, 2, and 3

The Committee met with the NRC staff and representatives of the Tennessee Valley Authority (TVA) to review the License Renewal Application (LRA) for Browns Ferry Units 1, 2, and 3 and the associated final SER. TVA has requested approval for continued operation of each unit for a period of 20 years beyond the current license expiration dates of December 10, 2013 for Unit 1, June 28, 2014 for Unit 2, and July 2, 2016 for Unit 3. The three Browns Ferry Units are General Electric BWR 4 reactors in Mark I containments with nearly identical materials, systems, components, and environments. TVA will eliminate the differences between the current licensing basis of Unit 1 and Units 2 and 3 prior to Unit 1 restart in May 2007. To address concerns raised by the Committee in its interim report, TVA described the applicability of operating experience from Units 2 and 3 to Unit 1 and the attributes of the Unit 1 Periodic Inspection Program. The objective of this aging management program is to verify that no latent aging effects are occurring in Unit 1 piping components that were in layup but were not replaced prior to restart. TVA also described the process for tracking license renewal commitments, the status of the implementation of aging management programs, and the implementation of the Maintenance Rule for Unit 1. The staff provided highlights of its review of this LRA and described the EDO response to the Committee's interim report. The final SER issued in January 2006 describes the resolution of four open items and two confirmatory items. In March 2006, the staff reopened one of the open items based on new information provided by TVA regarding drywell inspection results. Ultrasonic inspections identified a small inclusion in the drywell liner of Unit 1. The staff will document its evaluation of this information in a supplemental SER.

Committee Action

The Committee issued a report to the NRC Chairman dated March 23, 2006, recommending that the license renewal application for Browns Ferry Units 1, 2, and 3 be approved under two conditions. The first condition is that the drywell refueling seals should be included within scope of license renewal and subjected to periodic inspections or the drywell shells should be subjected to periodic volumetric inspections to detect external corrosion. The second condition is that if an extended power uprate is implemented before the period of extended operation, the staff should require that TVA evaluate Units 1, 2, and 3 operating experience at the uprated power level and incorporate lessons learned into their aging management programs prior to entering the period of extended operation.

4. Draft Final Revision 4 (DG-1128) to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"

The Committee heard presentations by and held discussions with representatives of the staff regarding the draft final Revision 4 Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." The staff provided a summary of the comments received during the public comment period along with its responses to those comments. The staff explained the changes to the draft final Regulatory Guide based on the public comments. The Committee expressed a concern with Regulatory Position 1, which states, "If a current operating reactor licensee voluntarily converts to the criteria in Revision 4 of this guide, the licensee should perform the conversion on the plant's entire accident monitoring program to ensure a complete analysis." The Committee stated that this position is to restrictive.

Committee Action

The Committee issued a letter to the EDO, dated March 28, 2006, recommending that the staff not issue the draft final Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4. The Committee recommended that the staff revise Regulatory Position 1 to allow licensees to adopt the IEEE 497-2002 Standard to modify individual accident monitoring instruments without a complete analysis of all accident monitoring instrumentation. The Committee agreed that licensees should not be allowed to partially use the new Standard to eliminate or reclassify accident monitoring instrumentation required by earlier standards unless Revision 4 of the Regulatory Guide is adopted in its entirety.

5. Evaluation of Precursor Data to Identify Significant Operating Events

The Committee heard presentations by and held discussions with representatives of the staff regarding the evaluation of Accident Sequence Precursor (ASP) data to identify significant operating events. The staff provided a background of the ASP program, status of ASP analyses, ASP program accomplishments, interesting 2004 analyses, potentially interesting fiscal year 2005 analyses, and ASP trends from SECY-05-0192, "Status of the Accident Sequence Precursor Program and the Development of Standardized Plant Analysis Risk Models." There were no significant precursors (conditional core damage probability greater than or equal to 1×10^{-3}) in fiscal years 2003, 2004, or 2005.

Committee Action

This was an information briefing and no Committee action was required.

6. Draft Final 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 March meeting, the Committee discussed its draft final 2006 report to the Commission on the NRC Safety Research Program.

Committee Action

The Committee forwarded an advance copy of its 2006 report on the NRC Safety Research Program to the Commission on March 15, 2006. The final report will be issued as NUREG-1635, Vol. 7.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

The Committee considered the EDO's response of February 9, 2006, to comments and recommendations included in the January 4, 2006 ACRS report on the proposed Vermont Yankee Extended Power Uprate. The Committee decided that it was satisfied with the EDO's response.

The EDO response noted that the letter included some additional comments from several ACRS members which addressed a proposed approach for consideration of uncertainties as part of an assessment of crediting containment overpressure. The NRC staff will consider the ACRS comments as it develops more explicit guidance as part of the ongoing revisions to Regulatory Guide (RG) 1.82. Based on discussions with the ACRS, during NRC staff presentations related to the proposed revisions to RG 1.82, the staff understands that the ACRS would prefer that licensees use a statistical approach for the analysis related to crediting containment overpressure. The staff is currently developing guidance for this new approach and will bring the revised RG 1.82 to the Committee in the future.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from February 9, 2006, through March 8, 2006, the following Subcommittee meetings were held:

- Thermal-Hydraulic Phenomena — February 14-16, 2006

The Subcommittee heard presentations from the staff concerning licensee responses to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors, and the results of efforts by the Office of Nuclear Regulatory Research to understand several phenomenological issues that have arisen as part of the GSI-191 effort, including chemical effects, downstream effects, and hear loss correlations through debris beds.

- Early Site Permits — March 8, 2006

The Subcommittee reviewed the application for an early site permit for the Clinton site, and the associated NRC staff's final Safety Evaluation Report. The Subcommittee discussed at length the applicant's performance-based seismic hazard analysis methodology.

- Planning and Procedures — March 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.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The staff committed to provide to the Committee the Supplemental Safety Evaluation Report related to the license renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3
- The Committee plans to review the staff's resolution of issues raised by the Committee regarding Revision 4 to Regulatory Guide 1.97.
- The Committee plans to continue to work with the staff on PWR sump performance issues.

PROPOSED SCHEDULE FOR THE 531st ACRS MEETING

The Committee agreed to consider the following topics during the 531st ACRS meeting, to be held on April 5-8, 2006:

Safeguards and Security Matters
Application of TRACG Code to ESBWR Stability
Hazards Analysis Associated with the Grand Gulf Early Site Permit Application and the Associated NRC Staff's Evaluation
Safety Conscious Work Environment/Safety Culture
Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants"
Review of 1994 Addenda for Class 1, 2, and 3 Piping Systems to the ASME Code Section III and the Resolution of the Differences Between the Staff and ASME

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

CERTIFIED

Date Issued: 4/11/2006
Date Certified: 4/19/2006

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MARCH 9-11, 2006

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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

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MINUTES OF THE 530th MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MARCH 9-11, 2006
ROCKVILLE, MARYLAND

The 530th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on March 9-11, 2006. Notice of this meeting was published in the *Federal Register* on February 24, 2006 (65 FR 9611) (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. J. Sam Armijo, Dr. Mario V. Bonaca, Dr. Richard S. Denning, Dr. Thomas S. Kress, Mr. Otto L. Maynard, and Dr. Dana A. Powers. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

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

Dr. Graham B. Wallis, Committee Chairman, convened the meeting at 8:30 a.m. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee. In addition, Dr. J. Sam Armijo was introduced as the newest Committee member.

II. Final Review of the Clinton Early Site Permit Application (Open)

[Note: Mr. David Fischer was the Cognizant Staff Engineer and Mr. Michael Snodderly was the Designated Federal Official for this portion of the meeting.]

The Committee heard presentations by and held discussions with representatives of Exelon Generation Company, LLC (Exelon) and the NRC staff regarding the early site permit application for the Clinton site and the associated NRC staff's Final Safety Evaluation Report.

Exelon introduced its ESP Project and support teams and then outlined significant changes that have been made since issuance of the draft safety evaluation report (DSER). All open items in the DSER have been closed and all confirmatory items have been completed. Exelon described each open item and briefly explained how each was closed. Based on staff documented criteria the number of permit conditions was reduced from 15 in the DSER down to six in the final safety evaluation report (FSER). However, the number of combined license (COL) action items increased from 17 in the DSER to 32 in the FSER. Exelon noted that the staff has accepted it's proposed alternative, performance-based, method for the determination of the safe shutdown earthquake (SSE) ground motion spectrum. This alternative is based on an industry standard (ASCE 43-05) that itself is based on work done by the Department of Energy for the seismic safety of its facilities. The geotechnical approach used, and the seismic evaluation conducted, to determine the SSE ground motion were summarized for the Committee. The alternative performance-based approach uses a target mean frequency of 1E-5 per year for seismically induced onset of significant inelastic deformation of SSCs. This alternative in contrast to the RG 1.165 approach which uses a reference probability based on not exceeding the median seismically-induced CDF from 29 plant IPEEEs.

The NRC staff outlined past and future milestones associated with the review of the Clinton ESP application. The FSER documents the staff's technical review of the applicant's site safety analysis report and emergency planning information. Exelon requested that the ESP site be approved for total core thermal power rating between 2400 and 6800 Mwt. Exelon's early site permit application is based on the now familiar "plant parameter envelope" approach since the applicant has not identified the particular reactor technology that will be adopted. The NRC staff identified the key areas reviewed in the ESP application and the principal staff reviewers. The numbers of open items, permit conditions, and COL action items in the DSER were in contrast to the numbers in the FSER. Overall, the staff concluded that the safety and emergency planning associated with the Clinton ESP is acceptable and meets the regulations.

The NRC staff provided the full Committee with a more detailed discussion of the geologic and seismologic review of the Clinton ESP application. The NRC staff formed an inter-office Seismic Technical Advisory Group to help guide the staff's review. The NRC staff also had technical support from the U.S. Geologic Survey and the Brookhaven National Laboratory. The NRC staff concluded that the performance-based approach used by Exelon was technically sound, that the seismic design using the performance-based SSE achieves a safety level generally higher than currently operating plants, and that the performance-based SSE adequately reflects the local ground motion hazard. The NRC staff explained to the ACRS its basis for reaching each of these conclusions. The performance-based approach used by Exelon is technically sound because it achieves both a high and consistent level of seismic safety, it takes no credit for seismic margin, it utilizes a conservative performance target (i.e., mean target frequency of 1E-5 per year for seismically induced onset of significant inelastic deformation of SSCs), and it is based on conservative parameter and modeling assumptions. The performance-based SSE for the Clinton ESP is the product of the site-specific seismic hazard curve and probability density function for SSC seismic fragility. The staff's evaluation included an independent assessment of the analysis results by direct integration of the seismic risk equation. The performance-based approach used by Exelon achieves a safety level generally higher than currently operating plants because the median seismic core damage frequency for IPEEEs, as represented in RG 1.165, is 1E-5 per year whereas Exelon's performance-based approach uses this same frequency as the target for the onset of inelastic deformation. This target provides a rather substantial margin to core damage and containment failure. The seismic core damage frequency that can be inferred from the proposed ground motion spectrum (~$2X10^{-6}$/yr) is significantly less than the median found in seismic probabilistic risk assessments for 29 existing nuclear power plants. Thus, the performance-based alternative method yields results that are in concert with the Commission's expectation that future reactors be safer than currently operating reactors. The performance-based approach used by Exelon adequately reflects local ground motion hazards in that it considered large events in the New Madrid seismic zone in the past 2,000 years, large events in the Wabash Valley/Southern Illinois seismic zone in the past 12,000 years, as well as a moderate energy event near Springfield, Illinois, which occurred approximately 6,000 years ago and had a magnitude of 6.2 to 6.8 on the Richter scale. Exelon also conducted paleoliquification surveys on streams near the ESP site and found no evidence of repeated, moderate to large earthquakes comparable to the Springfield earthquake. Overall, the NRC staff concluded that the site is acceptable from a geologic and seismologic standpoint and meets the requirements of 10 CFR 100.23.

Committee Action

The Committee issued a letter report to Chairman Diaz dated March 24, 2006, on this matter, that supported issuance of the ESP. No additional followup activities were identified for either the applicant, NRC staff, or the Committee.

III. 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" and Results of the Chemical Effects Tests Associated with PWR sump Performance (Open)

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

The staff discussed the licensee responses to Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," and presented the results of efforts by the Office of Nuclear Regulatory Research (RES) to understand several phenomenological issues that have arisen as part of the GSI-191 effort, including chemical effects, downstream effects, and head loss correlations through debris beds. The responses to the GL and the results of the recent research have raised new questions. Present plans by licensees to increase the size of their sump screens will reduce the head loss across these screens, but the staff's ability to assess the adequacy of the reduction may be limited by uncertainties in the available knowledge base. In addition, downstream effects may be exacerbated by some screen designs and configurations.

The staff reported that exploratory chemical effects tests have revealed that some chemical species can be produced under certain conditions that can have a substantial effect on screen pressure drop. The staff has concluded that plant-specific evaluations of the response to this phenomenon are required. Additional experiments to reproduce previous screen head loss data have produced significantly different results, and these results indicate that the structure of the debris bed and the way in which it is formed can have a huge influence on the head loss. Unless the assumption of a homogeneous bed can be justified, it will be necessary to develop an adequate model for these effects or find a way to scale them in the proof tests now planned by industry.

With regard to debris that passes through the screens into the reactor coolant system, the staff and industry representatives stated that they thought that the core would be adequately cooled in a number of scenarios, however, they presented no physical models or analytical predictions to show a validated, quantitative basis for these conclusions. The Committee believes that additional research is needed to develop an adequate understanding of the effects of the various debris species which enter the reactor vessel and reach the core.

Committee Action

The Committee issued a report to Chairman Diaz dated March 24, 2006, recommending that additional work is required to provide the technical basis by which the staff can assess the adequacy of the planned modifications to PWR sump screens. Improved predictive methods and guidance should be developed for particle/fiber mixtures and chemical reaction products that are deposited on sump screens. Methods for predicting the quantity and properties of debris that bypasses the sump screens should be developed, and their potential adverse effects on downstream components should be evaluated. Equilibrium chemistry models should be validated further and guidance should be developed for their use. The results of tests of coating debris formation and transport should be included in the assessment of core coolability as they become available, which should include the development of adequate predictive capability for the effects of coating debris on screen pressure drop and bypass.

IV. Final Review of the License Renewal Application for Browns Ferry Units 1, 2, and 3 (Open)

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

The Committee met with the NRC staff and representatives of the Tennessee Valley Authority (TVA) to review the license renewal application (LRA) for the Browns Ferry Nuclear (BFN) Plant Units 1, 2, and 3 and the associated final SER. The operating licenses for Units 1, 2, and 3 expire on December 10, 2013, June 28, 2014, and July 2, 2016, respectively. TVA has requested approval for continued operation of each unit for a period of 20 years beyond the current license expiration dates.

Mr. Crouch, TVA, stated that the LRA was submitted on December 31, 2003 and is based on the currently licensed thermal power levels for each unit. Unit 1 is currently shutdown but is scheduled to restart in May 2007. Units 2 and 3 are currently operating at 105% of their originally licensed thermal power. Appendix F of the LRA describes the difference between the current licensing basis (CLB) of Unit 1 and Units 2 and 3. These differences will be eliminated prior to the restart of Unit 1. The approximate durations of operation are 10 years for Unit 1, 23 years for Unit 2, and 18 years for Unit 3.

Mr. Crouch described the applicability of operating experience from Units 2 and 3 to Unit 1. The three BFN units are General Electric BWR 4 reactors in Mark I containments with nearly identical materials, systems, components, and environments. The three units have a total of 51 calendar years of operational experience. Unit 3 was shutdown for 10 years with the same layup philosophy, processes, and conditions as Unit 1. No layup-induced aging effects have been found in Unit 3 during the following 10 years of operation. As a result of the layup experience with Unit 3, TVA replaced piping in the residual heat removal service water system and the raw cooling water system. Mr. Crouch added that Unit 1 piping susceptible to intergranular stress corrosion cracking will be replaced. TVA did not credit the Unit 1 Layup Program as the sole means to establish acceptability of components for restart or license renewal. The same aging management programs (AMPs) are being implemented for all three units for the duration of the original license and the period of extended operation. Dr. Bonaca, ACRS Member, noted that an NRC inspection report found that initial layup conditions were uncontrolled.

Mr. Valente, TVA, listed the reasons for major equipment replacements and repairs for Unit 1. These reasons include reliability, fidelity with Units 2 and 3, regulatory issues, dose reduction, maintenance reduction, lessons learned from recovery of Unit 3, and extended power uprate.

As a compensatory action, the Unit 1 Periodic Inspection Program will be implemented to verify that no latent aging effects are occurring in piping that was in layup but not replaced. Mr. Valente stated that this program supplements other AMPs. A baseline inspection will be performed before restart. Additional inspections will be performed after Unit 1 restart and again within the first ten years of entering the period of extended operation. The subsequent inspection frequency will be based on the inspection results. The inspection samples will be grouped by common material types and environments. In response to a question from Mr. Sieber, ACRS Member, TVA stated that a minimum of 59 points will be inspected for each of these group. The sample size will be based on a 95/95 confidence level and will include locations susceptible to degradation and areas where degradation is not expected. TVA listed the 25 systems which would be subject to these periodic inspections. In response to a question from ACRS Member Dr. Apostolakis, TVA stated that the core damage frequencies for Unit 1, 2, and 3 are 1.77×10^{-6}, 2.6×10^{-6}, and 3.3×10^{-6}, respectively. These frequencies are for internal events only.

Mr. Brune, TVA, stated that no major exceptions were taken to the Generic Aging Lessons Learned (GALL) Report. The 39 AMPs are adequate to manage the aging effects for which they are credited. Mr. Brune listed the eight programs which have taken only minor exceptions to the GALL Report.

Mr. DeLong, TVA, described the Corrective Action Program (CAP) and the tracking of license renewal commitments. The CAP applies to all TVA units so that any condition identified at any unit is reviewed for generic implications to all other units at TVA sites. In addition, internal and external plant operating experience is incorporated into the CAP. To date there are 110 license renewal commitments. These include implementing new AMPs, enhancing existing AMPs, and eliminating the differences in CLB between Unit 1 and Units 2 and 3. These license renewal commitments are tracked with an onsite commitment tracking system and the CAP.

Mr. DeLong provided the status for implementing the AMPs. The only revision to 11 of the AMPs is to include Unit 1 and these revisions will be completed in 2006. Revisions to 11 other AMPs that do not require enhancement will be completed in 2007. Revisions to the 11 other AMPs that require enhancement for all three units will be completed in 2008. The remaining six AMPs are new programs that will be developed by 2009. Mr. DeLong added that the implementation packages for all 39 AMPs have been developed, reviewed, and approved.

Mr. Crouch described the implementation of the Maintenance Rule for Unit 1. In 1997 the NRC granted a temporary exception to the Maintenance Rule for some of the Unit 1 systems that do not perform their intended function in a defueled condition. Those Unit 1 systems that support operation of Units 2 or 3 are included by the Maintenance Rule Program. The temporary exception will be eliminated when a system is required to be operable by technical specifications.

The staff's presentation provided highlights of the review for this LRA and described the EDO response to the Committee's interim letter. Ms. Sanabria, NRR, stated that the SER with open items was issued in August 2005. The final SER issued in January 2006 described the resolution of two confirmatory items and four open items.

In March 2006 the staff reopened the item regarding drywell shell corrosion based on new information provided by TVA regarding drywell inspection results. The final SER explains that this item was closed out with a commitment by TVA to perform one-time inspections of the drywell shell in each unit. The staff accepted one-time inspections based on information provided by TVA that indicated no significant degradation has been observed in the drywell. Mr. DeLong described these inspection results provided to the staff. The first ultrasonic inspection of the Unit 1 drywell shell was performed in 1987 in response to a generic letter. Another inspection in 1999 discovered a small inclusion in the drywell shell at a depth of 0.76 inches. This result was confirmed during subsequent inspections performed in 2002 and 2004. These inspection results showed no degradation of the drywell shell thickness due to corrosion. Mr. Crouch added that improved transducers used in the 1990's are what enabled inspectors to detect this inclusion which was likely preset during the 1987 inspection. TVA stated that the drywell liner is subject to ASME Section XI IWE inspections. Mr. Jang, NRR, stated that these IWE examinations are visual inspections that would not be capable of detecting corrosion on the inaccessible side of the drywell liner. Mr. Kuo, NRR, stated that the staff is considering the issuance of interim staff guidance regarding the subject of drywell shell corrosion.

TVA will formally provide these inspection results to the staff in writing. The staff plans to issue a supplemental SER that will evaluate this information and describe the resolution of this open item. The supplemental SER will also provide additional details of the Unit 1 Periodic Inspection Program.

The open item regarding stress relaxation of core plate hold-down bolts was resolved by a commitment from TVA to perform a plant specific analysis consistent with BWRVIP-25. TVA will submit this analysis for the staff's review and approval two years prior to entering the period of extended operation. The open item regarding the Unit 1 Periodic Inspection Program was also resolved. The staff's evaluation of this program is described in the final SER. TVA will implement this program prior to Unit 1 restart. The open item regarding inspection of the residual heat removal service water piping was identified from the regional inspections. TVA will confirm that there is no blockage in these pipes by using the buried piping and tanks inspection program. The staff considers this a confirmatory item pending formal submittal of this commitment by TVA.

In order to address one of the Committee's concerns, Ms. Sanabria stated that TVA will evaluate BFN operating experience at the uprated power level and incorporate lessons learned into their aging management programs prior to entering the period of extended operation. The staff concluded that pending resolution of the open item regarding drywell shell corrosion, the requirements of the license renewal rule have been met.

Committee Action

The Committee issued a report to the NRC Chairman dated March 23, 2006, recommending that the license renewal application for Browns Ferry Units 1, 2, and 3 be approved under two conditions. The first condition is that the drywell refueling seals should be included within scope of license renewal and subjected to periodic inspections or the drywell shells should be subjected to periodic volumetric inspections to detect external corrosion. The second condition is that if an extended power uprate is implemented before the period of extended operation, the staff should require TVA to evaluate Units 1, 2, and 3 operating experience at the uprated power level and incorporate lessons learned into their aging management programs prior to entering the period of extended operation.

V. Draft Final Revision 4 (DG-1128) to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (Open)

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

The Chairman of the Plant Operations Subcommittee provided background and an introduction to the staff. The Committee had the benefit of presentations and discussions with representatives of the staff regarding the draft final Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4 (the draft was issued as DG-1128, dated June 2005).

The staff provided background information, described Regulatory Guide 1.97 Revision 3, described IEEE Standard 497-2002, and explained the information contained in Regulatory Guide 1.97 Revision 4.

The staff endorses IEEE Standard 497-2002 in Regulatory Guide 1.97 Revision 4 subject to eight regulatory positions. The staff described the eight regulatory positions.

The staff provided the Committee with a summary of the letters that the staff received during the public comment period. The staff described their responses to the public comments. The staff explained the changes to the draft final Regulatory Guide based on the public comments.

The Committee had concerns with Regulatory Positions 1 and 4. Regulatory Position 1 states, "If a current operating reactor licensee voluntarily converts to the criteria in Revision 4 of this guide, the licensee should perform the conversion on the plant's entire accident monitoring program to ensure a complete analysis." Regulatory Position 4 states, "Modify the last sentence in Clause 4.1 as follows: 'Type A variables include those variables that are associated with contingency actions that are within the plant licensing basis and may be identified in written procedures.'"

Committee Action

The Committee issued a letter dated March 28, 2006, recommending that the staff not issue the draft final Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4. The Committee recommended the staff revise Regulatory Position 1 to allow licensees to adopt the proposed standard to modify individual accident monitoring instruments without a complete analysis of all accident monitoring instrumentation. The Committee agreed that licensees should not be allowed to partially use the new standard to eliminate or reclassify accident monitoring instrumentation required by earlier standards unless Revision 4 of the regulatory guide is adopted in its entirety.

VI. Evaluation of Precursor Data to Identify Significant Operating Events (Open)

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

The Chairman of the Plant Operations Subcommittee provided an introduction to the staff. The Committee had the benefit of presentations and discussions with representatives of the staff regarding the evaluation of Accident Sequence Precursor (ASP) data to identify significant operating events. The staff provided a background of the ASP program, status of ASP analyses, ASP program accomplishments, interesting 2004 analyses, potentially interesting fiscal year 2005 analyses, and ASP trends from SECY-05-0192. There were no significant precursors (conditional core damage probability greater than or equal to 1 x 10^{-3}) in fiscal years 2003, 2004, or 2005.

There were four interesting 2004 ASP analyses: (1) Palo Verde loss-of-offsite power (LOOP), (2) Palo Verde Emergency Core Cooling System (ECCS) piping voids, (3) Saint Lucie LOOP during Hurricane Jeannee, and (4) Calvert Cliffs trip and potential overcooling.

The Palo Verde LOOP occurred at all three units, the grid LOOP was complicated with a breaker failure, and Unit 2 had a Emergency Diesel Generator fail. The ASP results for the Palo Verde LOOP were 9E-6, 4E-5, and 9E-6 for Units 1, 2, and 3, respectively.

The significance determination process (SDP) for the Palo Verde ECCS piping voids conservatively assumed that low pressure recirculation would not work for a medium-sized loss-of-coolant accident. ASP used expert panel opinion to create probability distribution for system operability and the result was consistent with the SDP. The ASP results for the Palo Verde ECCS piping voids were 1E-5 for each unit.

Salt spray on switchyard equipment created uncertain recoverability for the Saint Lucie LOOP during Hurricane Jeannee. The full power model was adjusted to credit pre-hurricane shutdown procedures. The ASP results for the Saint Lucie LOOP during Hurricane Jeannee was 1E-5 per unit.

The Calvert Cliffs trip and potential overcooling event was caused by a reactor trip due to low steam generator level caused by the loss of a main feedwater pump. A relay failure caused the excessive cooldown. The Standardized Plant Analysis Risk models were modified to include over-steam demand sequences. The ASP results for the Calvert Cliffs trip and potential overcooling event was 5E-5.

Four precursors identified in fiscal years 2002 - 2004 had a conditional core damage probability greater than 1×10^{-4}. The four precursors are Davis-Besse reactor head event, the potential common mode failure of the auxiliary feedwater at Point Beach Units 1 and 2, and another potential common mode failure of auxiliary feedwater at Point Beach Unit 2.

Committee Action

This was an information briefing and no Committee action was required.

VII. Draft Final 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 March meeting, the Committee discussed its draft final 2006 report to the Commission on the NRC Safety Research Program.

Committee Action

The Committee forwarded an advance copy of its 2006 report on the NRC Safety Research Program to the Commission on March 15, 2006. The final report will be issued as NUREG-1635, Vol. 7.

VIII. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

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

The Committee considered the EDO's response of February 9, 2006, to comments and recommendations included in the January 4, 2006 ACRS report on the proposed Vermont Yankee Extended Power Uprate. The Committee decided that it was satisfied with the EDO's response.

The EDO response noted that the letter included some additional comments from several ACRS members which addressed a proposed approach for consideration of uncertainties as part of an assessment of crediting containment overpressure. The NRC staff will consider the ACRS comments as it develops more explicit guidance as part of the ongoing revisions to Regulatory Guide (RG) 1.82. Based on discussions with the ACRS, during NRC staff presentations related to the proposed revisions to RG 1.82, the staff understands that the ACRS would prefer that licensees use a statistical approach for the analysis related to crediting containment overpressure. The staff is currently developing guidance for this new approach and will bring the revised RG 1.82 to the Committee in the future.

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 March 8, 2006. The following items were discussed:

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

Member assignments and priorities for ACRS reports and letters for the March 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 May 2006 was addressed. The objectives were:

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

During this session, the Subcommittee 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 that:

> 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.

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 and the associated resource needs.

During the February 2006 ACRS meeting, the Committee authorized the ACRS Executive Director to work with the Planning and Procedures Subcommittee and develop a final response. A final response, which reflected incorporation of comments received from the Planning and Procedures Subcommittee members, was sent to the members in February 2006.

The ACRS Chairman and Executive Director reconciled comments received from ACRS members and a revised final draft was prepared and distributed to Committee members for comment. Following the March 2006 full Committee meeting, the ACRS Chairman will forward to the Commission the Committee's proposal for handling the anticipated workload increase.

ACRS Conference Room Upgrade

During the February ACRS meeting, members were informed about the upgrade to the ACRS conference room audiovisual equipment. The upgrade began on March 13, 2006 and is expected to be completed on or before April 24, 2006. Arrangements will be made to hold ACRS Subcommittee and Full Committee meetings in other conference room locations.

Reappointment of Dr. Powers for a Fourth Term

The Commission took exception to its current policy of the maximum three-term limit to the ACRS members and reappointed Dr. Powers for a fourth term.

Interview of Candidates to Fill the Vacancy on the Committee

The members interviewed several candidates for membership on the ACRS on March 8-9, 2006. Subsequent to interviewing the candidates on March 8 and 9, the members should provide their feedback to the ACRS Chairman. The ACRS Chairman will provide the members' views to the ACRS Candidate Screening Panel during the March meeting.

Quadripartite Meeting Status

Planning for the 2006 Quadripartite Meeting continues as scheduled and full participation is expected from the Member Countries. There will be 15 participants from France's GPR and IRSN; 18 participants from Germany, including 11 RSK members, three from BMU and four from the RSK secretariat. Participants from Japan are anticipated. Among the invited participants, Switzerland's KSA will send two attendees. Sweden and Finland are expected to send participants.

Assignments have been made to the staff engineers on specific topics to assist the ACRS members in preparing the abstracts which are due on March 31, 2006.

The next major steps include: identifying and inviting key note speakers; formally inviting the Commission and selected NRC staff; selecting translators for the Japanese and the French. Additionally, there are a number of other administrative issues being addressed by the ACRS staff.

April ACRS Meeting

During the April 6-8, 2006 ACRS meeting, the Committee is scheduled to write a report on security-related research activities and plant-specific mitigation strategies. Since a large amount of information needs to be discussed, Dr. Bonaca, Chairman of the ACRS Subcommittee on Safeguards and Security, suggested that the April ACRS meeting start at 1:30 p.m. on Wednesday April 5, 2006 to discuss the safeguards and security matters.

Staff Requirements Memorandum Related to ACRS/ACNW Coordination

In an SRM dated February 9, 2006, which resulted from the ACNW meeting with the NRC Commissioners on January 11, 2006, the Commission stated that the ACNW should find, with input from the ACRS, an approach to provide the Commission with a coordinated set of recommendations on how the Center for Nuclear Waste Analyses (CNWRA) might broaden its assistance to NRC. For example, to support NRR programs and/or other new and significant regulatory research activities. Additionally, in an SRM dated February 7, 2006, regarding the ACNW Action Plan, there were some additional activities that the ACNW had been tasked to perform, such as staying abreast of new approaches to reprocessing technology and fuel cycle, that could involve coordination with the ACRS.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 531st ACRS Meeting, April 6-8, 2006.

The 530th ACRS meeting was adjourned at 12:00 p.m. on March 11, 2006.

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

April 19, 2006

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

FROM: Graham B. Wallis
ACRS Chairman

Graham B. Wallis

SUBJECT: CERTIFIED MINUTES OF THE 530th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), MARCH 9-11, 2006

I certify that based on my review of the minutes from the 530th 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.

type of information contained on the tape (*i.e.*, calendar files, index files, documents files, note files, and residual files).

NARA Action

NARA will proceed to dispose of 9,193 PROFS backup tapes created during the Clinton Administration by WHCA staff as specified in the EFFECTIVE DATE of this notice, because NARA has determined that they lack sufficient administrative, historical, informational, or evidentiary value. This notice constitutes NARA's final agency action pursuant to 44 U.S.C. 2203(f)(3).

Dated: February 17, 2006.

Allen Weinstein,

Archivist of the United States.

[FR Doc. E6–2641 Filed 2–23–06; 8:45 am]

BILLING CODE 7515–01–P

NATIONAL SCIENCE FOUNDATION

Astronomy and Astrophysics Advisory Committee #13883; Notice of Meeting

In accordance with the Federal Advisory Committee Act (Pub. L. 92–463, as amended), the National Science Foundation announces the following Astronomy and Astrophysics Advisory Committee (#13883) meeting:

DATE AND TIME: March 10, 2006. 11 a.m.–6 p.m. EST.

PLACE: Teleconference. National Science Foundation, Room 1045, Stafford I Building, 4121 Wilson Blvd., Arlington, VA, 22230.

TYPE OF MEETING: Open.

CONTACT PERSON: Dr. G. Wayne Van Citters, Director, Division of Astronomical Sciences, Suite 1045, National Science Foundation, 4201 Wilson Blvd., Arlington, VA 22230. Telephone: 703–292–4908.

PURPOSE OF MEETING: To provide advice and recommendations to the National Science Foundation (NSF), the National Aeronautics and Space Administration (NASA) and the U.S. Department of Energy (DOE) on issues within the field of astronomy and astrophysics that are of mutual interest and concern to the agencies.

AGENDA: To discuss the Committee's draft annual report due 15 March 2006.

Dated: February 17, 2006.

Susanne E. Bolton,

Committee Management Officer.

[FR Doc. 06–1705 Filed 2–23–06; 8:45 am]

BILLING CODE 7555–01–M

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 March 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, March 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.: Final Review of the Clinton Early site Permit Application (Open)—The Committee will hear presentations by and hold 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.

10:45 a.m.–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)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the staff's evaluation of the licensees' responses to Generic Letter 2004–02 on PWR sumps.

1 p.m.–3 p.m.: Results of the Chemical Effects Tests Associated with PWR Sump Performance (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and its contractor regarding results of the chemical effects tests related to PWR sump performance.

3:15 p.m.–5:15 p.m.: Final Review of the License Renewal Application for Browns Ferry Units 1, 2, and 3 (Open)—The Committee will hear presentations by and hold 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.

5:30 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Friday, March 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.: Draft Final Revision 4 (DG–1128) to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final revision 4 to Regulatory Guide 1.97.

10:15 a.m.–11:45 a.m.: Evaluation of Precursor Data to Identify Significant Operating Events (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the staff's evaluation of precursor data to identify significant operating events.

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

2 p.m.–2:15 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

2:30 p.m.–4:30 p.m.: Draft Final 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.

4:45 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, March 11, 2006, Conference Room T–2b3, Two White Flint North, Rockville, Maryland

8:30 a.m.–1:00 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

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

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

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

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

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

Dated: February 17, 2006.

Annette L. Vietti-Cook,

Secretary of the Commission.

[FR Doc. E6–2664 Filed 2–23–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Availability of Documents Regarding Spent Fuel Transportation Package Response to the Caldecott Tunnel Fire Scenario

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

FOR FURTHER INFORMATION CONTACT: Allen Hansen, Thermal Engineer, Criticality, Shielding and Heat Transfer Section, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20005–0001. Telephone: (301) 415–1390; fax number: (301) 415–8555; e-mail: *agh@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Introduction

Under contract with the Nuclear Regulatory Commission (NRC), The Pacific Northwest National Laboratory prepared a draft NUREG/CR report, "Spent Fuel Transportation Package Response to the Caldecott Tunnel Fire (CTF) Scenario." Highway tunnel fire accidents are very low frequency events, but can be severe, in terms of fire duration and peak temperatures. The CTF was chosen for the study because it represents a severe historical highway tunnel accident, even though it is a very low frequency event. This NUREG/CR documents the thermal analysis of one spent fuel transportation package, the NAC International Model No. LWT ("NAC LWT"), exposed to boundary conditions simulating the CTF scenario.

The results of this study strongly indicate that no spent nuclear fuel (SNF) particles or fission products would be released from the NAC LWT or a similar spent fuel shipping cask involved in a severe tunnel fire such as the Caldecott highway tunnel fire. The peak internal temperatures predicted for the NAC LWT in the analysis of the CTF scenario were not high enough to result in rupture of the fuel cladding. Therefore, it would not be expected that any radioactive material (*i.e*, SNF particles or fission products) would be released from within the fuel rods.

The maximum NAC LWT temperatures experienced in the regions of the lid, vent and drain ports exceeded the seals' rated service temperatures, making it theoretically possible for a small release to occur, due to CRUD that might spall off of the surfaces of the fuel rods. However, any release is expected to be very small due to a number of factors. These include: (1) The tight clearances maintained between the lid and cask body by the lid closure bolts; (2) the low pressure differential between the cask interior and the outside; (3) the tendency of the small clearances to plug; and (4) the tendency of CRUD particles to settle or plate out. The potential releases calculated in Chapter 8 of this report for the NAC LWT truck cask indicate that the release of CRUD from the cask, if any, would be very small—less than an A_2 quantity.

II. Summary

The purpose of this notice is to provide the public an opportunity to review and comment on the Draft NUREG/CR thermal analysis, the consequence analyses and the conclusions.

III. Further Information

The document related to this action is available on-line at *http://www.nrc.gov/reading-rm/doc-collections/nuregs/docs4comment.html.* In addition, a copy of this document has been posted electronically at the NRC's Electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html.* From this site, you can access the NRC's Agencywide Document Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The ADAMS accession number for the document related to this notice is ML060330028. If you do not have access to ADAMS or if there are problems in accessing the document located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.*

This document may also be viewed electronically on the public computers located at the NRC's PDR, O 1 F21, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852. The PDR reproduction contractor will copy documents for a fee. Comments and questions on the draft NUREG/CR can be entered on-line or directed to the

APPENDIX II

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. **9:30 AM** 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. 1:25 PM *****LUNCH*****

4) ~~1:00 - 3:00~~ P.M. **1:25-3:30 PM** 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. 3:30-3:45 PM	*****BREAK*****
5)	~~3:15 - 5:15~~ P.M. **3:45-5:45 PM**	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. **9:45 AM**	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. 9:45	*****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. **3:15 PM** 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)

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

14) 8:30 - ~~1:00~~ P.M. 12:00 PM **(10:30-10:45 A.M. BREAK)**

Preparation of ACRS Reports (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.

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

530th ACRS MEETING
MARCH 9-11, 2006

NRC STAFF (3/9/2006)

K. Hsu, NRR
T. Scarbrough, NRR
M. Widman, RII
J. Rowley, NRR
L. Tran, NRR
K. Chang, NRR
K. Tanabe, NRR
L. Lund, NRR
J. Zimmerman, NRR
Y. Diaz, NRR
D. Cullim, NRR
J. Grobe, NRR
R. Architzel, NRR
P. Klein, NRR
M. Yoder, NRR
L. Whitney, NRR
W. Jensen, NRR
J. Lehminey, NRR
J. Hannon, NRR
R. McNally, NRR
M. Hart, NRR
M. Stutzke, NRR
E. Murphy, NRR
H. Wagaji, NRR
M. Evans, RES

J. Jolicoeur, RES
M. Gavrilas, RES
S. Lee, NRR
W. Bateman, NRR
H. Chernoff, NRR
M. Chernoff, NRR
T. Le, NRR
R. Sullanoh, NRR
B. Elliot, NRR
R. Karas, NRR
Y.C. (Renee) Li, NRR
D. Jeng, NRR
A. Pal, NRR
M. Hartzman, NRR
D. Reddy, NRR
H. Hamzehee, NRR
C. Moulton, NRR
G. Cheruvenki, NRR
K. Parczewsky, NRR
J. Storch, OIG
R. Lanksbury, RIII
J. Lee, NRR
L. Dudes, NRR
B. Ibrahim, NMSS
B. Harvey, NRR

B. Moudy, NSIR
C. Munson, NRR
D. Barss, NSIR
R. Karas, NRR
M. Kotzaras, NRR
J. Segola, NRR
G. Bagchi, NRR
S. Ali, RES
N. Patel, NRR
T. Hafera, NRR
S. Lu, NRR
L. Berg, NRR
W. Krotiuk, RES
S. Uwilewicz, NRR
E. Geiger, RES
B. P. Jain, RES
R. Reyes, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

P. Blaney, Legin Group
M. Homiack, Legin Group
C. Church, Legin Group
R. Gralks, Legin Group
R. Jansen, TVA
K. Brune, TVA
D. Arp, TVA
M. Heath, PGN
G. Little, TVA
R. Jennings, TVA
R. Moll, TVA
B. Crouch, TVA

J. Valente, TVA
M. Bajestani, TVA
J. McCarthy, TVA
K. Hanson, Geomatrix Consult.
C. Stemp, EHS
B. Kennedy, RPK Struct. Mech.
T. Miller, DOE
B. Youngs, Geomatrix Consult.
M. Maher, Exelon
C. Kerr, Exelon
E. Grant, Exelon
T. Mundy, Exelon

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
T. Andreychell, Westinghouse
J. Butler, NGI
S. Woo, Korea Institute of Nuc Safety
F. Kim, Korea Power Engineering Co.
Y. Hayashi, Kansai Electric Power
T. Yamada, INES
S. Dolley, Inside NRC-McGraw Hill
D. Raleigh, LIS Scientech
M. Gallagher, Exelon

NRC STAFF (3/10/2006)
G. Tartal, RES
B. Marcus, NRR
S. Ardnt, RES
B. Kemper, RES
M. Waterman, RES
M. Evans, RES
P. Appignani, RES
D. Marksberry, RES
S. Wong, NRR
C. Hunter, RES
A. Rubin, RES
J. Kauffman, RES
s. Sancaktan, RES
D. Rasmuson, RES
A. Grady, RES
G. DeMoss, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
J. Kenney, GE BWROG
M. Presley, MIT

APPENDIX IV: FUTURE AGENDA

March 21, 2006

SCHEDULE AND OUTLINE FOR DISCUSSION
531st ACRS MEETING
APRIL 5-8, 2006

WEDNESDAY, APRIL 5, 2006, T-8E8, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 1:30 - 6:30 P.M. **(3:30-3:45 P.M. BREAK)**

Safeguards and Security Matters (Closed) (MVB/EAT)
1.1) Remarks by the Subcommittee Chairman
1.2) Briefing by and discussions with representatives of the NRC staff regarding safeguards and security matters.

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

THURSDAY, APRIL 6, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

2) 8:30 - 8:35 A.M.

Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)
2.1) Opening statement
2.2) Items of current interest

3) 8:35 - 10:30 A.M.

Application of TRACG Code to ESBWR Stability (Open/Closed) (GBW/RC)
3.1) Remarks by the Subcommittee Chairman
3.2) Briefing by and discussions with representatives of the NRC staff and General Electric Nuclear Energy regarding application of the TRACG Code for analyzing the Economic Simplified Boiling Water Reactor (ESBWR) Stability.

[NOTE: A portion of this session may be closed to discuss General Electric Proprietary information pursuant to 5 U.S.C. 552b(c) (4).]

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

4) 10:45 - 11:45 A.M.

Hazards Analysis Associated with the Grand Gulf Early Site Permit Application and the Associated NRC Staff's Evaluation (Open) (DAP/DCF/MRS)
4.1) Remarks by the Subcommittee Chairman
4.2) Briefing by and discussions with representatives of the NRC staff, and System Energy Resources, Inc. as needed, regarding the hazards analysis associated with

the Grand Gulf Early Site Permit Application and the associated NRC staff's evaluation.

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

5) 12:45 - 2:45 P.M. Safety Conscious Work Environment/Safety Culture (Open) (MVB/JHF)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding staff activities associated with responding to the Commission's Staff Requirements Memorandum on Safety Conscious Work Environment/Safety Culture, and related matters.

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

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

6) 3:00 - 4:30 P.M. Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants" (Open) (GEA/JGL)

6.1) Remarks by the Subcommittee Chairman

6.2) Briefing by and discussions with representatives of the NRC staff regarding draft final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants," and related matters.

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

4:30 - 4:45 P.M. *****BREAK*****

7) 4:45 - 6:45 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

7.1) Application of TRACG Code to ESBWR Stability (GBW/RC)

7.2) Hazards Analysis Associated with the Grand Gulf Early Site Permit Application (DAP/DCF/MRS)

7.3) Safety Conscious Work Environment/Safety Culture (MVB/JHF)

7.4) Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants" (GEA/JGL)

7.5) Safeguards and Security Matters (Closed) (MVB/EAT)

FRIDAY, APRIL 7, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 - 10:00 A.M. Review of 1994 Addenda for Class 1, 2, and 3 Piping Systems to the ASME Code Section III and the Resolution of the Differences Between the Staff and ASME (Open) (JSA/CS)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff and the American Society of Mechanical Engineers (ASME) regarding the 1994 Addenda for Class 1, 2, and 3 Piping Systems to the ASME Code Section III and the resolution of the differences between the NRC staff and ASME.

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

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

10) 10:15 - 10:45 A.M. Subcommittee Reports (Open)

10.1) Report by and discussions with the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim review of the Nine Mile Point license renewal application and the associated NRC staff's draft Safety Evaluation Report (JDS/JGL)

10.2) Report by and discussions with the Chairman of the ACRS Subcommittee on Power Uprates regarding interim review of the Ginna core power uprate application and the associated NRC staff's Safety Evaluation (RSD/RC)

11) 10:45 - 11:45 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)

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

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

12) 11:45 - 12:00 Noon — Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

12:00 - 1:00 P.M. — *****LUNCH*****

13) 1:00 - 1:30 P.M. — Quality Assessment of Selected NRC Research Projects (Open) (DAP/HPN)
Selection of projects and assignments for assessing the quality of the selected research projects.

14) 1:30 - 6:30 P.M. **(3:15-3:30 P.M. BREAK)** — Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

14.1) Application of TRACG Code to ESBWR Stability (GBW/RC)
14.2) Hazards Analysis Associated with the Grand Gulf Early Site Permit Application (DAP/DCF/MRS)
14.3) Safety Conscious Work Environment/Safety Culture (MVB/JHF)
14.4) Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants" (GEA/JGL)
14.5) Review of 1994 Addenda for Class 1, 2, and 3 Piping Systems to the ASME Code Section III and the Resolution of the Differences Between the Staff and ASME (JSA/CS)
14.6) Safeguards and Security Matters (Closed) (MVB/EAT)

SATURDAY, APRIL 8, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

15) 8:30 - 12:30 P.M. **(10:30-10:45 A.M. BREAK)** — Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 14.

16) 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 V

LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
530th ACRS MEETING
MARCH 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 March 9-11, 2006

2 Final Review of the Clinton Early Site Permit Application
2. Early site Permit Application, Clinton Power Station Site, Final Safety Evaluation Report presentation by Exelon [Viewgraphs]
3. Exelon Early Site Permit Safety Review Status presentation by NRR [Viewgraphs]

3 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"
4. Generic Safety Issue 191 presentation by B. Sheron, NRR
5. Overview of Resolution Status and Plans for Generic Safety Issues (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance" presentation by NRR Staff
6. Industry Activities to Address PWR ECCS Sump Performance presentation by NEI

4 Results of the Chemical Effects Tests Associated with PWR Sump Performance
7. Overview of NRC-Sponsored Research Supporting GL 2004-02 Resolution presentation by RES [Viewgraphs]
8. Advanced Nuclear Power, The Magazine of Framatome ANP [Handout]
9. Letter from NEI to Brian Sheron, NRR, dated 2/28/2006, Subject: NRC Requests for Additional Information to PWR Licensees Regarding Responses to Generic Letter 2004-02

5 Final Review of the License Renewal Application for Browns Ferry Units 1, 2, and 3
10. Tennessee Valley Authority Browns Ferry Nuclear Plant License Renewal presentation [Viewgraphs]
11. Browns Ferry Nuclear Plant Units 1, 2, and 3 License Renewal Safety Evaluation Report staff presentation by NRR [Viewgraphs]

8 Draft Final Revision 4 (DG-1128) to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"

12. Regulatory Guide 1.97, Revision 4 "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" presentation by RES

9 Evaluation of Precursor Data to Identify Significant Operating Events

13. Evaluation of Accident Sequence Precursor Data to Identify Significant Operating Events presentation by RES

10 Future ACRS Activities/Report of the Planning and Procedures Subcommittee

14. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - March 8, 2006 [Handout #10.1]

11 Reconciliation of ACRS Comments and Recommendations

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

Appendix V
530th ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Early site Permit Application for the Exelon Generation Company, LLC and the Associated Final Safety Evaluation Report
	1. Table of Contents
	2. Proposed Agenda
	3. Status Report
	4. ACRS Interim Letter dated September 22, 2005
	5. EDO response to ACRS Interim Letter dated October 26, 1005
	6. FSER/Appendix A, dated February 17, 2006
3	Licensee Responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors" and PWR sump Research Issues Including Chemical Effects Tests
	7. Table of Contents
	8. Proposed Schedule
	9. Status Report
	10. Document List
	11. Draft Summary of Thermal-Hydraulic Subcommittee Meeting Minutes, February 14-16, 2006
5	Review of the License Renewal Application and Final Safety Evaluation Report for the Browns Ferry Nuclear Plant Units 1, 2, and 3
	12. Table of Contents
	13. Meeting Schedule
	14. Status Report
8	Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4
	15. Draft Proposed Agenda
	16. Status Report
	17. Redline/Strikeout Version of Regulatory Guide 1.97 (Draft was issued as DG-1128, dated June 2005), "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4 dated December 2005
	18. Memorandum from J. Wiggins, RES, to J. Larkins, ACRS, Subject: Request for ACRs Review of Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation of Nuclear Power Plants," Revision 4 dated January 30, 2006
	19. Regulatory Guide 1.97 (draft was issued as DG-1128 dated June 2005), "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, dated December 2005
	20. Staff Responses to Public Comments on DG-1128 dated January 31, 2006
	21. IEEE Standard 497-202, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Generating Stations"

22. Memorandum from J. Larkins, ACRS, to L. Reyes, EDO, Subject: Proposed Regulatory Guide (DG)-1128, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants (Revision 4 to Regulatory Guide 1.97) dated July 8, 2005

9. <u>Status of the Accident Sequence Precursor (ASP) Program</u>
    23. Draft Proposed Agenda
    24. Status Report
    25. SECY-05-0192, Subject: Status of the Accident Sequence Precursor (ASP) Program and the Development of Standardized Plant analysis Risk (SPAR) Models dated October 24, 2005

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
530th FULL COMMITTEE MEETING

March 9-11, 2006

TODAY'S DATE: March ~~10,~~ 9 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	K. R. Hsu	NRR/DLR
2	Thomas Scarbrough	NRR/DCI
3	Marcus T. Widmann	RII/DRP/Branch 6.
4	Jonathan Rowley	NRR/DLR
5	Linh Tran	NRR/DLR
6	Ken Chang	NRC/DLR
7	Kiyoto Tanabe	NRC/DLR/RLRC/F.A.
	Louise Lund	NRC/DLR/RLRA
9	Jake Zimmerman	NRC/DLR/RLRB
10	Yoira Diaz	NRR/DLR/RLRA
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
530th FULL COMMITTEE MEETING

March 9-11, 2006

TODAY'S DATE: March 9, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	David Cullison	NRR/DSS
2	JACK GROBE	NRR D:DCI
3	Ralph Architzel	NRR : SSIB
4	Paul · Klein	NRR/DCI
5	Matt Yoder	NRR/DCI
6	Leon Whitney	NRR/DSS/SSIB
7	Walton Jensen	NRR/DSSA
	John Lehning	NRR
9	John [illegible]	NRR/DSS
10	Richard McNally	NRR/DCI
11	Michelle Hart	NRR/DRA
12	MARTIN STUTZKE	NRR/DRA/APLA
13	Emmett Murphy	NRR/DCI/CSGB
14	Henry Wagage	NRR/DSS/SSIB
15	Michele Evans	RES/DFERR/ERA
16	JOHN JOLICOEUR	RES/DRASP/NRCA/SMB
17	MIRELA GAVRILAS	RES/DRASP/NRCA/Adv. Rx
18	SAMSON LEE	NRR/DCI
	W. Bateman	NRR/DCI
20		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
530th FULL COMMITTEE MEETING

March 9-11, 2006

TODAY'S DATE: March 9, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Jack Grobe	NRR DiDCI
2	Harold Chernoff	NRR/DRP/PGCB
3	MARGARET CHERNOFF	NRR/ADRO/DORL/LPLD
4	Tommy Le	NRR/DLR
5	Ram Subbaratnam	NRC/DLR.
6	BARRY ELLIOT	NRC/DCI/CVID
7	Rebecca Karas	NRR/DE/EGCB
8	Y. C. (Renee) Li	NRR/DE/EEMB
9	David Jeng	NRR/DE/EGCB
10	Peter Blaney	Legin Group
11	Matt Homiack	Legin Group
12	Cheryl Church	Legin Group
13	Aman Pal	NRR/DE/EEEB
14	MARK HARTZMAN	NRR/DE/EEMB
15	Devender Reddy	NRR/DLR
16	Rafael Gracia	Legin Group
17	HOSSEIN HAMZEHEE	NRR/DE
18	Charles Moulton	NRR/DRA/AFPB
19	Ganesh Cheruvenki	NRR/DCI.
20	Kris Parczewski	NRR/DCI
21.	Jaclyn Storch	NRC/OIG

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
530th FULL COMMITTEE MEETING

March 9-11, 2006

TODAY'S DATE: March 9, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Roger Lanksbury	RIII/DRS
2	Jay Lee	NRR/DRA/AADB
3	Laura A. Dudes	NRR/DNRL
4	Baker Ibrahimi	NMSS/HLWRS
5	Brad Harvey	NRR/DRA/AADB
6	Bob Moody	NSIR/DPR/EPD
7	Cliff Munson	NRR/DE
	Dan Barss	NSIR/DPR/EP
9	Rebecca Karas	NRR/DE/EGCB
10	Margie Kotzalas	NRR/DRA/AADB
11	John Segala	NRR/DNRL
12	Goutam Bagchi	NRR/DE
13	Syed Ali	NRC/RES
14	Nitin Patel	NRR/DNRL
15	Thomas Hafera	NRR/DSS
16	Shanlai Lu	NRR/DSS
17	Larry Berg	NRR/DSS
18	William Krotiuk	RES
	Steven Unikewicz	NRR/DCI/CPTB
20	Ervin Geiger	RES/DEFFER

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
530th FULL COMMITTEE MEETING
March 9-11, 2006

March 9, 2006
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

	NAME	AFFILIATION
1	Russell Jansen	TVA
2	Kenneth Brune	TVA
3	Don Arp	TVA
4	Mike Heath	PGN
5	GILBERT V LITTLE	TVA
6	Roger Jennings	TVA
7	Robert J. Moll	TVA
8	RICH DELONG	TVA
9	BILL CROUCH	TVA
10	Joe Valente	TVA
11	Masoud Bajestani	TVA
12	[illegible]	
13	Joe McCarthy	TVA
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
530th FULL COMMITTEE MEETING
March 9-11, 2006

March 9, 2006
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
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	NAME	AFFILIATION
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
530th FULL COMMITTEE MEETING
March 9-11, 2006

March 9, 2006
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

	NAME	AFFILIATION
1	Kathryn L Hanson	Geomatrix Consultants.
2	Carl Stepp	EHS
3	Bob Kennedy	RPK Struct. Mech.
4	Tom Miller	DEPARTMENT OF ENERGY
5	Bob Youngs	Geomatrix Consultants
6	Bill Maher	Exelon
7	Chris Kerr	Exelon
8	Eddie R Grant	Exelon
9	THOMAS MUNDY	Exelon
10	TIM ANDREYCHEK	WESTINGHOUSE
11	John BUTLER	NEI
12	Sweng-Woong Woo	Korea Institute of Nuclear Safety
13	Eun-Kee Kim	Korea Power Engineering Company
14	Yuichi HAYASHI	KANSAI ELECTRIC POWER
15	Tomoho Yamada	JNES
16	Steven Dolley	Inside NRC - McGraw Hill
17	Diann Raleigh	LIS Scientech
18	~~[illegible]~~ B. P. Jain	RES/NRC
19	Ruth Reyes	SSIB/NRC
20	Michael P. GALLAGHER	EXELON

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
530th FULL COMMITTEE MEETING

March 9-11, 2006

TODAY'S DATE: March 10, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	George Tartal	RES/DFERR
2	Barry Marcus	NRR/DE/EICB
3	Steven Arndt	RES/DFERR
4	Bill Kemper	RES/DFERR
5	Mike Waterman	RES/DFERR
6	Michele Evans	RES/DFERR
7	Peter Appignani	RES/OREA
8	Don Marksberry	RES/OEGIB
9	See Meng Wong	NRR/DRA/APOB
10	Christopher Hunter	RES/OEGIB
11	Alan Rubin	RES/PRB
12	John Kauffman	RES/OEGIB
13	Selim Sancaktar	RES/OERA
14	Dale Rasmuson	RES/OERA/PRB
15	A Brd	RES/OERA/PRB
16	Gary DeMoss	RES/OEGIB
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
530th FULL COMMITTEE MEETING
March 9-11, 2006

<u>March 10, 2006</u>
Today's Date

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ITEMS OF INTEREST

530th ACRS MEETING

MARCH 9-11, 2006

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
530th MEETING
March 9-11, 2006

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No. S-06-002

"NEW PLANT DESIGN, CERTIFICATION AND LICENSING"

Remarks as Prepared for Delivery
Chairman Nils J. Diaz
U.S. Nuclear Regulatory Commission
at the
Platts Conference
Washington, D.C.
February 13, 2006

Good morning. Thank you for the kind introduction, and thanks to Platts for the opportunity to present my views on "Nuclear Energy: Opportunities for Growth and Investment in North America." Indeed, it is a pleasure to be here today, at a time when our nation, and many other nations, have to address national security, energy security, and economic security holistically, and when nuclear energy generation is being seriously considered as one of the solutions. It is always a challenge to speak first at a large meeting dealing with a broad range of dynamic issues, including sociopolitical, financial, economic, energy security, and, yes, regulatory issues, every one of them important to the potential growth and utilization of nuclear energy. However, I noticed, with pleasure, that Secretary Bodman will be speaking right after me. This is a unique opportunity for me to offer short, polite, bland remarks and pass the buck to Secretary Bodman. I would probably get away with it too. But I won't do it.

I believe that safe, reliable, and secure nuclear energy has been and can continue to be part of the solution to energy security and environmental stewardship, and thus contribute to the well-being of our people. I also believe that this next time around, nuclear power plant deployment should be carefully planned and key issues and interfaces resolved at the front end, executed on budget and on schedule, with all the safety and engineering know-how developed and learned over the last 25 years. The development, review, and potential deployment of reactors must contain all the safety checks and balances required by the law and demanded by the need to ensure the protection and security of our people.

The Nuclear Regulatory Commission has new and difficult issues to resolve in a short period of time to discharge well our licensing responsibilities, while not missing a step in our continuing safety oversight of nuclear facilities and materials. We realize the full scope of our responsibilities, are facing them with all our resources, and plan to do them well, and do them openly. Therefore, I must today answer broad questions for a broad audience.

First, where does the Commission stand overall? This Commission clearly, deliberately, and openly set the objective that governs our activities. The Commission stated in its Strategic Plan that the NRC's objective is 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.

From my vantage point, I can tell you that the NRC is true to this objective, and the agency will continue to be true to it. To further this objective, we continue to improve the organization, to prioritize, manage, and use resources well, and to revisit and create ways to better implement every major agency function. I believe the agency has achieved and continues to achieve results that leave no doubt of the agency-wide commitment to the objective of enabling the beneficial uses of nuclear energy, within the proven and improving safety framework for which we are responsible, in an effective, efficient, realistic, and timely manner. In fact, we have the record to prove it, and any occasional mistake or deficiency becomes obvious because it is the exception to the rule. And when such a mistake occurs, we take care of it.

Therefore, I do get concerned when I hear and read about perceptions of NRC "regulatory instability" or "lack of regulatory predictability." I want to be completely clear on this: the Nuclear Regulatory Commission is a regulatory agency with a high degree of predictability for a given set of circumstances. But we are not miracle workers; the agency will work well, and better, when we have high quality inputs and, correspondingly, well-defined processes, tasks, and schedules. Obviously, a multitude of circumstances will define the playing field.

A lot of the buzz centers around the predictability of outcomes from the use of 10 CFR Part 52, which contains the requirements for Early Site Permits; Standard Design Certifications; and Combined Licenses (COL) for Nuclear Power Plants. Outcomes depend on many factors, and one of the key factors is the quality of the application submitted. Timely outcomes also depend on the planning and processes that I will discuss today. It is true that the combined license component of Part 52 has not yet been used. Clearly, we now have experience with early site permits and extensive experience with design certifications. The reality is that the staff and the Commission also have extensive experience in performing the critical elements of a COL review. We have learned much from these experiences, which include safety evaluations, environmental impact assessments, ACRS reviews, public interactions, Federal/State/local interactions, and public hearings.

The primary purpose for establishing the new Part 52 process for licensing nuclear power facilities was to encourage early resolution of issues to increase regulatory predictability in advance of major financial commitments, while maintaining the requisite safety reviews. Yet, questions are frequently asked about whether the use of Part 52 will provide regulatory predictability at the COL stage. I believe that some are questioning the regulatory predictability for new reactors mainly because of two particular aspects of this new Part 52: the mandatory hearing that must precede a decision to issue a COL and the potential for a second hearing prior to fuel loading. The NRC is established with an adjudicatory Board consisting of legal and technical members, with the capability and legal authority to conduct hearings and rigorous reviews of alleged deficiencies in applications. Although the agency has not processed a COL application and therefore has not been through a hearing for these aspects of the Part 52 process, the Commission and its Atomic Safety and Licensing Board have extensive experience with licensing and with adjudications for various types of facilities. Recently, we have been conducting mandatory hearing proceedings, and, for the most part, they have proceeded in an organized and timely fashion. It is noteworthy that Atomic Safety and Licensing Board and Commission decisions have consistently been upheld when challenged in courts of appeals and the Supreme Court.

Moreover, this Commission has a record that stands out in assuring that adjudication is fair and equitable, as well as effective and efficient. In 1998 we issued a Policy Statement on conduct of our hearings that set the stage for efficient conduct of proceedings on license renewals and license transfers. We followed that statement with a revision of our rules of practice to improve the accessibility, effectiveness, and efficiency of the hearing process. The Commission has provided model schedules to guide our Boards and expedite adjudicatory proceedings for both pending and future proceedings. It has also required the participants to comply with NRC procedural rules. Litigating COL adjudicatory proceedings will undoubtedly present new possibilities for promoting both effective and efficient resolution of issues, particularly with respect to common issues. For example, for cases proceeding in parallel, a party may seek, or a Board may convene, separate Licensing Boards to resolve discrete, common issues in a consistent fashion and in parallel with the resolution of other issues. The point here: A final decision on an issue that is common to a number of cases can become precedent setting, potentially reducing the need to revisit it in future cases. Thorough and sound work by all involved when issues are first presented will be key to take advantage of these potential efficiencies.

Let me briefly address the potential for a second hearing. The threshold for granting such a hearing is high. If a plant is built in accordance with the license, then the Commission has the capability, and in fact the obligation and the responsibility, to allow the plant to operate. If a hearing is granted, operation may be permitted for an interim period while the hearing is conducted. Part 52 provides criteria and procedures under which the Commission must and will ensure that no frivolous means are used to create a second hearing. However, the responsibility rests squarely on the applicants to maintain a complete and accurate record, showing that the facility is constructed and will be operated in accordance with the license, to allow the NRC to confidently make the necessary findings.

A couple of personal comments. I do not mind when the NRC is called demanding on safety, exacting and driven on security and emergency preparedness, intrusive on oversight; or to the contrary not sufficiently demanding in these areas. If I do not know the answer to any of these challenges, I will check and probe to make sure we are where we should be pursuant to the law and Commission policy, but I don't mind being questioned. But unpredictable? No way.

When we talk about predictability for licensing new reactors, I believe that we need to talk about "overall predictability," not only NRC's. Predictability begins when an applicant starts to consider an application, and extends through licensing, construction, and operation of the facility. With the present projected schedules, and the need to establish the requisite infrastructure to meet those demanding schedules, resolving significant issues at the front end becomes very important. The industry and the NRC can and should do much better than in the 70's and 80's. Having said that, let me just emphasize that predictability in reactor licensing is everybody's business; and the NRC accepts its share of the responsibility. I will now turn to how the NRC is addressing, predictably, the issue of new reactor licensing and our internal and external expectations.

The Commission just approved a proposal to revise 10 CFR Part 52 to clarify it and enhance its usability. I know that the proposed changes to Part 52 are extensive, and it has been argued that some of these are marginally beneficial. However, we can benefit from a better and clearer Part 52 that would facilitate the upcoming safety reviews for new plants. I encourage all stakeholders to submit their comments on the proposed rule early so that the staff can finish its work on this rulemaking in October 2006 and the Commission can make its decision. What we need to do at this point is to get this rulemaking done.

One of the planned activities for new reactor licensing is in the area of security. The NRC has three important security rulemakings planned or underway to codify security requirements for power reactors. The first is the rulemaking on the design basis threat for radiological sabotage. The proposed rule is currently out for public comment and a final rule will be issued later this year. The second rulemaking will amend the power reactor security regulations in 10 CFR 73.55, 73.56, 73.57, and Part 73 appendices to align them with the series of orders the Commission issued following September 11, 2001, and to ensure safety-security interface issues are properly considered in plant operations. The Commission intends to finalize this rule as early in calendar year 2007 as possible. Finally, the Commission's expectations on security design for new reactor licensing activities are to be codified in a third rulemaking by September 2007. The expectation of the Commission is that the lessons learned by the agency and reactor licensees pre- and post-9/11/2001 should be considered by the vendors at the design stage. We have learned much and I believe improvements can be realized without major design or construction changes.

To set the stage for my next set of comments, I would like to discuss where potential applicants are today, in the dynamic front of new reactor applications. To date, 11 potential COL applications have been publicly announced, distributed among the 3 major reactor vendors now competing for the U.S. marketplace. Nine months to a year represents a schedule for completion of any contested proceeding, which begins early in the staff review process, as well as the mandatory hearing, which follows completion of the staff's review.

In order to effectively review multiple COL applications in parallel, the staff is now preparing to implement a design-centered approach for NRC's reviews of COL applications, to the extent possible, for as many issues as possible. This approach involves the use, for each issue, of one review and one position for multiple applications. It could also be called the "one-for-all" approach. It is ready for use now; however, it needs the nuclear industry's commitment. One-for-all is one thorough, comprehensive NRC safety evaluation to be used repeatedly, as appropriate. Although the U.S. nuclear industry has not necessarily been endowed with "oneness," the one-for-all approach might not be too bad for those who plan to apply for COLs. Using the design-centered approach, the NRC staff would use a single technical evaluation to support multiple combined license applications for the same technical area of review, as long as the applications standardize the licensing basis to a level that would make this approach viable. For technical review areas amenable to this approach, the staff can complete the evaluation for a "reference" case, can determine if the design proposed by other applicants is the same as the design reviewed, and proceed to issue the evaluation, without further review. Let me emphasize, again, that standardization is key for this approach to work; in fact, the term "oneness" comes to mind.

The design-centered approach could also be applied to parallel reviews of a design certification application and COL applications referencing the design. For example, NRC reviews for the ESBWR and the EPR designs are likely to be conducted in parallel with reviews of the first few COL applications referencing these designs. The NRC could proceed with its review of each design and issue a safety evaluation report with open items, just as was done in the case of the AP1000 and earlier designs. Using the design-centered approach, the resolution of generic open items in the NRC safety evaluation report could be coordinated among the vendor and the applicants for COLs referencing the vendor's design. The resolution of these generic issues could then be incorporated into the design and included in the rulemaking certifying the design. In this manner, they would be available to future applicants referencing the design.

I believe that applying the design-centered approach to parallel design certification and COL reviews, and relying on disciplined standardization, will result in a better, more detailed, and more thorough safety evaluation for each design. When an applicant references a standard design certified by rulemaking, all design matters within the scope of the design certification rule have been resolved using a fair and equitable process and need not be re-addressed in the COL proceeding. The design-centered approach could also lead to a significantly higher level of efficiency in the licensing process thereby reducing the amount of staff resources necessary to conduct each review. We will continue to review our funding needs to determine what is necessary to carry out our responsibilities.

Furthermore, in the Part 52 rulemaking the Commission is soliciting comments on an approach that would facilitate amendments to design certification rules after completion of the initial certification. With such a provision, a detailed standard certified design would be able to incorporate additional features that are generic to the design. NRC will be predictably more efficient if industry adopts a standardized approach.

Let me now use the AP1000 to show how a more detailed Design Certification Rule could be beneficial to COL applicants, the NRC, and public participants. The present AP1000 Certified Design does not include specific design details in a few important areas, such as instrumentation and control systems, and control room and piping designs. This was done to allow utilization of the rapidly changing technologies in advanced designs; these areas are currently addressed by Design Acceptance Criteria. The Design Acceptance Criteria are a special set of inspections, tests, analyses, and acceptance criteria to be used at the COL stage to ensure that specific designs meet applicable regulatory requirements. Since specific design details for these areas were not included in the AP1000 rulemaking, they would have to be addressed by each COL application and potentially each COL hearing. Again, I believe that if proposals to address these areas were to be standardized to the extent practicable, their review could be conducted once in the context of an amendment to the Design Certification Rule to codify a design that the NRC has found acceptable. The rulemaking could be conducted prior to or in parallel with the review of the "reference" COL application and completed prior to adjudication on the "reference" COL.

Amendments to Design Certification Rules and implementation of the design-centered approach are consistent with the goal of standardization and the safety benefits associated with such standardization, as envisioned by the developers of Part 52 and the Congress of the United States. It is also consistent with the U.S. Department of Energy 2010 Initiative, which is centered on standardization.

Clearly, I am extolling the predictability and benefits of standardization, including increased resolution and closure of design safety issues. I know that utility executives that have expressed an interest in applying for a COL are also seriously interested in standardization. I also note that rulemaking affords the benefit of broad public participation and allows interested parties to focus on particular areas of concern.

Could it be done differently? Of course it could, and the law clearly says so. The NRC has the obligation to conduct licensing reviews in the different manners outlined in Part 52, if requested by applicants, and to do so as effectively as possible. However, considering the number of potential applications for new plants that are expected to use the AP1000, the ESBWR, and the EPR, there is much appeal in an approach that resolves specific design details for all important areas early in the process. I also believe that early resolution of environmental issues and emergency preparedness, prior to submittal of the COL application, could be beneficial to the timely completion of COL reviews. For example, this combination, with a design-centered approach, could shorten NRC's review schedule by about one year. Regardless, the agency needs to be prepared to act on multiple applications using several designs in a timely manner, using the provisions of Part 52. Once we have reviewed multiple applications, and new applications have been standardized, I believe that it may be possible for the NRC to complete the reviews, including the hearings, in approximately 24 months.

In another world, in another time, it might be different. But, here and now, the path forward for nuclear power safety, predictability, and growth seems clear: standardization. The benefits of detailed certified standard designs, early site permits or equivalent with much use of generic-to-a-design environmental impact statements, and standard COLs should be seriously considered.

What is my major concern today regarding a predictable schedule for new reactor licensing? It is if and when the NRC will receive a complete, high quality COL application.

In summary, the sociopolitical, financial, economic, technical, and regulatory framework for reactors in this country has changed dramatically since the last plants were designed, licensed, and built. This is the twenty-first century, and I can assure you that the NRC is much better at doing what it must do. Many of the old assumptions are no longer valid. The NRC is continuing to forge a new licensing and regulatory framework for today, for tomorrow, and for the future. The Commission and the staff of the NRC are meeting the challenge, indeed the demand, to do our job well. I am proud of the people I work with day-in and day-out, and their dedication to the safety, security, and the well being of the people of our country, indeed proud of the strength and stability of the institution we have forged together.

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No. S-06-003

Regulatory Perspectives on U.S. Nuclear Power Infrastructure - Current and Future

Massachusetts Institute of Technology
February 28, 2006

Peter B. Lyons
Commissioner
Nuclear Regulatory Commission

I was sworn in as a Commissioner a year ago, and I've been rapidly learning details of the Commission's operations since then. Based on my education during those months, I welcome the opportunity to share with you today some perspectives on the current and potential future of nuclear power generation in the U.S. from the Nuclear Regulatory Commission's point of view.

My previous career in national security at Los Alamos and then on Capitol Hill, underpinned by my graduate training, has led me to define national security in very broad terms - to encompass the military, the economy, the environment, and certainly to include our nation's energy supply.

There is no doubt in my mind that our nation will be challenged to meet its growing needs for electricity generation in future decades. I believe that the nation should encourage fuel diversity as it strives to meet these challenges, seek to minimize pressure on limited supplies of natural gas, and reduce its dependance on foreign energy sources.

For this new electricity generation, the nation will need to tap renewables as much as possible. But the intermittent character of solar and wind systems means that they cannot play a dominant role in supply of baseload electricity unless we invent new, very low cost, energy storage systems. Our large coal reserve provides another opportunity for expanded electricity generation, but significant expansion of that resource will depend on development of cost effective, low emission plants.

The only other potential source of significant new electricity generation within the next few decades is nuclear energy. But answers to many questions will dictate whether nuclear energy will play a strong supporting role.

In any discussion of nuclear power and the potential for new plant construction, we must always remember that the entire industry has a vital job to attend to first: safe and secure operations of existing plants. The public needs to be confident of ongoing safe and secure performance of existing nuclear plants to support the potential for new nuclear plants.

The NRC has the responsibility to establish and enforce the safety and security standards for all civilian applications of nuclear technologies. Its Congressionally mandated mission 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.

In my view, without the nuclear power industry's continued perseverance toward adequate safety and security, nuclear energy will not play a future role, and our nation will have an immense energy shortfall.

I presented a talk similar to this one at the 2005 American Nuclear Society annual conference. The theme of that conference focused on a "half century" view of opportunities for new nuclear power plants.

The requirement for safe and secure operation of our nuclear plants certainly will remain during that half century, or at least for as long as we operate nuclear plants. But my own view is that the time frame within which we will determine our nation's future capabilities in nuclear energy is far more compressed than half a century, perhaps a couple of decades at the most. Unless near-term progress is demonstrated in the United States within that shorter time window, which includes construction of a significant number of new plants, we may lose much of our technical capability to support nuclear energy using domestic resources.

The United States led the world's development of nuclear energy, but there hasn't been a new construction permit issued here since 1978. That dearth of new plants was driven by several factors, but its impact has been enormous. Our nation's capacity for new plant construction has had limited exercise and has partially atrophied. We are no longer the world's only leader in these areas. Today, we have enough of the infrastructure, both human capital and industrial capability, to recover, but we are in danger of losing these capabilities in the not too distant future.

However, it is evident that the nuclear power industry enjoys strong support from recent Administration and bipartisan Congressional actions. The visits last year of President Bush to Calvert Cliffs and of former President Carter to D.C. Cook, along with their endorsements for the future of nuclear power, helped to underpin the growing national confidence in the important role that nuclear energy can play. The President's signing of the Energy Policy Act of 2005 authorized a host of important new programs and opportunities for this industry, including production tax credits and loan guarantees. And the FY 2006 Appropriations Bill provided strong support for nuclear energy, including increased funding for the NRC to perform security reviews and new reactor licensing activities. Furthermore, although the exact numbers are subject to change, the NRC currently is expecting to receive applications[1] in late 2007 and in 2008 for Combined Construction and Operating Licenses to build and operate up to 17 advanced power reactors at 11 sites.

As I mentioned previously, this positive climate for new construction requires that the NRC and industry ensure the safety and security of existing plants. How will we accomplish our current safety goals and thereby provide the foundation for possible future growth?

First, the industry must maintain a clear focus on safe operations and assure no blemish on its stellar safety record - that no member of the public has ever been injured by any release from a civilian plant in the United States.

With this focus, the industry under the watchful oversight of the NRC must constantly guard against another serious incident like that encountered in 2002 at Davis-Besse when boric acid corroded through most of the pressure vessel.

The Commission needs to observe and report on industry's continued safety performance, as we further risk-inform and performance-base our regulations and implement our oversight processes. In general, industry's safety trends have shown improvements over the last decade.

The NRC revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants in 1999-2000. The new oversight process uses more objective, timely, and safety-significant criteria in assessing performance, while seeking to more effectively and efficiently regulate the industry. It also takes into account improvements in the performance of the nuclear industry over the past 20 years.

The objective is to monitor performance in three broad areas – reactor safety (avoiding accidents and reducing the consequences of accidents if they occur); radiation safety for both plant workers and the public during routine operations; and safeguards for protection of the plant against sabotage or other security threats. To measure plant performance, the oversight process focuses on seven specific "cornerstones" which support the safety of plant operations in the three broad performance areas. In addition to the cornerstones, the reactor oversight program features three "cross-cutting" areas, so named because they affect and are therefore part of each of the cornerstones.

The revised oversight process provides more information on plant performance than in the past, and the information is available on a more frequent basis. This information is placed on the NRC's Web site.

The public credibility of this assessment process rests both on each plant's full commitment to accurate and unbiased performance indicator data collection and reporting, and on the dedication and knowledge of NRC resident and regional inspectors. In this respect, both the industry and the NRC work toward maintaining public confidence in this process.

In addition to public assurances on safety and security, nuclear power will not advance unless the industry and the public have confidence that the Commission's licensing procedures are well understood, incorporate significant public input, and result in timelines. The Commission's performance on license renewals, power uprates, and new plant licenses will be measured in this process.

License renewals began with Calvert Cliffs in 2000, and now the Commission has renewed licenses at 39 plants. Renewal applications are currently under review for 12 plants. With few exceptions, the Commission has processed these renewals within about 22 months.

However, where renewal applications were not of sufficient quality, the Commission has not hesitated to return a licensee's application package or to delay its approval until quality had improved – applications for four units have fallen into this category.

Power uprates have also been processed on a generally reliable schedule by the Commission, even though some of the larger uprate requests require very careful evaluation of the effect of increased power on internal reactor components. This is currently an area of careful study at the Commission.

Turning now to the future, but still focusing primarily on the broad area of reactor safety, licensing of the first new reactors will be a process watched carefully by all stakeholders, both public and private. Here the Commission will use an untested new process described in our regulations. This framework was instituted in 1989 and provides for a combined construction and operating license or COL. The creation of a COL process was to address the uncertainty inherent in the historical process of permitting construction of a plant without the full assurance it would be granted an operating license.

This new framework also includes the Early Site Permit or ESP process and the Standard Design Certification. Both the ESP and the design certification may be referenced to simplify a utility's application for a COL. The overall goal of the COL process is to provide a more stable, efficient, and predictable regulatory framework for utilities that might wish to pursue a new reactor license. At the same time, the Commission has been careful to include appropriate opportunities for public input throughout the parts of the COL process.

The ESP process allows early resolution of site-related issues and effectively allows a utility to "bank" a site for future construction. Three applications have been received, for the North Anna, Clinton, and Grand Gulf sites, and the Commission is scheduled to issue final decisions in 2007 for Clinton and Grand Gulf. The North Anna application was originally on a similar track but was recently revised by the applicant and a new schedule is being determined.

The first standard design certification was issued in 1997. Today four advanced designs are certified, the latest being the just-approved Westinghouse AP1000, another certification review is in progress, and others are expected to be filed. The Commission has estimated times for completion of a certification to range from 42 to 60 months depending on the complexity of the design and its departure from previously certified designs.

The COL application process enables a utility to reference an ESP and a certified design to expedite the process. If both the ESP and certification are in hand, the review and hearing process for the combined license can be anticipated in less than 30 months. Nevertheless, the first utility that tests the COL procedure will be moving into uncharted waters, but into an area that the Commission has anticipated and is prepared to address.

As seen here, the NRC currently has been actively engaged with industry and potential applicants who have expressed serious interest in submitting COL applications in the 2007 and 2008 timeframe. Due to its ongoing dynamic nature, this table represents a huge challenge to NRC's budgeting, resourcing, and staffing plans.

One component of these new licensing activities involves international activities. I am highly supportive of the NRCs current plans to work with French and Finnish regulatory officials on our licensing review of the Evolutionary Pressurized Reactor (EPR). In the coming months, the Commission will consider participating in an even more expansive multi-national design approval program. I feel strongly that we must participate in the development of an international process soon, as a success will provide greater assurance that plants built in other nations will benefit from the regulatory practices and demands that we impose on our own plants. And if we delay in engaging other countries, I'm afraid that some type of program will evolve without us. It is far better for us to be involved with global standardization now, than to be faced with some form of international standard that does not meet the regulatory standards we demand.

Turning now to the broad area of security, I first must note that before 9/11 our nation's nuclear power plants were probably the most secure element of our civilian critical infrastructure. After September 11, 2001, the NRC and our licensees faced the need and challenge of progressively enhancing the security and preparedness of nuclear facilities and materials, while simultaneously continuing to perform, with undiminished attentiveness, the requisite safety mission. Fortunately, we were prepared to do both. During the last 4 years, the NRC staff has worked very closely with the Commission and our licensees in addressing an array of issues that are vitally important to the safety and security of the American people. The same is true of the nuclear industry and our sister Federal agencies. Most of the heavy work has been done; we are now doing the painstaking job of providing closure to the security framework through rulemakings.

Security was further enhanced by passage of the Energy Policy Act of 2005, which provided specific direction and provisions, some of which have been long sought by the agency. For example, the Act authorizes the Commission to allow security personnel at licensed facilities to carry and utilize a broader class of weapons. The Act also required the Commission to issue orders requiring fingerprinting, for criminal history purposes, of broader classes of individuals.

The NRC and the industry need to know the consequences of potential terrorist events. In this regard, the NRC has conducted detailed, site-specific engineering studies of a number of typical nuclear plants to assess their capabilities to withstand an attack using a commercial or general aviation aircraft as a weapon. Many other damage scenarios were addressed by licensees, as required by the 2002 and 2003 orders. Further analyses were recently performed, including spent fuel pool structural analyses to provide further insights regarding structural robustness of the spent fuel pools. The combination of results, including industry assessments, provide a sound framework for decision-making and for determining if additional analysis is needed.

From the studies we have conducted to date, we continue to believe that the likelihood is low that airplane or realistic vehicular bomb attacks would damage the reactor core or the spent fuel pool and cause a release of radioactivity capable of affecting public health and safety. Moreover, mitigative strategies are available to protect the public in the unlikely event of a radiation release, and additional practical enhancements of the mitigation capabilities are being analyzed and considered.

After 9/11, the NRC initiated a three-phase effort to conduct assessments of plant safety and security measures. Phase I assessments were done in accordance with the February 2002 Order which, among other things, required nuclear power plant licensees to identify readily available mitigative strategies addressing a range of potential scenarios that may result in the loss of large areas of nuclear power plants due to a large explosion or fire. As a result of these assessments, licensees were required to implement mitigative strategies, and the NRC staff has been and will continue to verify licensee compliance with the requirements of this Order.

The schedule calls for completion of all Phase I actions this year, and the documentation of these actions into licensee's security plans in early 2007. The results will then be a component of our established and stable regulatory framework.

Phase 2 focuses on additional independent spent fuel pool assessments. The NRC has completed site-specific independent assessments at each nuclear power plant to identify additional measures or strategies to mitigate the consequences of a wide range of terrorist attacks involving spent fuel pools. The assessments began in July 2005 and were completed in November 2005. We expect that the book will soon be closed for Phase 2.

Phase 3, which is aimed at possible measures beyond the scope of the February 2002 Order, focuses on Independent Reactor Core and Containment Assessments. As of today, 38 site assessments have been completed. NRC and its licensees are performing these site-specific, independent assessments, at each of the 64 nuclear power plant sites to identify additional measures or strategies to mitigate the consequences of a wide range of potential terrorist attacks. These assessments began last October and are scheduled to be completed at all sites by April 2006. The NRC's independent assessments include reviews of each licensee's identification of further alternative means for achieving safety functions in scenarios that might disable the normal front-line and backup systems used to achieve safety functions. Completion of the ongoing assessments is a necessary and sufficient condition to provide closure to the enhanced safety and security framework of U.S. nuclear power plants.

Going back to the nuts and bolts of physical protection, licensees have made very significant improvements in their defensive capabilities and strategies, and concurrently we have made significant improvements in force-on-force exercises and evaluations.

Some of the security enhancements are obvious as one approaches any plant perimeter such as this intrusion barrier. Many more changes are less obvious. They reflect improvements in internal operations, procedures, and physical arrangements. They also involve carefully negotiated and tested protocols between the NRC and local, state, and federal responders. Airborne threats are addressed through the operations of the Department of Homeland Security and the North American Aerospace Defense Command (NORAD). With these many enhancements, our nuclear plants are even more secure today.

Prior to 9/11 the NRC conducted mock attacks to test the capabilities of the licensees' security program. After 9/11 and following a successful pilot program, we implemented a full program of enhanced force-on-force exercises and evaluations. These enhancements included increasing the standards for mock adversary force physical fitness, training, and knowledge of attack strategies to emphasize offensive capabilities. The NRC will continue to oversee and evaluate approximately 22 force-on-force exercises each year, or about once at each site every three years, and each licensee also conducts their own drills each year.

To summarize our actions on security, I believe that we have established, using a risk-informed approach, the key NRC requirements needed to provide added assurance of the security of civilian nuclear facilities and materials in the United States. We started early, from a sound and exercised base, and progressed methodically. At the same time, many sister Federal agencies, especially DHS, have been engaged in bolstering homeland security and protecting the Nation's critical infrastructure. We have developed strong ties with these agencies, resulting in improved national capabilities.

All of us have a common purpose -- to protect our country, its people, and its way of life -- and we are working more closely together than ever before. We have worked extensively with the Department of Homeland Security and have adopted the National Response Plan, increased involvement with the development of the Department of Homeland Security National Infrastructure Protection Plan and partnered with DHS to conduct comprehensive reviews of offsite response. The NRC staff continues to interact with Department of Justice's Joint Terrorism Task Forces in the field and has enhanced NRC coordination and communication of threat intelligence and suspicious activities through increased access to various reporting sources. We have also established additional secure communication capabilities at NRC to facilitate timely and effective crisis communication with Federal partners. There can be no question that our civilian nuclear power plants are among the most secure civilian sites in the world.

Here at MIT, and as further highlighted by recent media interest, I think I should mention our nation's test and research reactors, which I strongly believe are a vital component of our nation's nuclear infrastructure in support of nuclear power as well as in the use of nuclear materials for medicine and industry.

Prior to 9/11, security plans and procedures were required for research reactors. These requirements employed a defense-in-depth approach that was geared to the specific radiological hazard for each reactor, and that was aimed at detecting, delaying, assessing and initiating responses to security events. Subsequent to September 11, the NRC ensured that numerous additional security-related measures were instituted at research reactors to enhance these defenses against facility sabotage or theft of nuclear material. In addition to these actions, the NRC assessed the security of the research reactors to further determine whether any additional security measures were warranted. Results to date indicate that there are no credible scenarios that could result in significant radiological consequences to the public.

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. 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.

The NRC maintains a thorough oversight program of all licensed research reactors. This oversight program includes safety and security inspections and evaluations to ensure that the public is protected. NRC also evaluates the current threat environment in coordination with the Department of

Homeland Security, the FBI, the intelligence community, and State and local law enforcement agencies.

As I'm sure you are aware, in 2005, the ABC television network aired a "Prime Time" story related to research reactor security and, in fact, prominently discussed the MIT research reactor facility. The NRC staff has evaluated the questions raised by ABC regarding research reactor security.

The NRC reexamined licensed research reactor security plans, procedures and systems to determine if the required security measures were in place. One example of ABC's concerns was that some doors to buildings housing reactors were open and unmonitored. Upon checking, the specific doors in question were found to be publicly accessible classroom/office buildings and were not required to assure adequate security of the reactor. Another example was ABC's assertion that so-called "guards" were not always present or alert. Our review determined that the specific traffic control and other campus personnel identified by ABC were not required by the approved security plans or for any other regulatory purpose. Based on our review of all questions raised by ABC, in one case we determined that implementation of a security requirement was deficient, and although it was considered to have low significance, the NRC has ensured that corrective action was taken.

Each specific concern for each research reactor was evaluated through NRC's allegation review process. Based on these evaluations, NRC continues to conclude that security plans, procedures and measures are adequate to protect public health and safety from the potential radiological effects of research reactors. NRC will continue to assess information from all sources to ensure adequate protection of public health and safety.

However, we have not limited our review to only those research reactors shown in the ABC story. We also issued letters to every research reactor licensee to obtain additional information and emphasize our expectations for maintaining effective security in the current threat environment. In these letters, we requested each licensee to verify its implementation of the previous site-specific security measures and provide additional details. The information we requested will help the NRC to re-validate how the existing security requirements, as supplemented by the additional security measures conveyed to the research reactor community after 9/11, are being implemented to help protect public health and safety.

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

Finally, I'd like to address another significant challenge for both the industry and the NRC: the impending loss of many of our most experienced employees who are nearing retirement, and the attendant loss of the historical and collective lessons that they have learned. It isn't sufficient to just hope that these lessons will have been passed on to younger generations. There must be proactive actions to mentor our less experienced employees to pass on the important values that are essential to continued safe use of the nuclear energy option.

Human capital in the nuclear arena is a subset of a much larger national issue. I have serious concerns with the current state of our nation's workforce preparation for science and engineering in general. This issue was recently discussed in significant detail in a comprehensive report issued by the Task Force on the Future of American Innovation.

That report noted that the number of science and engineering positions in the U.S. workforce has grown since 1980 at almost 5 times the rate of the U.S. civilian workforce as a whole. But in contrast, the number of science and engineering degrees earned by U.S. citizens is growing at rate below the growth in the total U.S. civilian workforce. Further, our preparation of qualified science and engineering graduates is falling further behind other nations with each passing year.

One measure of this issue, collected in the compendium of Science and Engineering Indicators compiled by the National Science Board, is the ratio of initial university science and engineering degrees to the population of 24 year-olds. In 1975, this ratio for the U.S. exceeded most of the surveyed nations, except Finland and Japan. By 2000, our ratio was exceeded by 16 nations, including again Finland and Japan, plus France, Taiwan, South Korea, UK, Sweden, Ireland, and Italy, to name a few.

The magnitude of this national issue was highlighted when the distinguished Norm Augustine testified before the U.S. House of Representatives on behalf of the recent National Academy of Sciences report entitled, "Rising above the Gathering Storm," which discusses the loss of competitive edge by the United States because of a lack of investment in education and research. My good friend Chuck Vest, who may be pretty well known here, helped develop that report. To quote just a bit of Norm Augustine's very sobering testimony, he said:

> It is the unanimous view of our committee that America today faces a serious and intensifying challenge with regard to its future competitiveness and standard of living. Further, we appear to be on a losing path.

Recently however, a package of three bills, known as the "PACE" Act, for Protecting America's Competitive Edge, has been introduced in the Senate with widespread bipartisan support (with more than 60 co-sponsors so far, including both the Senate Majority and Minority Leaders). If passed, this Act will support and complement the President's American Competitiveness Initiative (ACI) announced during his State of the Union address that, among other things, will substantially increase investment in research and development, education, and tax incentives to encourage innovation. Of particular interest here at MIT would be the proposed doubling of basic research funding over 10 years starting with an average of 9.6% funding increase in FY 2007 for the DOE Office of Science, the National Science Foundation, and the National Institute for Standards and Technology.

Turning now to the NRC's specific human capital challenges, I've been very impressed with the range of staff development and recruiting programs that are underway within the NRC. The Commission has provided fellowships and scholarships, as well as a number of cooperative education programs. We have strong participation in our Leadership Potential Program, our Nuclear Safety Professional Development Program, and in our Senior Executive Service Candidate Development Program. In past years, the Agency met its targets for staff recruitment.

Legislation introduced by the U.S. Senate Environment and Public Works Committee and incorporated in the Energy Policy Act of 2005 will provide additional tools to develop and attract qualified new staff. But it remains to be seen if we can meet our ambitious goal for this current year and similar goals in the future.

Knowledge management is an important part of staff development, and these programs are being emphasized at the Agency. By knowledge management, I mean the process by which knowledge gained over decades of work by senior scientists and engineers is translated, retained, and made available in ways that facilitate its transfer to new generations of workers.

I'm very pleased that the Commission sponsors a wide range of programs to encourage new graduates in specialties appropriate to our own needs. But the issues of workforce development and human capital are hardly unique to the Commission. The entire industry faces severe shortfalls. And if the advertised rebirth of new plant construction occurs, there will be increased needs and increased competition for new staff. While any new plant construction will help inspire more students to view nuclear technologies as a secure long-term career choice, it's unlikely that the supply of new candidates can increase very quickly.

Whether you are a member of the faculty or a student at this premier scientific and technology university, I challenge each of you to devote some time over your career to actively helping to increase secondary-level student interest in science and technology careers. All of us need to redouble our efforts in conveying to these students the excitement and opportunity that await them in these many fields, and of their importance to the future of our country.

At this point in the lecture, I'll insert a public service announcement: Whether you are a student, a member of faculty, or otherwise have a technical background, this message is for you. I can tell you from personal experience that public service can be an immensely satisfying component of any technical career. At the NRC we have had, and will continue to have, technical challenges covering the widest range of nuclear technologies, in the fields of power reactor and industrial and medical uses of nuclear materials. In addition, we continue to need fresh perspectives of technically knowledgeable people to contribute to the development of public policy for the safe use of nuclear technologies. The NRC was recently honored as one of the best federal agencies for employee satisfaction and I highly recommend it as a career choice or at least as a component of any technical career. Whether you are interested in regulatory aspects or in the research foundations for our regulatory decisions, there are exciting opportunities for you at the NRC.

I'd like to close with discussion of the challenges the Agency, the industry, and the public will face if the number of reactor license applications approaches the levels announced by industry. Industry has briefed the Commission on their tentative plans for COL license applications, plans that total about 17 reactors, each with the stated goal to be operational by 2015.

The NRC is going to be incredibly challenged to respond to any number of applications that approaches the plans advertised by industry. On the one hand, the NRC must and is doing all it can do to build the human capital resources to accommodate this number. But there are many actions that industry should be considering if their expectation is that the NRC can successfully evaluate this number of licenses.

Industry must maximize standardization of licensing applications, designs and construction activities so that the NRC can leverage, to the extent practicable, similar standardization in the Agency's review process. In addition, COL applications must meet very high quality standards. The NRC will not compromise our review standards to expedite approvals, and the burden is on the applicant to provide the required level of quality.

In summary, the industry's performance, as well as the Commission's regulatory oversight, will be carefully observed by the public. Only if both the industry and the Commission demonstrate strong performance can public confidence be maintained at a sufficient level to permit an objective and reasoned public dialogue on the future of nuclear energy in this country. The foundation for retaining the nuclear energy option in the future rests squarely on the continued safe nuclear plant performance of the current operating reactors and continued strong and independent NRC oversight. In addition, it also depends on improved security and stable NRC licensing processes with appropriate public input. Meeting these goals in as public a manner as possible, while balancing openness and information security, is absolutely necessary. Well-informed citizens are essential to better understanding operations, risks, and benefits involving the nuclear energy option.

Thank you for the opportunity to share these thoughts with you today, and I'd be happy to take any questions you might have.

1. As of February 23, 2006 - expected COL applications include:
Dominion (1 ESBWR at North Anna site)
Duke (2 AP1000s at TBD site)
Progress (2 AP1000 plants at Harris site)
Progress (2 AP1000 plants at TBD site)
NuStart (2 AP1000s at Bellefont site)
NuStart (1 ESBWR at Grand Gulf site)
Southern Nuclear Company (1 AP1000 at Vogtle site)
Constellation (2 EPRs at Calvert Cliffs site)
Constellation (2 EPRs at Nine Mile Point site)
Entergy (1 ESBWR at River Bend site)
SCE&G (1 AP1000 at Summer site).

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Office of Public Affairs Telephone: 301/415-8200
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No. S-06-004

"FACING SAFETY AND SECURITY CHALLENGES, A NATIONAL AND INTERNATIONAL PERSPECTIVE"

Chairman Nils J. Diaz
U.S. Nuclear Regulatory Commission
Given at IAEA Conference - Moscow

February 28, 2006

Good morning. It is indeed a pleasure and distinct honor to be here among fellow regulators and distinguished guests, to share my views on effective nuclear regulatory systems, with a few examples specific to the U.S., and a global perspective. We are all, one way or another, preparing to discharge new responsibilities in a changed and changing world; preparation appears to be turning quickly to implementation.

First, I want to thank the International Atomic Energy Agency (IAEA) for organizing this important conference. I especially want to thank IAEA Director General Dr. Mohamed ElBaradei for his direct role in making this meeting of senior nuclear regulators a reality, and Deputy Director General Tomihiro Taniguchi and the IAEA staff for their hard work and commitment to the effort. I would also like to express my sincere appreciation to our Russian colleagues, particularly Chairman Konstantin Pulikovsky, First Deputy Chairman Andrei Malyshev, for their extraordinary efforts in hosting this meeting, which is dedicated to the key role that national regulatory authorities should continue to play in society, supported by effective international bodies. And thank you, Mr. President, for laying out the necessary and sufficient components of an effective regulatory framework that will serve the international community of nuclear regulators.

I am confident that the resulting deliberations and recommendations will contribute to the effectiveness and sustainability of national regulatory systems, to new regulatory approaches for the use of advanced technologies and innovative designs, and to the development of additional instruments and mechanisms for cooperation among regulators in international forums.

Before I enter into the main topics I want to share with you, I would like to make a comment on the issue of nuclear proliferation, or better, on the issue of assuring nuclear non-proliferation. It is now unmistakably true that the overriding necessity to achieve nuclear non-proliferation - as a fait accompli - has become a dominant issue in international politics, and of course, at the IAEA. Its importance to world peace, trade, and geopolitical activities cannot be overstated. Yet, I will dare to say, that in a grand scheme of world prosperity, commerce, and international law, proliferation should not be more important than nuclear safety and security. In fact, in a world where abundant, economic, and well distributed energy becomes a global cornerstone, safety, security, and non-proliferation are interdependent components of a better and reliable framework for peace and prosperity. Worldwide nuclear non-proliferation efforts should be integrated with the safe and secure utilization of civilian nuclear power, and not become its deterrent.

This international conference on "Effective Nuclear Regulatory Systems" is more than a gathering of senior regulators and of nuclear technologists; it is truly an international assembly of those who implement nuclear safety, security, and emergency preparedness. The sessions should have a definitive underlying theme and purpose that support the objectives of the conference. A common understanding of the purpose of regulation in general, and nuclear regulation in particular, should provide the connectivity between every one of us, independent of country or organization. A good starting point for the common understanding of regulation would be:

> Regulation is done for the well-being of our people, for the common good, with full consideration of the National interests, and of International law and agreements.

Nuclear regulation is a disciplined national tool for establishing predictable safety and security frameworks. It works by establishing and improving technical and legal structures to define the acceptable safety case that serves the public interest.

Senior nuclear regulators, you and I, are coming together in Moscow, in winter, in 2006, to make a statement regarding our responsibilities and to deliver a series of products, sustained by a common understanding of nuclear regulation. Moreover, we are here because we care about our nations and because we can and want to work together, better. In this regard, I present for your thoughtful consideration here, as a purpose, the objective stated by the U.S. Nuclear Regulatory Commission in its current Strategic Plan:

> 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.

With that purpose in mind, it becomes clear why our presence here today is important. In fact, as inevitable as day and night, there is supply and there is demand. Unfortunately, there are also imbalances that may occur in supply and demand. The world is again experiencing that almost forgotten enemy: expensive and/or unreliable energy supply. Many times we have seen that society is disrupted and people suffer when energy is costly, scarce, or not available. The solutions to economic and reliable energy supply are surely important worldwide. In America's case, dependence on energy is somewhat unique; solutions are needed for the short term and solutions are needed that will endure the test of time and crises. Therefore, the U.S., like many other countries, is reviewing the strategic, economic, and environmental considerations of the Nation's overall energy supply and openly considering the contributions of nuclear power to meet its present and future energy needs. In fact, in America, President Bush and the Congress have taken positive steps to ensure that America's energy mix includes the reliability of supply, the environmental benefits, and the steady costs that are now ascribed to operating nuclear power plants. Maintaining the requisite focus on safety and security, the NRC has the obligation and responsibility to respond to the needs of the country. Although our particular needs may differ, you are surely being asked to be ready to implement a set of effective regulatory tools that are responsive to the energy, economic, and security demands of the present and the near future. I believe that we can agree that every Nation of the world would be better served by reducing imbalances in the energy supply and demand, and by supporting safe, economical and environmentally friendly electrical energy supply that meets the global demand.

Furthermore, our presence here is important because nuclear regulatory authorities have a key role to play in resolving the effectiveness and sustainability, indeed the predictability and reliability of regulatory decision making, and therefore, the role that nuclear power could play. Of course we, as regulators, have important duties regarding security and radiological materials safety in addition to reactors. We all need the instruments; mechanisms; resources; and the international, multinational, and bilateral cooperation that will strengthen our capability to serve our people better with regulatory resolution of issues, with openness, and credibility.

I want to summarize for you where the U.S. is in two areas that are important to the viability of nuclear power generation: safety and economics. These two interdependent factors have seen major improvements in the last 15 years with respect to the consideration of nuclear power in the energy mix for many countries. I believe that safe, reliable, and secure nuclear energy has been and can continue to be part of the solution to energy security and environmental stewardship, and thus contribute to the well-being of all our people. We have played and should continue to play a key role in ensuring the safety of nuclear installations, with the technical know how and regulatory practices of today for today's needs.

For over twenty years, specifically during the decades of the 1970s and 1980s, the economics of nuclear power did not fulfill the early expectations of the U.S. or the world. The reality is that commercial nuclear power did not have much of a chance to meet expectations during those years. In the U.S., and most other places, nuclear power deployment took place during the worst possible time for large capital-intensive projects. Financial, technical, or regulatory predictability was lacking.

The economic situation for nuclear power plants has changed significantly and the prospects for new plants have become more promising. Low inflation and low interest rates have been the norm for the last few years, and low production costs of nuclear generated electricity, including fuel, are now frequently highlighted in the press and in the halls of government. Today, there is stability in regulatory requirements. The U.S. plant capacity factor and total electrical generation are sustained at, or near, all time highs; nuclear production costs, at $0.0168/kw-hr, are now lower than coal.

I discussed economics as a necessary part of the global nuclear scenario, but assurance of safety is an essential component. The sociopolitical reality is that nuclear power needs to be safer than other forms of generation. In fact, it needs to be "safe" in both actual and perceived terms. To achieve "safe" status, the U.S. nuclear power industry needs to over-achieve both in actual safety performance and in how safety is regarded. According to the performance safety indicators used by the NRC, the U.S. nuclear industry has achieved overall better-than-ever performance. Beyond individual safety indicators, I can tell you with confidence that the U.S. nuclear power industry is performing with adequate safety margins, and that NRC oversight is resulting in reasonable assurance of the protection of the public health and safety, the environment, and national security. One of the key responsibilities of nuclear regulators is to define "safe enough." We all realize that there is no such thing as zero risk; therefore, we need to establish adequate safety margins while enabling the safe use of nuclear technology.

The improved industry performance has enabled the NRC to initiate and implement reforms that are progressively more safety-focused. A look at license renewal is indicative of the profound changes made by the Commission to regulatory effectiveness and efficiency. U.S. nuclear plants were initially licensed for 40 years, and license renewal authorizes an additional 20 years of operation after safety requirements for passive components and aging are met. The picture for the survival of nuclear power in the U.S. was not pretty in 1997; predictions of the accelerated demise of half of the licensed plants were abundant. The Commission undertook the task of reviewing the requirements for protecting public health and safety in deciding the renewal of licenses, and thus, served the National interest as articulated in the Commission's authorizing legislation. The resulting improvements in the license renewal process that the Commission put in place, along with changes to the hearing process, assured the Nation that a fair, equitable, and safety-driven process would be used for those applying for extension of their licenses. Today, 39 licenses have been renewed and 12 are being processed. Twenty-seven other licensees have announced their intention to apply for renewal of their licenses. The NRC is completing these license renewal approvals in approximately 22 months after receiving the applications. This process is focused on verifying the adequacy of licensee aging management programs. Moreover, the program has resulted in significant investments by industry that directly contribute to enhancing operational safety. In today's energy environment, the 20-year license renewal of 39 nuclear power plants provides a great value to the United States in terms of energy, national, and economic security, as would be the probable renewal of another 39 nuclear power plant licenses in the near future.

In today's world, to ensure protection of public health and safety, the assurance of security is essential. I believe that the NRC has established, using a risk-informed approach, the key regulatory requirements needed to provide added assurance of the security of civilian nuclear facilities and materials in the United States. We started early, progressed methodically, and are currently incorporating requirements into our regulations. These include three important security rulemakings planned or underway to codify security requirements for power reactors. The first is the rulemaking on the design basis threat for radiological sabotage, and a final rule will be issued later this year. The second rulemaking will amend the power reactor security regulations to align them with the series of orders the Commission issued following September 11, 2001, and to ensure safety-security interface issues are properly considered in plant operations. Finally, the Commission's expectations on security design for new reactor licensing activities are to be codified in a third rulemaking by September 2007. The expectation of the Commission is that the lessons learned by the agency and reactor licensees pre- and post-9/11 should be considered by the vendors at the design stage. We have learned much, and I believe improvements can be realized without major design or construction changes.

With this backdrop, I would like to discuss what the NRC is facing and doing to address the renewed commitment of the U.S. Administration and Congress to civilian nuclear energy as a means to address the demand for economic and environmentally benign electric power, and the expressed intentions of the U.S. nuclear power industry. To date, 11 potential Combined License (COL) applications for a total of 17 new nuclear power plant units, distributed among the three major reactor vendors now competing for the U.S. marketplace, have been publicly announced. They appear to be "bunched up" for submittal and review in a short period of time. There are, of course, significant infrastructure and logistic issues to be resolved by the industry and by the NRC, and a short time to do it.

In order to review effectively multiple COL applications in parallel, the NRC staff is now preparing to implement a design-centered approach for reviews of COL applications, to the extent possible, for as many issues as possible. This approach involves the use, for each issue, of one review and one position for multiple applications. It could also be called the "one-for-all" approach. It is ready for use now; however, it needs the nuclear industry's commitment. One-for-all is one thorough, comprehensive, NRC safety evaluation to be used repeatedly, as appropriate. Using the design-centered approach, the NRC staff would use a single technical evaluation to support multiple combined license applications for the same technical area of review, as long as the applications standardize the licensing basis to a level that would make this approach viable. For technical review areas amenable to this approach, the staff can complete the evaluation for a "reference" case, can determine if the design proposed by other applicants is the same as the design reviewed, and proceed to issue the evaluation without further review. Let me emphasize, that for each certified design, standardization is the key to making this approach work. Standardization is everybody's business in reactor licensing.

The design-centered approach could also be applied to parallel reviews of a design certification application and COL applications referencing the design. For example, NRC reviews for the ESBWR and the EPR designs are likely to be conducted in parallel with reviews of the first few COL applications referencing these designs. The NRC could proceed with its review of each design and issue a safety evaluation report with open items, just as was done in the case of the AP1000 and earlier designs. Using the design-centered approach, the resolution of generic open items in the NRC safety evaluation report could be coordinated between the vendor and the applicants for COLs referencing the vendor's design. The resolution of these generic issues could then be incorporated into the design and included in the rulemaking certifying the design. In this manner, they would be available to future applicants referencing the design.

I am confident that applying the design-centered approach to parallel design certification and COL reviews, and relying on disciplined standardization, will result in a better, more detailed, and more thorough safety evaluation for each design. When an applicant references a standard design certified by rulemaking, all design matters within the scope of the design certification rule have been resolved using a fair and equitable process and need not be readdressed in the COL proceeding. The design-centered approach could also lead to a significantly higher level of efficiency in the licensing process, thereby reducing the amount of staff resources necessary to conduct each review.

Could it be done differently? Of course it could, and the law clearly says so. In another world, in another time, it might be different. But, here and now, the path forward for nuclear power safety, security, predictability, and growth seems clear: standardization.

The worldwide expectation for large scale deployment of nuclear power is approaching decision making time in many places. However, uncertainties remain. The solution to new reactor deployment includes thorough, timely, and safety-focused decisions by nuclear regulatory authorities. I believe that we would agree that this time around nuclear power plant deployment should be carefully planned, and key issues and interfaces, including regulatory issues must be resolved at the front end, on budget and on schedule, with all the safety and engineering know-how developed and learned over the last 25 years. Obviously, there are many ways and various scenarios on how we make decisions in the regulatory process. Yet, it is essential that we ensure regulatory predictability by handling applications in a manner that is expeditious, in a manner that assures that decisions on safety and security are clear, and in a manner that is fair to all parties.

We should be ready to utilize fully international and multinational resources, including technical capabilities and research efforts, to deal with the realities of the increasing "internationalization" of nuclear technology. We must recognize that changes in the marketplace, technology, and regulation have taken place; international partnerships of industry and international partnerships of independent regulators are needed to make a difference.

At the same time, we should recognize that the world's regulatory authorities and nuclear operators need to maintain a steadfast focus on the safety and security of existing nuclear power reactors. In order to meet this challenge and the added burden of new reactor licensing and construction, innovative approaches will need to be considered to make the best use of regulatory and industrial resources. It is frequently stated by the IAEA that the safety and security of nuclear reactors, in many respects, should have no borders. We need to increase effectiveness by adding international solutions to issues, as appropriate.

As a key example of an international solution to a global issue, the U.S. Department of Energy recently announced a Global Nuclear Energy Partnership (GNEP) as a comprehensive strategy to increase U.S. and global energy security, encourage clean development around the world, reduce the risk of nuclear proliferation, and improve the environment. GNEP is intended to develop and demonstrate new and improved proliferation-resistant technologies to recycle nuclear fuel and reduce waste. The U.S. will work with other nuclear nations to develop a fuel supply and services program for developing nations. In return, this would necessitate their commitment to refrain from developing enrichment and recycling technologies. In the 1980s, "do it once, do it right, do it internationally," became a mantra of the industrial sector in the European community. This sounds like a usable path for developing meaningful effective and efficient approaches for new technologies, including their regulatory treatment.

We will share four days in the beautiful city of Moscow; the cold weather only highlights the warmth of our relationships and the strength of our purpose. Some worry that our differences would impede lasting and effective solutions and that turf battles would diminish the benefits we could accrue from converging on safety and security practices and predictable decision making. I disagree. It matters not whether your entry point or outcome is through the IAEA, or the Nuclear Energy Agency (NEA), or you used tripartite or bilateral agreements, or multiple combinations thereof. We need them all, and I believe we use them all, and will need them even more in the future. What matters is the resolve of nuclear regulatory authorities to ensure fair, predicable, safety-driven outcomes for the well-being of our people, for the common good, enabling the safe and secure use of nuclear energy and radioactive material for beneficial civilian purposes. Furthermore, it matters that international and multinational agencies provide strong and sustained support to the efforts of nuclear regulatory authorities.

I am confident that our expectations of this conference will become a reality, with increased regulatory effectiveness and responsibility, by addressing key challenges and strengthening nuclear safety and security through lasting partnerships.

Thank you.

IN RESPONSE, PLEASE
REFER TO: M060208A

February 22, 2006

MEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON NMSS PROGRAMS, PERFORMANCE, AND PLANS - MATERIALS SAFETY, 9:30 A.M., WEDNESDAY, FEBRUARY 8, 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on accomplishments and challenges concerning materials safety in the Office of Nuclear Materials Safety and Safeguards (NMSS). The Commission commends the staff on its significant and high-quality accomplishments, and it appreciates the staff's thoughtful and diligent planning for future work. Additionally, the staff should keep the Commission informed of its resource needs to update guidance or regulations as part of its planning to meet future challenges.

The staff should hold its forthcoming paper on amending the regulations for *in situ* leach facilities until it can incorporate Commission guidance on ways to move forward in a new regulatory structure concerning groundwater protection at these facilities.

The staff should keep the Commission informed of the status and progress of rulemakings mandated by the Energy Policy Act of 2005 (EPAct). At the meeting, the staff informed the Commission that a rulemaking for the implementation of Section 656 provisions in the EPAct may be delayed because more time may be needed to accommodate both an OMB information collection review and a public comment period on the draft rule. The staff should brief the Commissioner Assistants on this potential delay and provide the Commission with an alternative schedule for completion of this rulemaking, including options for completing it by the statutory deadline.

The staff should continue to keep the Commission fully informed of progress in working with the States and other stakeholders in meeting NRC's responsibilities for regulating naturally-occurring and accelerator-produced radioactive material (NARM).

The staff should keep the Commission informed on the current issues, staff activities, and staff readiness in the area of fuel reprocessing. Specifically, the NMSS staff should coordinate with the Office of Nuclear Regulatory Research on its paper on the regulatory impact of the closed fuel cycle proposed by DOE and provide it to the Commission as soon as possible.

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

IN RESPONSE, PLEASE
REFER TO: M060208B

February 17, 2006

MEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON RES PROGRAMS, PERFORMANCE, AND PLANS, 1:30 P.M., WEDNESDAY, FEBRUARY 8, 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on accomplishments and challenges in the Office of Nuclear Regulatory Research. The staff should keep the Commission fully informed of efforts to complete the NRC fuel clad research projects jeopardized by recently announced facility closures at Argonne National Laboratory.

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

IN RESPONSE, PLEASE
REFER TO: M060206A

February 17, 2006

MEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON MATERIALS DEGRADATION ISSUES AND FUEL RELIABILITY, 9:30 A.M., MONDAY, FEBRUARY 6, 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff and by representatives of the nuclear power industry on issues and research related to materials degradation and fuel reliability. The Commission supports the proactive approach both industry and NRC staff have taken to identify and understand failure mechanisms, predict future degradation, and address them before they become safety significant.

The Commission supports working collaboratively in the area of materials reliability research, whenever appropriate, with the Electric Power Research Institute (EPRI) and the international nuclear community in order to gain the maximum benefit from NRC's research investment. The Commission encourages the staff to seek additional means to disseminate research results and their application to nuclear safety.

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

February 22, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-05-0197 - REVIEW OF OPERATIONAL PROGRAMS IN A COMBINED LICENSE APPLICATION AND GENERIC EMERGENCY PLANNING INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA

The Commission has approved the use of the license conditions proposed by the staff for the "Operational Programs Reviewed in a Combined License Application" (COL) listed in Tables 1 and 2 of Attachment 1 to SECY-05-0197. The staff's Recommendation 1.c. to include a license condition that specifies that "the licensee shall make available to the NRC staff..." should be understood to mean that "the licensee shall submit to the NRC staff..." as explained in the staff's discussion of this license condition on Page 7 of SECY-05-0197. The Commission approves using the Standard Review Plan (SRP) update process to identify additional operational programs. The staff should inform the Commission of the identification of such programs through information papers. The staff should similarly inform the Commission if any applicant chooses to use an operational program to meet a regulatory requirement when the requirement does not call for an operational program, and as a result, the staff adds this program to the list addressed by the license condition. Such a Commission notification should be made to the extent permitted by the separation of functions rule.

The Commission also has approved the use of the generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria (EP ITAAC) included in Attachment 2 to the paper as the minimum set of ITAAC for EP included in a COL application, recognizing that the acceptability of proposed plant-specific EP ITAAC will be reviewed on a case-by-case basis.

Regarding the standard license conditions for fire protection and security, the Commission believes that codifying these conditions is more efficient than including them in each license issued. The staff should consider including these fire protection and security issues in the next rulemaking opportunity affecting the associated regulations for each condition and provide its assessment to the Commission as part of the proposed rule package.

February 20, 2006

The Honorable Joe Barton
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Barton:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter dated January 27, 2006, in which you expressed interest in the preparations the NRC has taken for the review of license applications for new reactors. We appreciate your interest in the critical tasks that lie ahead for the agency in this area.

A stable and predictable licensing process for new reactors is a top priority for the NRC. To better facilitate such a licensing process, the NRC developed 10 CFR Part 52, which allows public participation while streamlining the licensing review process. Part 52 provides for certification of advanced reactor designs through rulemaking for later use, for Early Site Permits (ESP) to resolve siting issues early, and for combined construction and operating license (COL) authorizations.

The new reactor licensing environment is very dynamic and, since passage of the Energy Policy Act, the NRC has seen an increase in the number of prospective applicants indicating that they plan to apply for a COL. To date, the NRC has certified four advanced reactor designs in our regulations and is close to issuing final Environmental Impact Statements (EISs) in support of two ESPs. The nuclear industry has indicated that 11 COL applications are currently planned, with submittals beginning in 2007. To meet this challenge, the NRC has been anticipating the hiring of more than 350 new employees to support the COL reviews in a timely manner and has realigned the organization to provide a dedicated project management team for the new reactor licensing applications.

While the Part 52 process is fundamentally sound and efficient, the NRC is identifying areas in which more can be done, including updating the rule. For example, during the North Anna ESP review, an unexpectedly large number of public comments were received on the ESP draft EIS, requiring more time to address them than was originally planned. As a separate matter, the applicant also submitted a supplement to its application late in the process that impacts many sections of the application. When these difficulties were encountered, the NRC took prompt action to reduce the impact by shifting work priorities and increasing the level of staff involvement in the process. The NRC is incorporating the lessons learned from the North Anna ESP review into the ESP review process and expects that the same difficulties will not arise during future ESP reviews. However, also critical to the process is the quality of the license applications, responsiveness of the applicants, and standardization among the applications.

The Commission believes that an efficient, stable, and predictable licensing process that maintains safety is a goal that both the Commission and the Congress share, and I intend to see that the NRC meets this goal. Enclosed are the responses to the specific questions you raised about NRC's preparation for review of new reactor licensing. If you have any additional questions, please do not hesitate to contact me.

Sincerely,

/RA/

Nils J. Diaz

Enclosures (5):
1. Response to Questions Concerning
   Licensing Actions
2. List of 25 Most Recently Licensed Plants
3. ESBWR Design Certification Application
   Acceptance Review Checklist
4. Representative List of Federal, State,
   and Local Authorizations and
   Consultations (North Anna ESP example)
5. Letter to D.A. Christian from D. B. Matthews,
   dated February 10, 2006, North Anna Early
   Site Permit (ESP) Application Review Schedule

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Commissioners:

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

DOCKETED 03/03/2006

SERVED 03/03/2006

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT YANKEE LLC	)	Docket No. 50-271-OLA
and	)	
ENTERGY NUCLEAR OPERATIONS, INC	)	
	)	
(Vermont Yankee Nuclear Power Station)	)	
	)	

CLI-06-08

MEMORANDUM AND ORDER

By this order, we deny a request by the New England Coalition ("NEC") – submitted in the form of a letter – that we prevent or stay issuance of an operating license amendment to Entergy Nuclear Vermont Yankee, LLC, and Entergy Nuclear Operations, Inc. (together, "Entergy"). NEC believes the license amendment should not be allowed to take effect until after completion of a pending adjudication before our Atomic Safety and Licensing Board. The amendment has in fact now issued (on March 2, 2006). It allows an increase in the maximum power at Entergy's Vermont Yankee Nuclear Power Station in Windham County, Vermont. NEC is an intervenor in the power uprate adjudication. The Licensing Board has not yet held a hearing on NEC's contentions.

NEC's request asks the Commission itself to "abstain" from issuing the license amendment until the Licensing Board finishes its adjudication. But it is the NRC Staff, not the Commission, that considers applications for license amendments. Indeed, our regulations expressly instruct the Staff not to let pending hearings delay licensing decisions: the Staff is "to issue its approval or denial of the application promptly" once it completes its own review of the application, notwithstanding the "pendency of any hearing."[1] And the Staff action on a licensing application is "effective upon issuance," except (in the case of power reactor license amendments) where there are "significant hazards considerations."[2] Here, following publishing of its proposed findings for public comment, the Staff made a "no significant hazards consideration" finding, and issued the power uprate amendment, on March 2, 2006, just two days after we received NEC's letter asking "the Commission" to abstain from issuing the license.

The NEC's argument is extremely general and it does not invoke any NRC regulation or case precedent. NEC says only that it will be denied "effective redress and due process" if the license amendment is granted now, because first there should be a full hearing on its contention that Vermont Yankee may not withstand natural phenomena, such as earthquakes, when operating under increased power.

Even if we were to give NEC's request a generous construction and treat it as a request for invocation of our discretionary supervisory authority over the NRC Staff to stay the Staff's issuance of the power uprate amendment, it would still be deficient.[3] To obtain a stay, a party must meet four familiar standards: likelihood of success on the merits; irreparable harm; absence of harm to others; and the public interest.[4] Irreparable harm is the most important of

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[1] *See* 10 C.F.R. § 2.1202(a).

[2] *Id.*

[3] *See Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), CLI-01-7, 53 NRC 113, 118 (2001).

[4] *See* 10 C.F.R. § 2.342(e) (standards for considering whether to stay presiding officer decisions). While technically not applicable to a request for a stay of NRC Staff action, the section 2.342(e) standards simply restate commonplace principles of equity universally followed when judicial (or quasi-judicial) bodies consider stays or other forms of temporary injunctive relief. *See Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-90-3,

the four standards – the *sine qua non* of obtaining a stay.[5] A party seeking a stay must show it faces imminent, irreparable harm that is both "certain and great."[6] NEC's unproved speculation does not equate to irreparable harm. "Merely raising the specter of a nuclear accident" does not demonstrate irreparable harm.[7] And, contrary to NEC's view, an NRC Staff decision to grant Vermont Yankee's power uprate license amendment does not leave NEC without "effective redress." If the Board determines after full adjudication that the license amendment should not have been granted, it may be revoked (or conditioned).

NEC appears to believe that granting the license amendment prior to a Board decision bypasses NEC's right to a hearing. But the Atomic Energy Act expressly authorizes the NRC to grant license amendments, and to make them immediately effective "in advance of the holding and completion of any required hearing," so long as the NRC determines that the amendment involves "no significant hazards consideration:"

> The Commission may issue and make immediately effective any amendment to an operation license ... upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person. Such amendment may be issued and made immediately effective in advance of the holding and completion of any required hearing. [8]

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31 NRC 219, 257 (1990).

[5] *See USA Recycling, Inc. v. Town of Babylon*, 66 F.3d 1272, 1295 (2d Cir. 1995). *Accord U.S. Department of Energy* (High-Level Waste Repository), CLI-05-27, 62 NRC 715, 718 (2005).

[6] *See, e.g., Cuomo v. NRC*, 772 F.2d 972, 976 (D.C. Cir. 1985), *quoting Wisconsin Gas Co. v. FERC*, 758 F.2d 669, 674 (D.C.Cir. 1985).

[7] *Massachusetts Coalition of Citizens with Disabilities v. Civil Defense Agency*, 649 F.2d 71, 75 (1st Cir. 1981). *Accord Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-90-3, 31 NRC 219, 259 (1990); *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-84-5,19 NRC 953, 964 (1984).

[8] *See* Atomic Energy Act, §189a(2)(A), 42 U.S.C. § 2239a(2)(A). *See also* 10 C.F.R. § 2.1202(a); 10 C.F.R. § 50.58(b)(6); 10 C.F.R. § 50.92.

The other factors governing the grant or denial of stays also do not favor NEC's request. A party seeking a stay must show that it is likely to prevail on the merits of the dispute. NEC has not even addressed the substance of its merits claims in the adjudication, let alone shown it is likely to prevail. The final two factors are whether the relief would harm the other parties and where the public interest lies. NEC does not address these factors either. On the face of things, though, it would appear that delaying the license amendment, as NEC requests, would harm Entergy without any obvious benefit to the public interest.

NEC's request is denied.[9]

IT IS SO ORDERED.

For the Commission[10]

/RA/

ANNETTE L. VIETTI-COOK
Secretary of the Commission

Dated at Rockville, MD
This 3rd day of March, 2006

Concurring opinion by Commissioner Gregory B. Jaczko:

My approval of today's decision should not be construed as agreement with the determination that this license amendment should be immediately effective. My concerns regarding this license amendment being immediately effective are being addressed in another forum.

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[9] Nothing in today's decision should be understood as expressing our views on the validity of the amendment at issue here, as we may have to review it in our adjudicatory capacity after completion of Licensing Board proceedings.

[10] Chairman Diaz was not present when this item was affirmed. Accordingly the formal vote of the Commission was 4-0 in favor of the decision. Chairman Diaz, however, had previously voted to approve this Order and had he been present he would have affirmed his prior vote.

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

February 13, 2006

NRC INFORMATION NOTICE 2006-04: DESIGN DEFICIENCY IN PRESSURIZER HEATERS FOR PRESSURIZED-WATER REACTORS

ADDRESSEES

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

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees about pressurizer heaters that failed following replacement because the heater elements provided by the vendor did not match the licensees' design specification. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

During the Fall 2004 refueling outage, the licensee at Palo Verde Generating Station, Unit 3, replaced all 36 pressurizer heaters with replacements supplied by Framatome that had heater internals manufactured by Thermocoax. From December 2004 through February 2005, four of the replaced heaters in the proportional heater banks failed. On May 23, 2005, with Palo Verde Unit 3 in Mode 5 (cold shutdown), the licensee replaced nine Framatome/Thermocoax heaters with General Electric (GE) heaters. During the subsequent reactor heatup, five Framatome/Thermocoax heaters in the backup heater banks failed. As a result of the continued heater failures, the licensee returned to Mode 5 to replace all remaining Framatome/Thermocoax pressurizer heaters with GE heaters. When the Framatome/Thermocoax heaters failed, all were grounded, and all but one tripped a circuit breaker to clear a ground fault. The licensee discovered one heater grounded while maintenance was being performed during the outage. There was no damage to any other equipment such as power cables as a result of the heater failures.

During the Spring 2005 refueling outage, Waterford Steam Electric Station, Unit 3, replaced 29 pressurizer heaters with replacements supplied by Framatome that had the heater internals manufactured by Thermocoax. During plant heatup but prior to reactor startup, two of the replaced heaters experienced partial ejection of the epoxy in the receptacle area due to heat

ML060100450

transfer to electrical connections in the receptacle area, six experienced failure due to grounding, and several experienced partial melting of the silicon-type material used to seal the bottom end of the receptacles. The licensee replaced 23 Framatome/Thermocoax heaters with Watlow heaters and abandoned the remaining 6 Framatome/Thermocoax heaters in place by electrically disconnecting them. There was no damage to any other equipment such as power cables as a result of the heater failures.

The vendor subsequently inspected the failed heaters from the Palo Verde and Waterford plants and determined that the heaters had been incorrectly fabricated with a longer heating element than the licensees' design specification. The longer heating elements extended down into the heater sleeves and pressurizer shell thereby changing the location of the transition joint that separates the heated and unheated portion of the heater assembly. This resulted in a reduced ability to transfer that heat away from the heater and also allowed more heat transfer to electrical connections in the receptacle area.

DISCUSSION

Technical specifications for PWRs specify a minimum required available capacity of pressurizer heaters to ensure that the RCS pressure can be controlled to maintain subcooled conditions in the RCS. Plant operation with failed pressurizer heaters can affect a facility's ability to control reactor pressure. Following a reactor trip, unnecessary safety injection actuations could occur due to inability to maintain RCS system pressure above the actuation set point.

Additionally, the longer heating elements extended down into the heater sleeves and pressurizer shell resulted the potential to exceed the allowable temperature limits by the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code.*

The Palo Verde and Waterford licensees had each supplied Framatome the correct design specification regarding the location of the transition joint between the heated and unheated portions of the heater assembly. However, Framatome supplied pressurizer heater assemblies that did not match the design specification. The licensees did not obtain vendor specifications and drawings that were sufficiently detailed to allow them to identify that the replacement pressurizer heaters were not consistent with the licensees' design specification.

At Palo Verde Generating Station, Unit 3, one heater was discovered grounded while maintenance was being performed during the outage. Sensitive ground-fault protection on low voltage circuits such as 480 V pressurizer heater circuits, can help in the detection of a ground fault.

Additional information on this subject is available in a Title 10 of the *Code of Federal Regulations* Part 21 (10 CFR Part 21) report from Framatome dated July 28, 2005, which is accessible using NRC's document control system (Agencywide Documents Access and Management System (ADAMS), Accession No. ML052140277).

CONTACTS

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

/RA/
Christopher I. Grimes, Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Technical Contacts:

David N. Graves, R-IV/DRP/RPB-E
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NRR Project Manager:
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E-mail: OTY@nrc.gov

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

NRC Yellow Announcement

UNITED STATES
NUCLEAR REGULATORY COMMISSION

Announcement No. 012

Date: February 14, 2006

To: **All NRC Employees**

SUBJECT: MANAGERIAL ASSIGNMENTS IN THE OFFICE OF NUCLEAR REGULATORY RESEARCH

The Office of Nuclear Regulatory Research is reorganizing in order to better align the organization with evolving program and mission needs, consolidate operating experience and new/advanced reactor functions, and enhance organizational effectiveness. I am pleased to announce the following associated managerial assignments.

James T. Wiggins remains the Deputy Office Director. Mabel F. Lee will remain the Director of Program Management, Policy Development and Analysis Staff.

As part of the reorganization, two Division Directors will report to the Office Director as follows: Mark A. Cunningham, Director of the Division of Fuel, Engineering and Radiological Research, and Farouk Eltawila, Director of the Division of Risk Assessment and Special Projects.

Reporting to Mr. Cunningham will be Sher Bahadur, who will become Assistant Director of the Division of Fuel, Engineering and Radiological Research; Jennifer Uhle, who will become Deputy Director for Materials Engineering; Michele G. Evans, who will become Deputy Director for Engineering Research Applications; and Nilesh C. Chokshi, who will become Deputy Director for Radiation Protection, Environmental Risk and Waste Management.

Reporting to Mr. Eltawila will be Charles E. Ader, who will become an Assistant Director of the Division of Risk Assessment and Special Projects; Richard J. Barrett, who will become an Assistant Director of the Division of Risk Assessment and Special Projects; Christiana Lui, who will become Deputy Director for New Reactors and Computational Analysis; John D. Monninger, who will become Deputy Director for Probabilistic Risk and Applications; and Patrick W. Baranowsky, who will become Deputy Director for Operating Experience and Risk Analysis.

The new organization is effective on February 19, 2006. The reorganization chart can be accessed at http://www.internal.nrc.gov/RES/RES_OrgChart_2.19.06.pdf.

/RA/

James T. Wiggins (for)
Carl J. Paperiello, Director
Office of Nuclear Regulatory Research

NRC Yellow Announcements Index

Inside NRC

Volume 28 / Number 5 /March 6, 2006

Exelon tritium leaks come under state and federal scrutiny

A series of tritium leaks at Exelon Nuclear plants in Illinois has prompted actions by state environmental officials, interest groups and members of Congress.
The revelation last fall of leaks of tritiated water from the Braidwood plant resulted in Exelon receiving a notice of violation from the Illinois Environmental Protection Agency in December (INRC, 6 Feb., 9). Another notice was issued last month by IEPA for leaks discovered at Dresden. IEPA and Exelon are engaged in discussions of appropriate means to address the violations.

"The Illinois Department of Public Health has been provided with and has reviewed analytical results from private well tests near the Braidwood plant," and "have not seen tritium levels in the well tests to date that pose a health hazard," IEPA said in a February 7 fact sheet. "Illinois EPA will continue to work with IDPH to evaluate any potential health impacts and keep area residents informed," IEPA said.
On March 1, legislation titled the "Nuclear Release Notice Act of 2006" was introduced in the Senate (S. 2348) by Senators Barack Obama and Richard Durbin and in the House (H.B. 4825) by Representative Jerry Weller, all Democratic members of the Illinois congressional delegation.
"While most of the issues associated with this situation [Braidwood tritium leaks] are still under investigation, one issue is clear. Community residents, particularly the state and local officials responsible for the safety and health of their constituents, did not receive full or immediate notification of this contamination — either from Exelon, or the Nuclear Regulatory Commission, the federal agency with oversight over nuclear plant operations," Obama said in his statement introducing the legislation.

The bill would require, as a condition of every NRC license for a "utilization facility" such as a power reactor, that the licensee "immediately notify" the NRC "and the state and county in which the facility is located" of any "unplanned release" of radioactive substances. These are defined in the bill as "any unplanned release of quantities of fission products or other radioactive substances (i) in excess of allowable limits for normal operation established by the Commission or other applicable federal laws or standards; and (ii) within allowable limits for normal operation ... but that occurs more than twice within a two-year period originating from the same source, process, or equipment at a facility."

"When radioactive substances are released into the environment outside of normal operating procedures, notifying state and local officials should not be a courtesy; it should be the law," Obama said.

Tommy Vietor, Obama's press secretary, said March 2 that the appearance of NRC commissioners at a March 9 NRC oversight hearing by a Senate Environment and Public Works subcommittee will provide Obama and other committee members "a chance to discuss" these issues.

Weller, who represents the 11th congressional district in which Braidwood and Dresden are located, wrote to NRC Chairman Nils Diaz in a February 15 letter requesting that the NRC "strongly enforce its regulations on Exelon Corporation and make sure the public is fully aware of the actions being taken by both the agency and the company." Weller also requested "a detailed summary and complete time-line of events involving the tritium leak at Braidwood ... and also the recent on-site leak at the Dresden facility," including "a detailed summary of how the tritium leak[s] are going to be resolved."

Weller asked that the commission "do an independent audit of all the nuclear power facilities in Illinois, with a strong emphasis on system components dealing with tritium. With instances now occurring at more than one facility I believe an entire systems-wide audit is necessary."
Exelon is "reviewing the particulars of the bill," Craig Nesbit, director of communications at Exelon Nuclear, said in a March 2 e-mail. "We welcome any dialogue that will bring clarity to the notification issue and look forward to working with Senators Obama and Durbin and Rep. Weller to address our shared concerns. Exelon is committed to full notification to all affected people and all levels of government, and we are further committed to going beyond the formal requirements of the law," Nesbit said.
Eliot Brenner, director of public affairs at NRC, declined

March 2 to comment on the bill because the agency had not yet had an opportunity to review it.

Leaks at Braidwood, Dresden and Byron

Last month, Exelon confirmed that tritium leaks had also occurred at its Dresden and Byron plants. As a result, Exelon is "launching an initiative across its 10-station nuclear fleet to systematically assess systems that handle tritium and take the necessary actions to minimize the risk of inadvertent discharge of tritium to the environment," Exelon said in a February 15 statement.

The "small leak" at Dresden "dripped at a rate of about a half-cup per minute and was discovered within a few weeks after it began," Exelon said. "The leak was confined to shallow ground in a small area near the center of the plant property," is "not expected to approach the edge of the plant property, and poses no health or safety threat," Exelon said. Tritium found in one test well at Dresden measured 500,000 picocuries per liter, 25 times the limit imposed by the federal Environmental Protection Agency for drinking water. However, "surrounding test wells 10 to 20 feet away showed tritium concentrations of 20,000 picocuries per liter [the EPA standard] or less, indicating a small area of tritium that dissipates rapidly at the edges," and "testing along the site boundary confirmed that no tritium has approached the property edge," Exelon said in its statement.

Recent inspections at Byron found tritium concentrations of 86,000 picocuries per liter in standing water located in concrete vaults along the plant's "blowdown line," a pipe which carries water discharged from the plant to the Rock River. These vaults "house valves known as 'vacuum breakers' that can malfunction and leak," Exelon said. "The Byron tritium concentrations pose no health or safety threat to employees or the public," the company said. Exelon has posted information on the tritium leaks on the Braidwood Web site at http://www.braidwoodtritium.info.

Arjun Makhijani, president of the Institute for Energy and Environmental Research, an energy policy interest group in Takoma Park, Maryland, challenged these assessments in a February 6 statement on the Braidwood leaks. "The current drinking water standard for tritium of 20,000 picocuries per liter does not take non-cancer effects of tritium, such as miscarriages, into account," Makhijani said. He noted that "the surface water standard for tritium in the state of Colorado is 500 picocuries per liter, which is 40 times more stringent than the EPA drinking water standard."

"I am of the opinion that the NRC has not been vigilant enough in trying [to] make reactor operators reduce their tritium discharges," Makhijani said.

Petition under review

In January, the Union of Concerned Scientists and 21 other interest groups filed a petition with NRC requesting that the agency issue a demand for information from power reactor licensees on "the potential for undetected, longstanding leaks of radioactively contaminated water into the ground." The NRC will take action on the petition "within a reasonable time" as provided in regulations, Christopher Grimes, director of NRC's division of policy and rulemaking, said in a March 1 letter to David Lochbaum of UCS. The petition is "being reviewed by organizations in several NRC offices to ensure [the petitioners'] concerns are being considered in ongoing NRC activities related to the issue of the release of contaminated water at NRC-licensed facilities," Grimes said.

"Many of our activities in this area may not have been public knowledge" before the petition was submitted, Grimes said in his letter. "Actions underway or being considered include the conduct of special inspections, revisions to NRC regulations and related guidance documents, revisions to NRC inspection procedures, issuance of one or more generic communications, and the scheduling of public meetings with licensees, industry groups, and other stakeholders," Grimes said.

NRC will hold a public meeting with the Nuclear Energy Institute on March 22 at the agency's headquarters in Rockville, Maryland "to discuss the occurrence of abnormal tritium discharges from nuclear power plants," the agency said in a February 28 meeting notice. NRC staff "would also like to have a technical review meeting within the next several weeks" with the petitioners "to discuss NRC activities and the requests in [the] petition," Grimes said in his letter. On March 2, the NRC released its annual assessment letters and inspection plans for Braidwood, Dresden and Byron. In each case, the NRC concluded that the plants operated in 2005 "in a manner that preserved public health and safety and fully met all cornerstone objectives" of the agency's reactor oversight process. The tritium leaks were not mentioned in the letters.— ***Steven Dolley, Washington***

NEI calls for commission review on direction of risk-informed regs

The Nuclear Energy Institute has asked for a meeting with the NRC commissioners to discuss the industry's disappointment with the direction risk-informed regulation is taking.

In a February 28 letter, Marvin Fertel, NEI's senior vice president and chief nuclear officer, said recent developments in two areas — NRC guidance on implementing 10 Code of Federal Regulations Part 50.69, which risk-informs NRC's special treatment requirements; and the proposed 10 CFR 50.46a, which would provide risk-informed changes to the loss-of-coolant accident technical requirements — "focus more attention on matters of low safety significance and managing residual risk through programmatic requirements and expectations."

Fertel said the industry surmises that NRC's approach is intended to ensure that any matters of low safety significance, individually or collectively, could never become safety significant. "We believe this approach is fundamentally wrong from a safety perspective," Fertel said.
He said the staff's recently issued regulatory guide 1.201, which is to accompany 10 CFR 50.69, a rule adopted by the agency in November 2004, contains some NRC expectations for the treatment of safety-related, but low safety significant components that "exceed current requirements."

Regarding the proposed 50.46a, Fertel said that the staff's proposal "bears little resemblance to the original concept of using risk insights and operational experience to refine the original deterministic loss-of-coolant pipe break size." Fertel went on to say that the industry believes the rule has "become encumbered with excessive and burdensome requirements for change control and operational restrictions unnecessarily layered atop current regulatory requirements." (INRC, 6 Feb., 3). NEI is expected to submit lengthy comments on 50.46a to NRC this week.

Fertel said that NEI is also concerned that other NRC initiatives could also be affected by the current staff approach to risk-informed regulation, including risk-informed fire protection and the licensing of new plants.

"We believe it is time for a constructive and open discussion on the expectations, limitations, and real issues that are at the heart of the development and implementation of riskinformed performance-based regulatory approaches," Fertel said. "If the concept has changed from what was originally intended by the commission, the industry needs to know so that we may adjust our efforts accordingly. This is a policy issue that warrants the commission's urgent attention," he said.—***Michael Knapik, Washington***

World nuclear regulators agree to meet again in three years

After their first summit last week in Moscow, senior regulators from around the world agreed to meet again in three years' time to review progress in creating effective nuclear regulatory systems and deepening international collaboration.
The 216 participants representing 57 countries and seven international organizations, meeting under the auspices of the IAEA February 28-March 3, approved the conclusions of the International Conference on Effective Nuclear Regulatory Systems, drafted by conference president Laurence Williams, containing the recommendation for a further summit. As IAEA Director General Mohamed ElBaradei had said in a speech prepared for the conference, that up to now world regulators have either met briefly at the IAEA General Conference, or attended meetings devoted to topics other than regulation. "Regulators need a meeting of their own," ElBaradei concluded. The idea of regular meetings, possibly on a triennial basis, provides that forum in recognition that despite having listened and talked for three days, the regulators had only scratched the surface of their common interests in Moscow.

The Moscow conference was conceived jointly by NRC Chairman Nils Diaz and former Russian regulatory chief Andrey Malyshev in 2004, and the idea was embraced by ElBaradei. It was hosted by the Russian Federation in the framework of Russia's G8 presidency this year.

K. Pulikovsky, chairman of Rostekhnadzor, the technical regulatory agency of which Malyshev's nuclear regulation division is now a part, told a press conference that the G8 summit in St. Petersburg in June would underline that "energy development is not possible without nuclear energy, and nuclear energy is not possible without safety."

Indeed, Williams said the conference had concluded that "the delivery of effective nuclear safety and security regulation is vital for the safe and secure use of nuclear energy and associated technologies...and is an essential prerequisite for the achievement of global energy security and global sustainable development....Regulators work for society and therefore they play a vital role."

During the three days, regulators hotly debated what is meant by "independence" of regulatory organizations and how to ensure that independence. According to Williams' conclusions, it meant freedom to make regulatory decisions "solely in relation to the need to maintain safety and security, without pressure" from promoters of the technology or, conversely, from its opponents. The discussions during the conference revealed that some regulators do see at least an implicit link between their job and making nuclear energy safe for society, a link that was criticized by German nuclear regulatory chief Wolfgang Renneberg as bordering on promotion (Nucleonics Week, 2 March, 10).

The conference spent considerable time debating the value of international safety standards and whether and how they can be achieved, including a presentation by NRC Chairman Nils Diaz of his initiative for a Multinational Design Approval Program (MDAP) (see story, p. 9). It touched on a wide variety of topics of interest to regulators from countries without nuclear power, notably the Code of Conduct on management of sealed sources and the associated Export-Import Guidance approved by the IAEA general conference in 2004.

Perhaps the most contentious issue was that of the relationship between safety and security (physical protection and prevention of terrorist acts), with some regulators saying they could both be governed by the same institutions and others reporting interface and information exchange problems. The debate centers essentially on whether safety is part of security, the other way around, or whether they can exist independently. The discussion on this subject was only the most recent of a growing series of debates, including during last July's IAEA conference on sources in Bordeaux, that sources said reflects above all the grappling of the IAEA leadership with the problem of coordinating safety and security programs within the agency itself.

Williams' conclusions skirted the controversy, saying only that ensuring both safety and security of nuclear facilities, materials and radioactive materials "requires effective coordination of safety and security regulation."
—Ann MacLachlan, Moscow

Exelon
Nuclear

ACRS Presentation
March 9, 2006

Early Site Permit Application
Clinton Power Station Site
Final Safety Evaluation Report

2

Exelon. Nuclear

Agenda

- Introductions
- Significant Changes Since Draft Safety Evaluation Report (DSER)
- Geotechnical Approach
- Seismic Evaluation
- Supplemental DSER Issue Closure
- Summary

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Introductions - ESP Project Team

- Marilyn Kray – Project Executive Sponsor
- Christopher Kerr – Sr. Project Manager
- Eddie Grant – Safety / EP Lead
- William Maher – Environmental Lead

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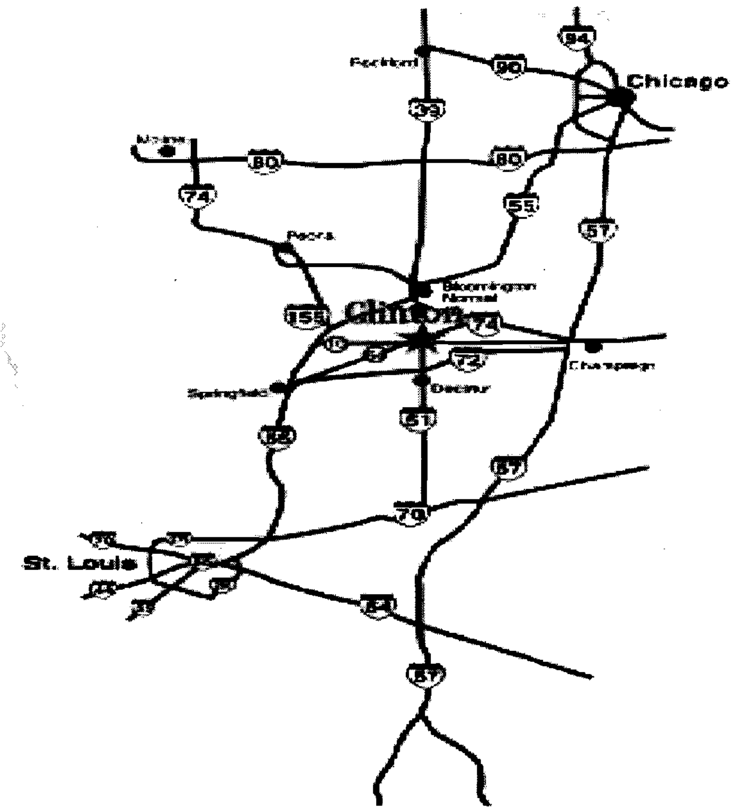
Introductions – Support Team

- CH2M Hill (Prime Contractor)
  - Environmental / Redress
  - Geotechnical
  - EP
- CH2M Hill Subcontractors
  - WorleyParsons
    - o Safety
  - Geomatrix
    - o Seismic
  - Seismic Board of Review
    - o Expert, independent review
  - Others
- RPK Structural Mechanics Consulting
  - Seismic
- Sargent and Lundy
  - Draft Application Review
- Morgan Lewis
  - Legal counsel

Introductions – Site Location

- ESP Site Location
  - Central Illinois
  - Clinton Power Station Property
  - AmerGen Owned (EGC Subsidiary)
- Applicant
  - Exelon Generation Company, LLC (EGC)
    - o Wholly owned subsidiary of Exelon Corporation



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Significant Changes Since DSER

- Closure of all Open Items
- Completion of all Confirmatory Items
- Acceptance of SSE ground motion spectra
  - Minor revisions in response to open items
- Documented Criteria for:
  - Permit Conditions
  - Combined License Action Items

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Geotechnical Approach

- Builds on existing CPS information
  - Regional geology
  - Site geology
  - Exploration
  - Laboratory testing
- EGC ESP work
  - Confirm conditions
  - Updated information

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Seismic Evaluation
SSE Ground Motion Determination

RG 1.165 Methodology	*EGC ESP Application*
➢ Investigations	➢ Same
➢ Seismic sources update	➢ Same
➢ SSHAC assessment	➢ Same
➢ PSHA	➢ Same
➢ Determine SSE ground motion spectra • **Relative based -- Reference Hazard Probability Criterion**	➢ Determine SSE ground motion spectra • **Performance-based – Core Damage Frequency Criterion**

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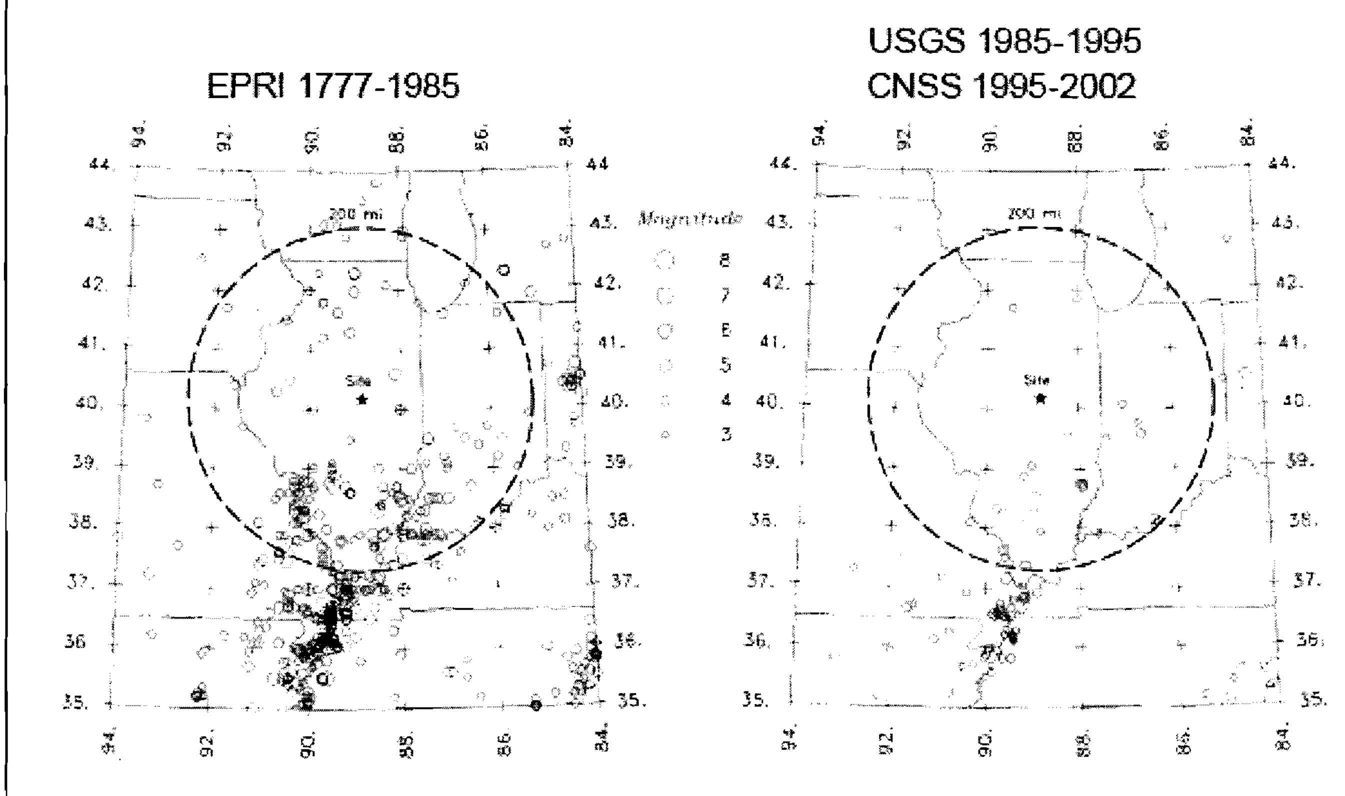
Seismic Evaluation (cont'd)

SSE Ground Motion Determination (cont'd)

RG 1.165 Methodology	*EGC ESP Application*
➢ De-aggregate to identify controlling earthquakes	➢ Same
➢ Account for site effects	➢ Same [NUREG/CR-6728]

Seismic Evaluation (cont'd)

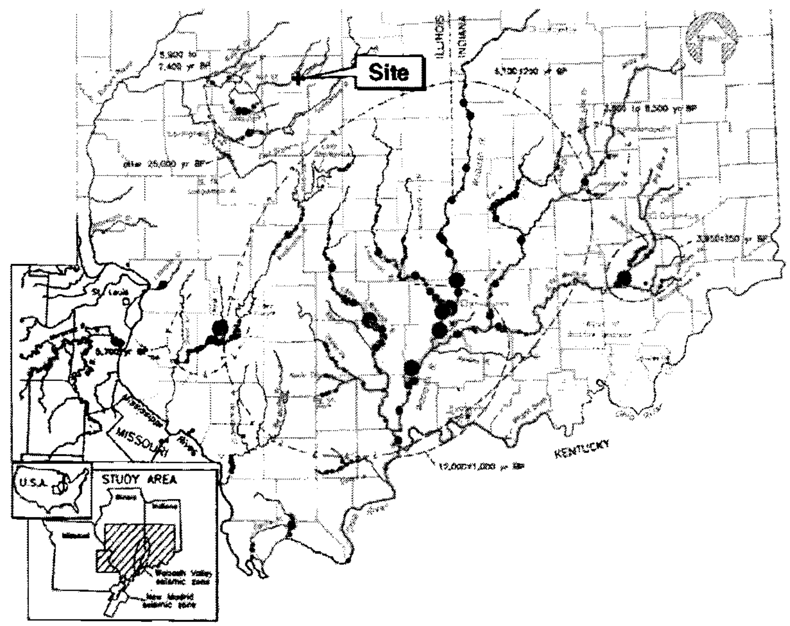


Seismic Hazards Report for the EGC ESP Site
Updates to Seismicity Catalog

Figure
2.1-11

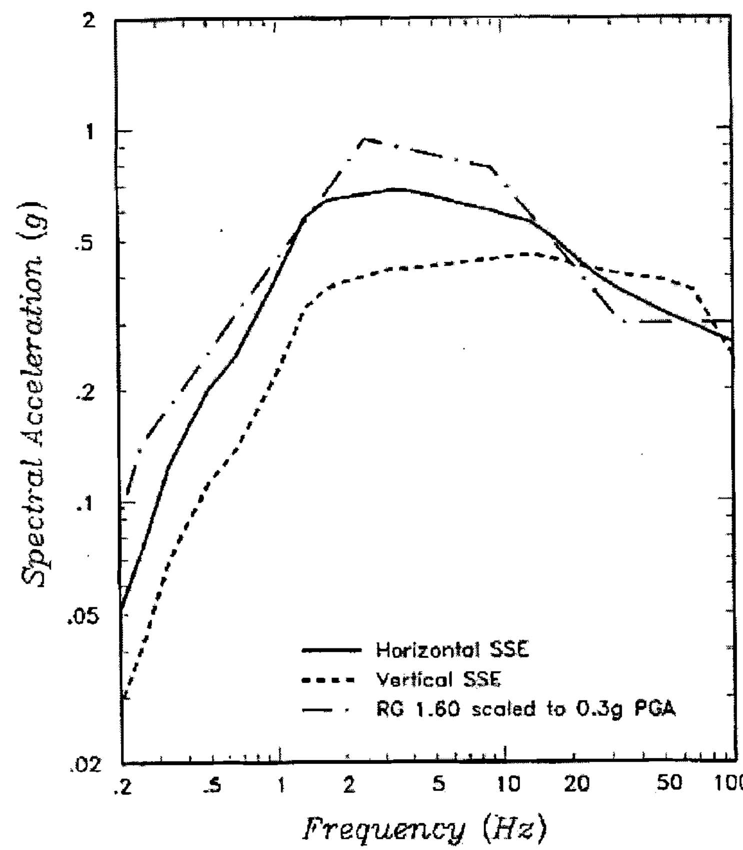
Seismic Evaluation (cont'd)

- Major new information
  - Repeated large events in New Madrid seismic zone in past 2,000 years
  - Large events in Wabash Valley/ Southern Illinois in past 12,000 years
  - One moderate event with energy center ~40 miles SW of site at Springfield ~6,000 years ago



Seismic Evaluation (cont'd)

- Performance-Based EGC ESP SSE Ground Motion Spectra
  - Horizontal DRS
  - Vertical DRS
  - RG 1.60 0.3g PGA (for reference only)
  - Acceptable to NRC Staff
  - Compared to Design Spectra at COL stage



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Supp. DSER Issue Closure

- Open Items (7) - Resolved
  - 2.5.1-1, New Madrid magnitude estimates
  - 2.5.2-1, Distance-conversion in EPRI '03 Ground Motion Model
  - 2.5.2-2, Site velocity model for response analysis
  - 2.5.2-3, Site dynamic response analysis
  - 2.5.2-4, SSE ground motion adequately represents local prehistoric earthquakes
  - 2.5.2-5, Performance-based method clarification
  - 2.5.4-1, Additional borings

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Summary

- All Open Items Closed
- All Confirmatory Items Completed
- SSE Ground Motion Spectra Accepted

(2)

Exelon Early Site Permit
Safety Review Status

UNITED STATES NUCLEAR REGULATORY COMMISSION

March 9, 2006

Advisory Committee on Reactor Safeguards
Full Committee Meeting

John Segala, Senior Project Manager
Office of Nuclear Reactor Regulation

Purpose

- To provide the ACRS an overview of the Exelon early site permit (ESP) application safety review
- Answer the ACRS's questions

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Meeting Agenda

- Project Milestones
- Exelon ESP Safety Review
- Key Review Areas
- Open Items
- Permit Conditions/COL Action Items
- FSER Conclusions
- Seismic Review
- Questions or Comments

Completed Milestones

- Received Exelon ESP application - September 25, 2003
- FRN published announcing acceptance – October 31, 2003
- FRN published for mandatory hearing – December 12, 2003
- RAIs issued to the Applicant – July, 27, 2004
- Draft SER issued – February 10, 2005
- Applicant responds to Draft SER open items – April 26, 2005
- Supplemental Draft SER issued – August 26, 2005
- ACRS Full Committee Meeting - September 8, 2005
- ACRS interim letter – September 22, 2005
- Staff provided Final SER to ACRS – February 9, 2006
- **Staff issued Final SER – February 17, 2006**
- ACRS Subcommittee Meeting – March 8, 2006

Remaining Milestones

- ACRS letter assumed – March 30, 2006
- Staff issue Final SER as NUREG – May 1, 2006
- Mandatory hearings begin Fall 2006
- Commission decision assumed mid 2007

Exelon ESP Safety Review

- Final SER documents the staff's technical review of the applicant's site safety analysis report and emergency planning information
- Exelon requests ESP site be approved for total core thermal power rating between 2400 and 6800 MWt
- Exelon has chosen not to submit specific design but instead has submitted a plant parameter envelope (PPE) based on a number of current and future reactor designs

Key Review Areas

- Exclusion Area Authority and Control
- Nearby Industrial, Transportation, and Military Facilities
- Meteorology
- Hydrology
- Seismology and Geology
- Radiological Effluents
- Thermal Discharges
- Radiological Consequences of Accidents
- Physical Security
- Aircraft Hazards
- Emergency Planning
- Quality Assurance

Principal Contributors

Brad Harvey - Meteorology

Goutam Bagchi – Hydrology

- Contract support from PNNL

Kazimieras Campe - Site Hazards

- Contract support from PNNL

Clifford Munson and Tom Cheng – Geology, Seismology, and Geotechnical

- Support from U.S. Geologic Survey and BNL

Jay Lee – Demography, Geography, and Radiological Consequence Analysis

Robert Moody - Emergency Planning

- Consultation with FEMA

Paul Prescott - Quality Assurance

Al Tardiff - Physical Security

Open Items

Review Area	Open Items
Exclusion Area Authority and Control	1
Meteorology	3
Hydrology	21
Seismology and Geology	7
Radiological Consequences of Accidents	1
Emergency Planning	6
Quality Assurance	1
Total:	40

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Proposed Permit Conditions and COL Action Items

- There are 6 proposed Permit Conditions (15 in the Draft SER)
- There are 32 proposed COL Action Items (17 in the Draft SER)

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FSER Conclusions

Overall:

- Site safety and emergency planning is acceptable and meets the regulations

Seismology and Geology:

- Site is acceptable from a geologic and seismologic standpoint and meets the requirements of 10 CFR 100.23

NRC Experience with the Performance-Based Methodology

- Use of a performance-based approach for determining the SSE first identified in Exelon's application in September 2003
- NRC formed a Seismic Technical Advisory Group
  - Seismic & Civil Engineers from NRR, NMSS, and RES
  - Served in an advisory role to NRR for review of performance-based approach

Exelon's Performance-Based (PB) Safe Shutdown Earthquake (SSE)

NRC staff concluded:

1. PB method based on sound technical approach
2. Seismic design using PB SSE achieves safety level generally higher than operating plants
3. PB SSE adequately reflects local ground motion hazard

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Conclusion 1

PB method based on sound technical approach

- PB approach is risk-based
- PB approach requires structures be designed to achieve target performance goal
- PB SSE determined by two approaches:
  - Design Factor Method (ASCE 43-05)
  - Direct Integration of Risk Equation

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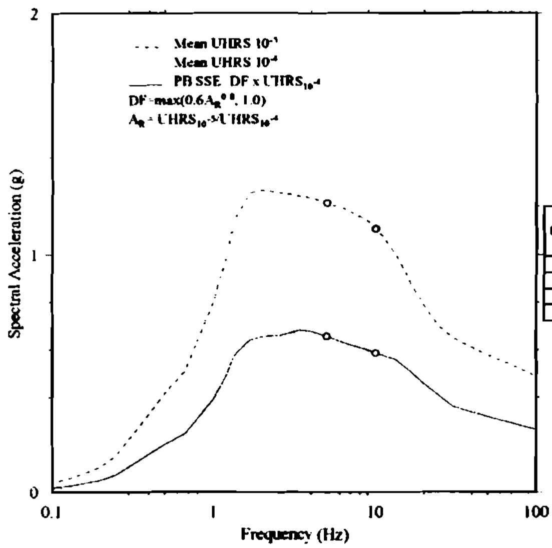
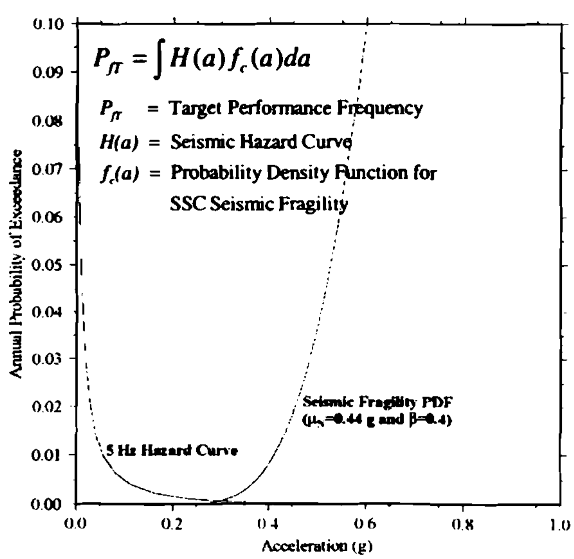
Conclusion 1 (Cont.)

Exelon Performance Based SSE



Spectral Frequency (Hz)	10-4 Mean UHRS (g)	10-5 Mean UHRS (g)	AR	DF	Horiz. SSE (g)
1	0.297	0.802	2.700	1.328	0.395
2.5	0.638	1.256	1.968	1.031	0.658
5	0.657	1.215	1.849	1.000	0.657
10	0.586	1.107	1.887	1.000	0.586

Conclusion 1 (Cont.)



Conclusion 1 (Cont.)

Parameter/Model Assumptions:

- Performance Target (P_{FT}) is $1x10^{-5}$ per year
  - P_{FT} corresponds to most stringent seismic design class
- ASCE 43-05 assumes a linear hazard curve between 10^{-4} and 10^{-5}
- SSC seismic fragility modeled using lognormal distribution
  - $\beta = 0.4$
  - Seismic Margin = 1

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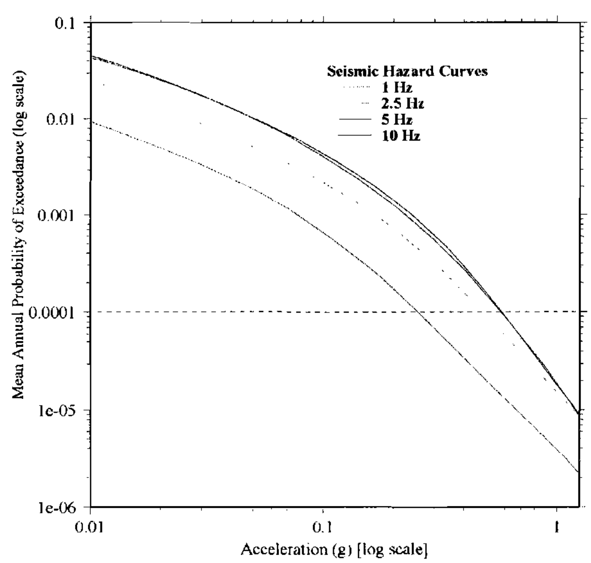
Conclusion 1 (Cont.)



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Conclusion 1 (Cont.)

Summary of Conclusion 1:

- PB Approach
  - Achieves both high and consistent level of seismic safety
  - No credit for seismic margin
  - Conservative performance target
  - Based on conservative parameter and modeling assumptions

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Conclusion 2

PB SSE achieves safety level generally higher than operating NPPs

- Using Clinton PB SSE values and HCLPF seismic margin of 1.67 (SECY 93-087)
  - What are SCDF values?
  - How do Clinton PB SCDF values compare to current NPPs?

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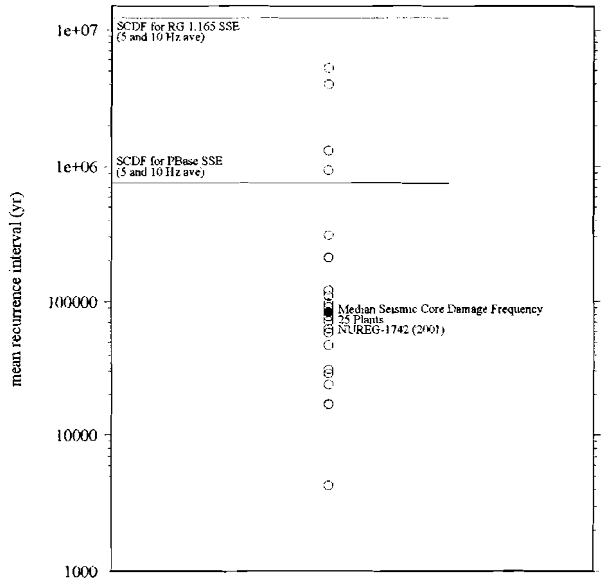
Conclusion 2 (Cont.)

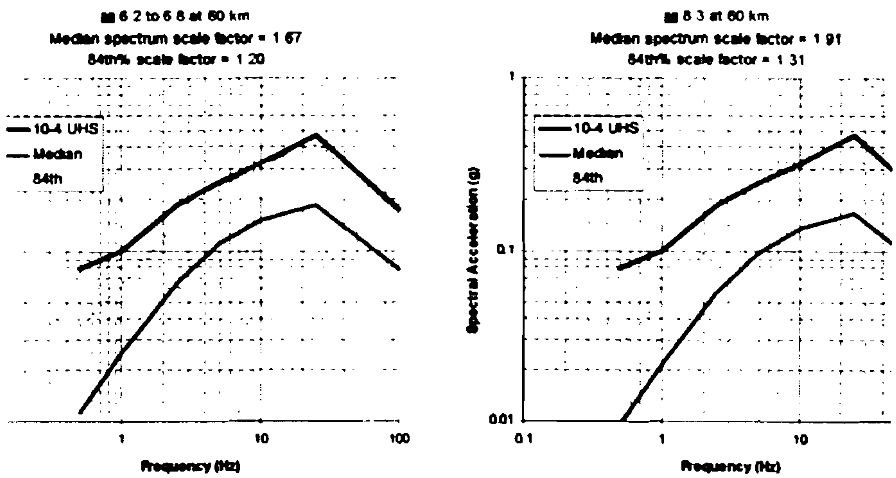


Conclusion 3

PB SSE adequately reflects local ground motion hazard

- Greatest local seismic hazard for central Illinois from Springfield earthquake
  - Prehistoric earthquake (5900 to 7400 years ago)
  - Near Springfield (60 km SW of ESP site)
  - Magnitude estimates (6.2 to 6.8)

Conclusion 3 (Cont.)



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Summary

- All open items resolved
- Looking forward to receiving the interim ACRS letter
- Questions or comments?

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Generic Safety Issue 191

Brian W. Sheron
Office of Nuclear Reactor Regulation

Advisory Committee on Reactor Safeguards
March 9, 2006

(4)

Background

- Chemical effects testing raised additional concerns about debris loading on screens. Industry initially did not aggressively pursue issue

- Many licensees approached the issue by planning significantly larger screens with excess margin to account for areas of uncertainty, in some cases literally the largest screens that the containment can accommodate

- A few licensees are pursuing an active strainer design

Status

- Staff has recently confirmed its expectation to licensees that modifications to address sump issues should be in place by end of 2007

- Both staff and industry believe that installing modified strainers at this time is correct thing to do. Downstream effects can be accommodated through engineering evaluation and component modification, as necessary

- Industry has said that a nominal amount of time (i.e., 6 months to a year) for additional analysis would not affect modified strainer installation plans, because modified strainers have already been designed, procured, and scheduled for installation

Path Forward

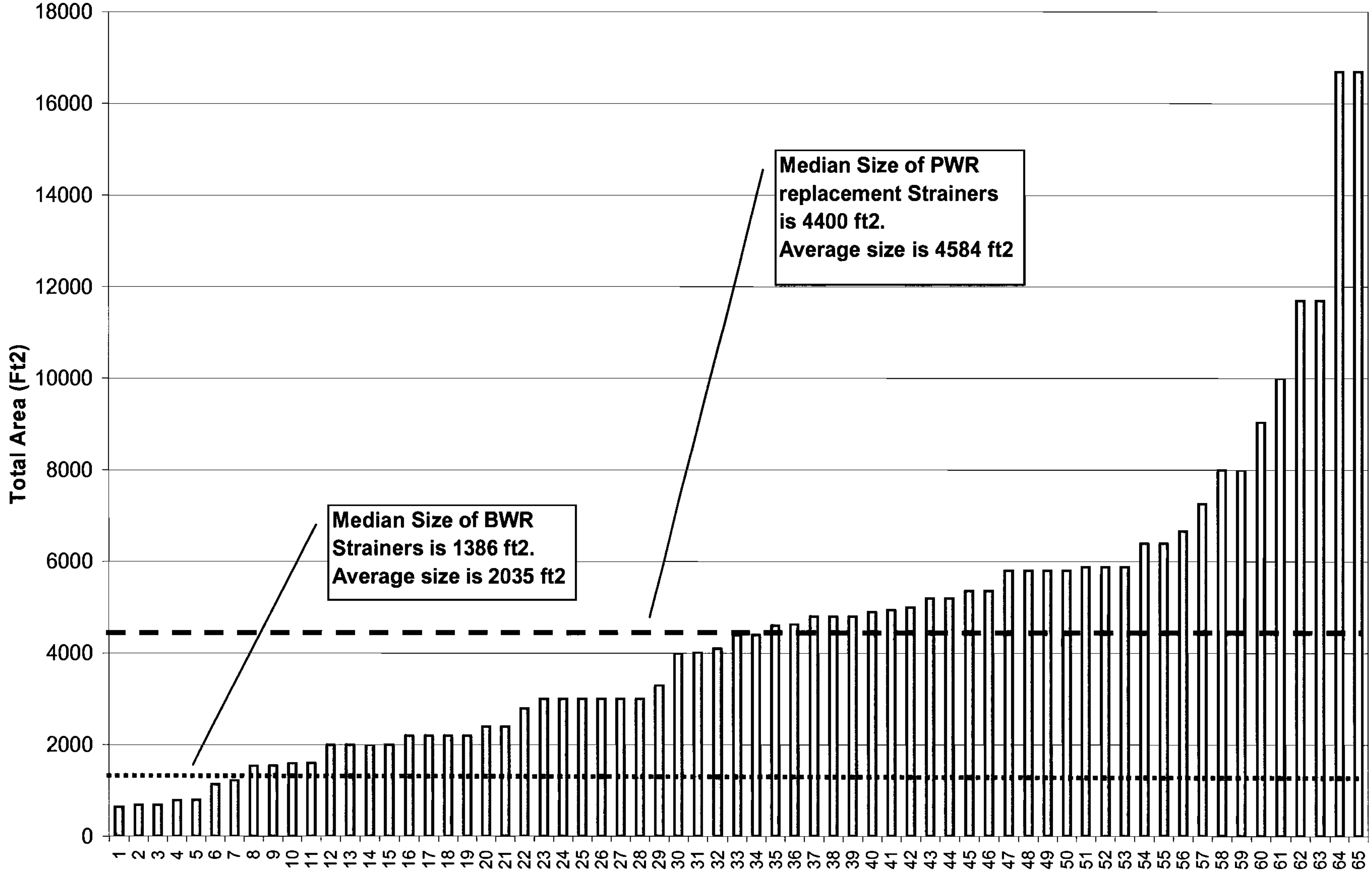
- Waiting until all testing and analysis is completed would result in unacceptable modified strainer installation dates, and would likely not significantly affect the size of the installed strainers

- Moreover, if subsequent testing and/or analyses show modified strainers still don't provide adequate margin, likely resolution would be further reduction of debris loading on strainers (e.g, fibrous insulation removal, alternate buffering agent)

- Further testing and/or analyses will be done to confirm acceptability of margins

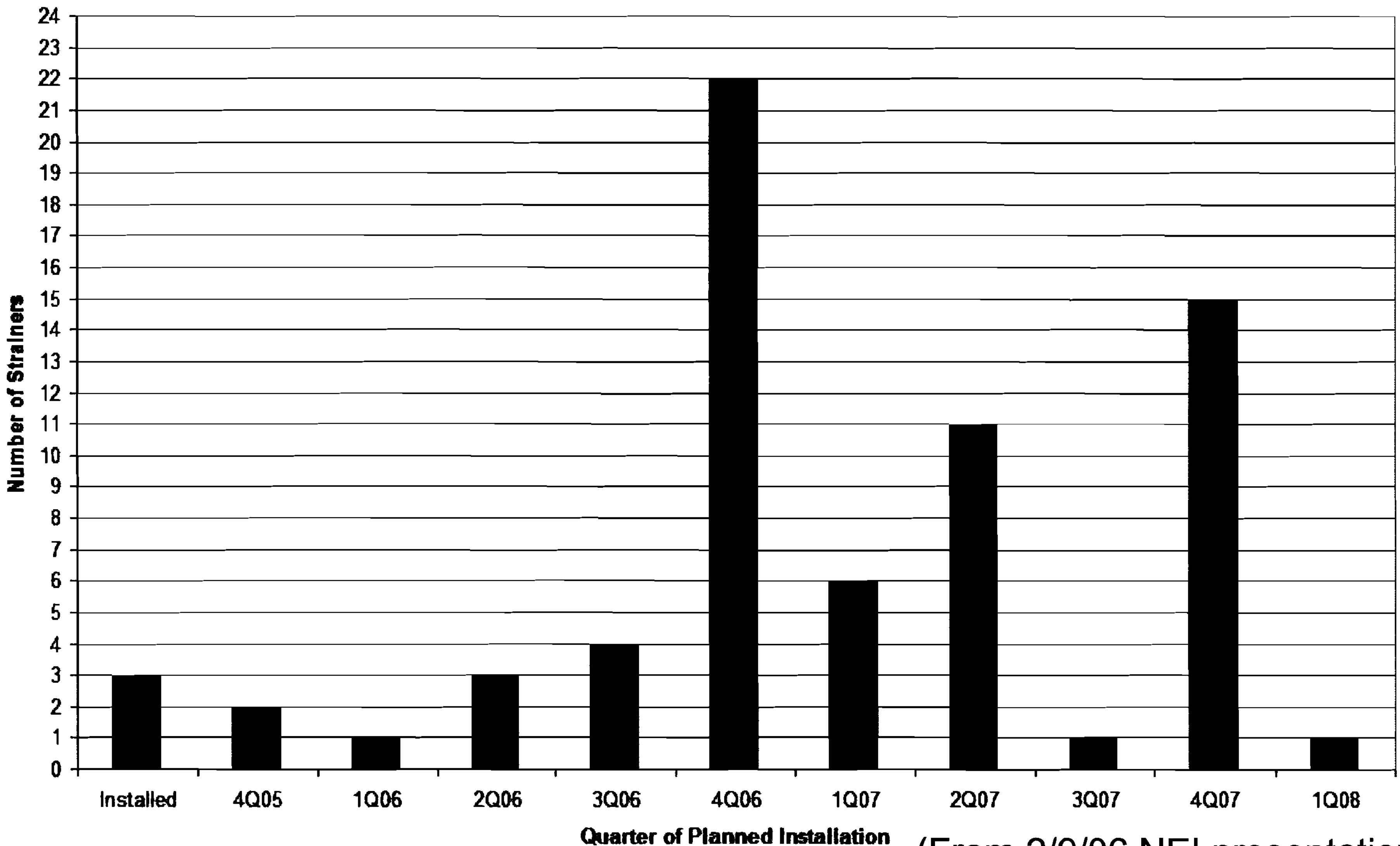
- Staff conclusion is that current schedule for modified strainer installation should be maintained and will provide significant improvement in safety compared to current strainers

Estimated Size of PWR Replacement Strainers (Passive Strainers only) *



* (From 2/9/06 NEI presentation)

Planned Strainer Installation



(From 2/9/06 NEI presentation)

Overview of Resolution Status and Plans for Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance"

Presented by:
Jon Hopkins
Thomas Hafera
Michael Scott
Office of Nuclear Reactor Regulation

Presented to: Advisory Committee on Reactor Safeguards
March 9, 2006

5

Purpose of Presentation

- Update the Committee on progress to date in addressing GSI-191, challenges and issues that remain, and plans for addressing the challenges and closing the GSI

Presentation Topics

- Background
- Chemical Effects
- Coatings Issues
- Downstream Effects
- Path Forward

GSI-191

- Objective: Ensure that post-accident debris blockage will not impede or prevent operation of PWR emergency core cooling system (ECCS) and containment spray system (CSS) in recirculation mode

UNITED STATES NUCLEAR REGULATORY COMMISSION

GSI-191 Milestones to Date

- Bulletin 2003-01 issued June 2003
- NEI methodology guidance document submitted May 2004
- Generic Letter 2004-02 issued September 2004
- NRC Safety Evaluation issued December 2004
- Licensee detailed responses to GL 2004-02, September 2005
- Information Notice 2005-26 issued September 2005
- IN 2005-26 Supplement 1 issued January 2006

UNITED STATES NUCLEAR REGULATORY COMMISSION

Generic Letter 2004-02

- Requests addressees to perform a mechanistic evaluation of the potential for the adverse effects of post-accident debris blockage and operation with debris-laden fluids to impede or prevent the recirculation functions of the ECCS and CSS
- Requests licensees to implement, by end of 2007, any plant modifications that the above evaluation identifies as being necessary to ensure system functionality
- By Sep 1, 2005, addressees were to provide:
  - Results of the evaluation
  - Modification implementation schedule
  - License amendments and/or exemption requests (if needed)

Generic Letter 2004-02 Responses

- Responses were due September 1, 2005
- All plants are upgrading or have recently upgraded their sump strainers
- While the responses were not complete, industry continues to make progress toward resolving this issue
  - Industry will provide updates to their September responses as information becomes available
  - Industry will meet with the staff periodically to keep the staff informed of the industry efforts to resolve this issue
- The staff issued requests for additional information in February 2006. Industry will respond in supplements to September 2005 GL responses
- Five units have requested additional time beyond 2007 to complete their corrective actions

Chemical Effects

- Corrosion products, gelatinous material, or other chemical reaction products that result from interaction between containment materials and the containment environment after a loss-of-coolant accident

- May affect head loss across sump strainers and downstream components

Chemical Effects Approach

Industry

- ICET (LANL)
- Bench Top Testing (WOG)
- Plant Specific Testing/Analysis

↓

Plant Specific Chemical Effects Evaluation → NRC Review

NRC

- ICET (LANL)
- Bench Top Testing (Various)
- Head Loss (ANL)
- Speciation Modeling (SwRI)

↓

NRC Review

Path Forward - Chemical Effects Evaluations

- Staff will receive and comment on Westinghouse Owners Group (WOG) report that proposes guidance for industry chemical effects evaluations
- Staff will continue interactions with screen vendors to resolve technical issues with plant-specific testing
- Staff will use information from confirmatory Office of Nuclear Regulatory Research work to perform independent evaluation of licensee chemical effect evaluations

Coatings Issues

- NRC adopted conservative positions for coatings zone of influence, coatings debris characterization, non-qualified coatings failure, and coatings debris transport

- Plants could deviate from these positions with an adequate technical justification (test data)

- Staff will evaluate testing that licensees provide to ensure that it is technically sound and applicable

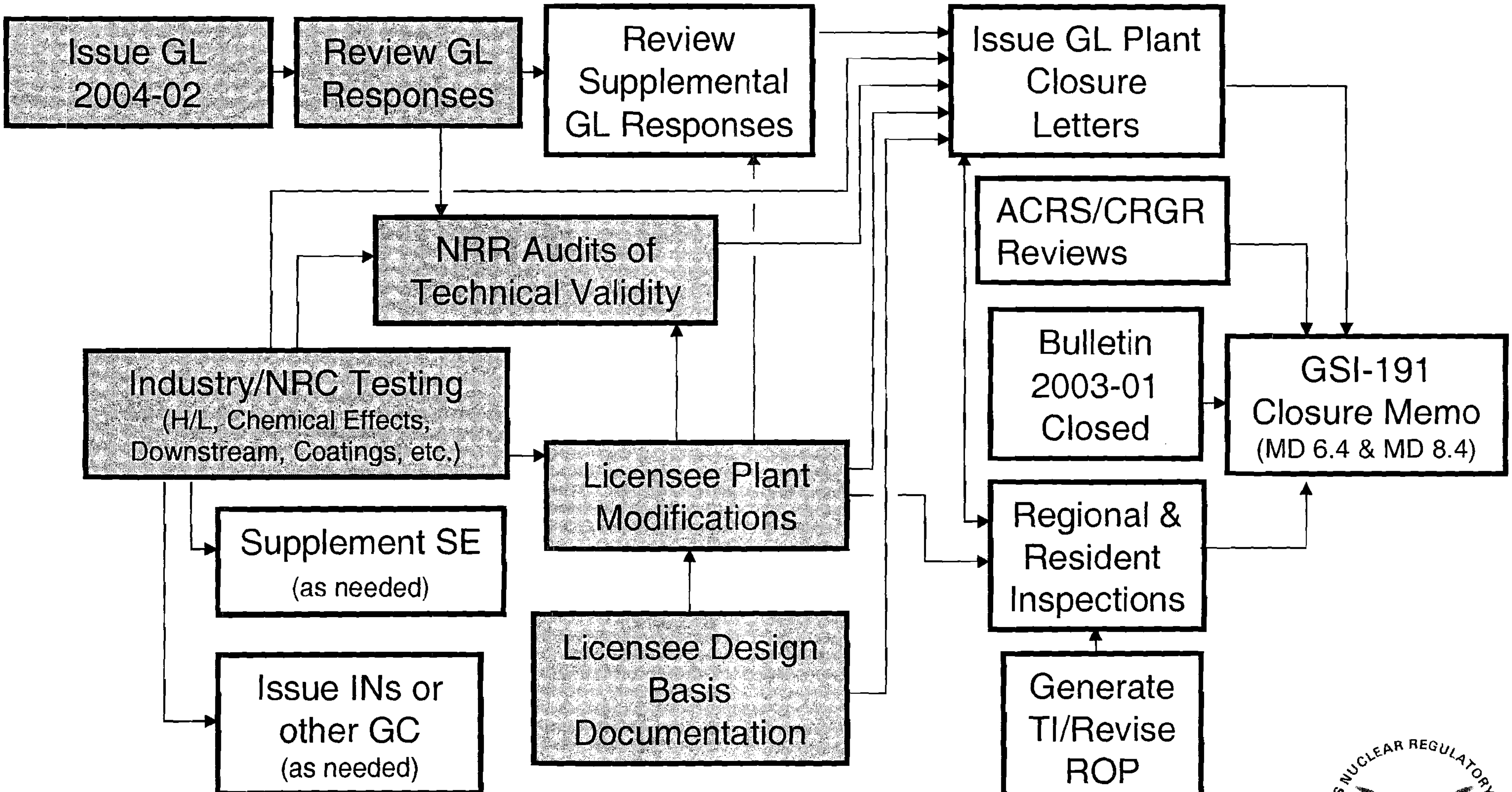
Downstream Effects

- Design of systems for handling debris-laden fluids is a mature science
- Almost all licensees are using the Westinghouse Owners Group (WOG) report WCAP-16406P for their evaluation methodology
- Staff reviewed WCAP and provided comments to the WOG in October 2005

Path Forward - Downstream Effects

- Staff has draft review guidance for fuel and reactor vessel issues
- Staff will continue to work with the WOG and licensees on WCAP issues, site-specific issues, and responses to staff's requests for additional information
- Staff will review licensee modifications and industry tests for downstream issues, including in-vessel issues
- Staff will run confirmatory computer analysis of effects of potential flow blockage in the vessel

GSI-191 Resolution Path Forward



Regulatory Approach to Issue Closure

- High confidence that enlarging strainers will enhance safety – staff expects modifications by end of 2007
- Additional measures may be identified as a result of ongoing testing
- NRC has provided an approved resolution methodology and will verify adequacy of implementation through inspections and audits
- Licensees are responsible for resolving sump issues at their plants
- Industry developing additional guidance, on which staff will comment
- Solutions are largely plant specific
- Issue closure based on reasonable assurance plants compliant with 10 CFR 50.46 and other applicable regulations

UNITED STATES NUCLEAR REGULATORY COMMISSION

Acronyms for Figures

ANL	Argonne National Laboratory
GC	generic communications
H/L	head loss
ICET	Integrated Chemical Effects Test
IN	Information Notice
LANL	Los Alamos National Laboratory
MD	Management Directive
ROP	Reactor Oversight Process
SE	safety evaluation
SwRI	Southwest Research Institute
TI	Temporary Instruction

UNITED STATES NUCLEAR REGULATORY COMMISSION

Industry Activities to Address PWR ECCS Sump Performance

ACRS Meeting
March 9, 2006

John Butler
Senior Project Manager
Nuclear Energy Institute
(202)739-8108
jcb@nei.org

GSI-191, PWR Sump Performance

- GSI-191 applies to all pressurized water reactor designs
  - 69 PWR units in U.S.
- Each unit is unique in one or more important design aspects:
  - Insulation materials
  - Containment coatings (both qualified and unqualified)
  - Containment design (compartmentalized, open)
  - Sump design
  - NPSH requirements
- The high level of design variation requires plant-specific resolution approach for each plant

NEI

Evaluation Guidance Development

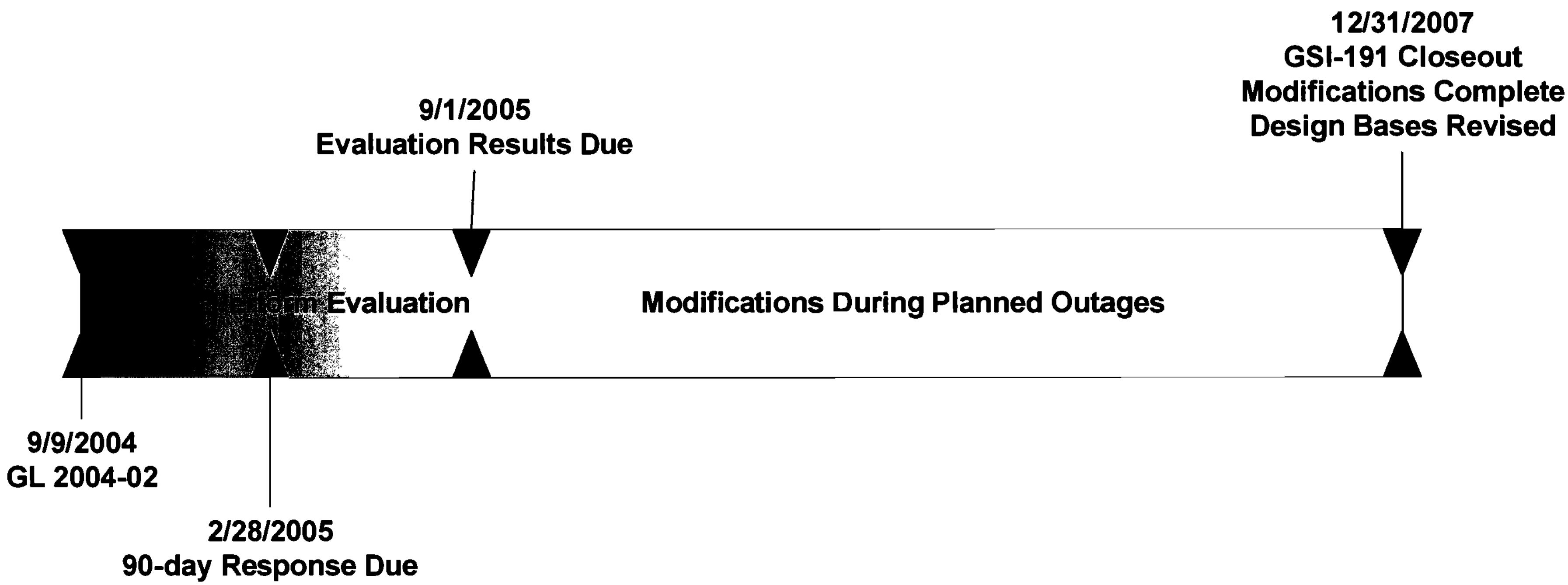
- Development of Industry Guidance began following issuance of NUREG/CR-6762, *Parametric Evaluation for PWR Recirculation Sump Performance* (2002)
- NEI 02-01, Debris Sources Inside Containment (2002) issued to begin plant data collection activities (development sponsored by WOG)
- Bulletin 2003-01, Potential Impact of Debris Blockage on Emergency Sump Recirculation at PWRs (2003) called for compensatory actions

GL 2004-02

- GL 2004-02, *Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors*, issued September 2004
- Requested PWR licensees to perform an evaluation of recirculation functions and, if appropriate, take additional actions to ensure system function
- GL schedule:
  - By 2/28/05 – provide description of evaluation methodology to be used and schedule for completion
  - By 9/1/2005 – provide results of evaluation
  - By 12/31/2007 – complete all actions, including necessary plant modifications

NEI

GL 2004-02 Schedule



Industry Guidance (NEI 04-07)

- Evaluation guidance, development led by WOG, issued December 2004
- Developed to provide a practical and realistically conservative set of methods
- Used to identify "problem areas" and focus on cost effective areas for refinement and resolution
- NRC issued SER, December 2004
  - SER added conservatism

NEI

Supplemental Guidance

- WOG guidance was prepared to support evaluation in two areas not addressed in NEI 04-07
  - Downstream Effects
  - Results from Joint Industry/NRC Chemical Effects tests and WOG Bench Top Chemical Tests

Downstream Effects

- WCAP 16406-P, *Evaluation of Downstream Sump Debris Effects in Support of GSI-191*
  - Provides methods to perform Downstream Effects Evaluations, issued June 2005
    - Addresses wear, abrasion and blockage impacts of sump screen bypass
  - NRC comments, October 2005
  - Submitted for NRC review in February 2006
  - Current WOG program to address NRC comments and obtain NRC approval

NEI

Integrated Chemical Effects Tests

- Jointly sponsored by Industry and NRC
  - WOG support included development of test plan
- Tests conducted between 11/2004 and 8/2005
  - Test reports published; compiling program information into NUREG document to be developed by March 2006

NEI

Bench Top Chemical Effects Tests

- WCAP-16530-NP, *Evaluation of Post-Accident Chemical Effects in Containment Sump Fluid to Support GSI-191*
  - Issued February 2006
  - Addresses chemical reactions and products in containment sump fluid
  - Provides input for use in plant-specific evaluation of chemical effects

NEI

Status of Industry Activities

- Survey conducted January 19th
- All 69 plants have completed evaluations necessary to assess need for strainer modifications
  - Three units have assessed that their current strainers are appropriately sized
  - Sixty-six units plan to replace their current strainers

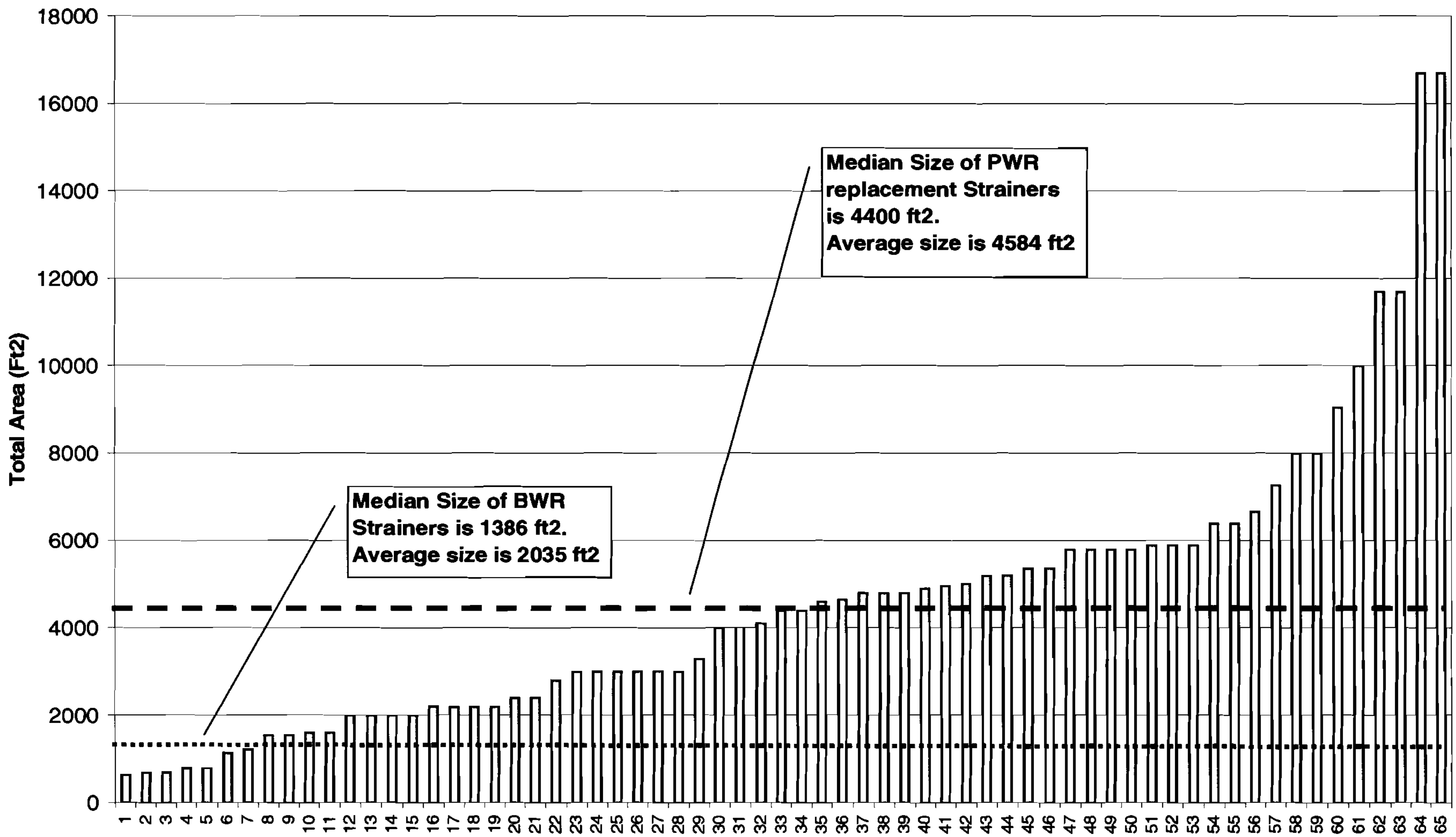
NEI

Strainer Vendors

- Of the 66 units planning to replace strainers, 65 have selected a vendor/design concept
  - One plant finalizing design evaluation before selecting vendor
  - Five strainer vendor teams:
    - Enercon/Alion/Westinghouse/Transco
    - Framatome/PCI
    - GE
    - CCI
    - AECL
  - Four units intend to install active strainers

NEI

Estimated Size of PWR Replacement Strainers (Passive Strainers only)



NEI

Factors Affecting Strainer Size

- The variability in sizes reflects a number of factors, including:
  - Plant design
  - Conservatism in methodology application
  - Retained Margin

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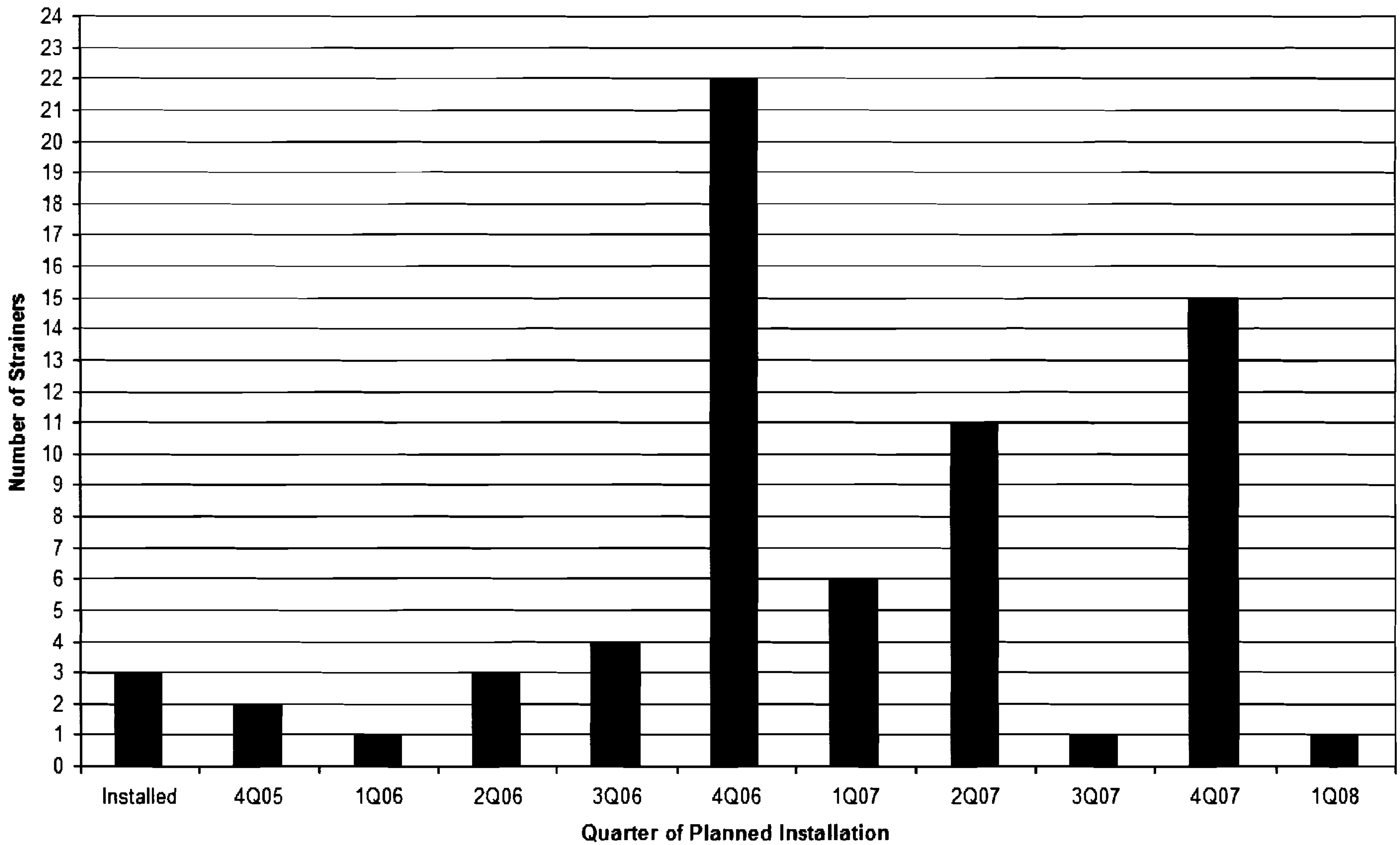
Planned Strainer Installation



Plant Specific Modifications

- Actions to address debris sources
  - ~45% identified near term actions to modify or reduce problematic insulation materials
  - ~20% identified non-programmatic changes to modify or reduce problematic coatings and latent debris
- Containment modifications beyond strainer installation
  - >30% identified modifications affecting debris transport (e.g., debris interceptors)
  - >20% identified other modifications affecting flood-up level, equipment storage
- Downstream effects
  - >50% indicated plans for modification of downstream flow pathways
- Programmatic changes

NEI

Plant Specific Testing

- All 69 units identified plans for prototypic strainer testing
- ~35% identified plans for plant specific testing of debris generation and transport
- ~46% identified plans for plant specific testing of coatings debris generation and transport
- >50% identified plans for plant specific testing for downstream effects of debris bypass

Industry Test Activities

- WOG Chemical Effects Testing
- Strainer Qualification Testing
- WOG Alternate Buffer Project
- STARS Coatings Tests
- FPL/AREVA NP Coatings Tests

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Summary

- WOG, EPRI and NEI activities are directed toward addressing key areas of uncertainty and minimizing plant impacts
- Activities for plant-specific resolution of GSI-191 are continuing

NEI

Overview of NRC-Sponsored Research Supporting GL2004-02 Resolution

Mark Cunningham
Robert L. Tregoning
Office of Nuclear Regulatory Research

Advisory Committee on Reactor Safeguards
March 9, 2006

7

Research Questions Investigated in Support of GL 2004-02 Resolution

- Does the post-LOCA environment generate chemical byproducts which may contribute to sump clogging?
- Can chemical byproducts cause head loss during post-LOCA recirculation scenarios?
- What variables affect debris penetration through sump screens? Will such debris clog surrogate throttle valves?
- Can debris head-loss data be used to develop predictive correlations?
- Can coatings debris transport within containment to the sump screen?

General Research Philosophy

- **Motivation**: Recognized that research was necessary in important technical areas to ensure adequate resolution of GL 2004-02
- **Broad Objectives**
  - Focus on technical areas having highest uncertainty (ACRS, staff, industry) and where generic evaluation provides the most impact
  - Conduct parametric and/or scoping studies to evaluate important variables over ranges of representative conditions
  - Interact with regulatory staff and industry to inform testing approach & conditions
- **Goals**
  - **Integrated Chemical Effects Testing (ICET) Program:** Provide basic technical knowledge to industry and staff on formation of chemical byproducts
  - **Other Programs**
    - Conduct confirmatory research for staff use in conducting an independent review and assessment of licensee GL 2004-02 evaluations
    - Make important results publicly available to inform ongoing industry activities

Technical Areas of Study

- **Chemical effects**: Investigate contributions to sump screen head loss
  1. Determine potential for chemical by-product formation within containment pool environments.
  2. Characterize, predict, and investigate head loss for significant by-products.
- **Particulate head loss**: Integrate testing results with analytical model to develop correlations for evaluating head loss of PWR insulation materials
- **Downstream effects**: Identify significant variables for consideration in ECCS performance evaluation.
  1. Quantity of ingested insulation debris
  2. Clogging within HPSI throttle valves
- **Coatings transport**: Evaluate the transportability of coating chips to the sump screen

Chemical Effects: Objectives

- Determine, characterize, and quantify the chemical reaction products that may develop in representative PWR containment pool environments
  - Integrated chemical Effects Testing (ICET) Program: Los Alamos National Laboratory (LANL).
- Investigate potential for chemical products to contribute to sump screen head loss
  - Argonne National Laboratory
- Evaluate accuracy of thermodynamic predictions on the quantities and species of chemical products which form
  - Center for Nuclear Waste Regulatory Analyses (CNWRA) @ Southwest Research Institute

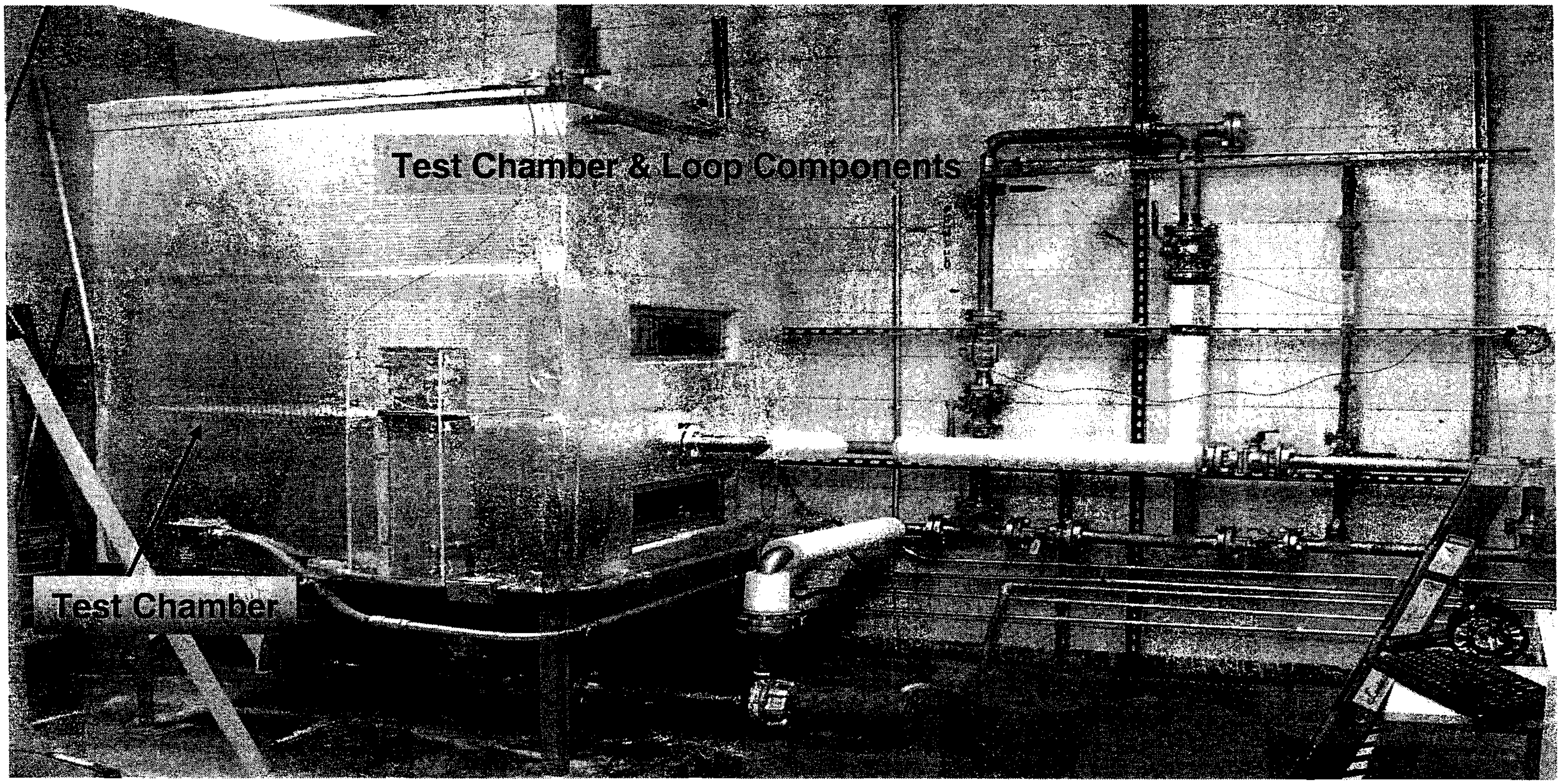
ICET Approach

- Evaluate chemical by-product formation over 30 day mission time
- Choose representative test parameters using industry surveys
- Consider contribution from submerged and un-submerged materials: Al, Cu, Zn, GS, concrete, fiberglass and calcium silicate insulation
- Simulate plant conditions using scaling constant: ratio of surface area of coupon material (or weight/volume of insulation) to water volume

Test	Temp (C)	Buffering Agent	Initial pH	Boron (ppm)	Insulation Mixture	Corresponding Plants*
1	60	NaOH	10	2800	100% fiberglass	25
2	60	Na_3PO_4	7	2800	100% fiberglass	20
3	60	Na_3PO_4	7	2800	80% cal-sil 20% fiberglass	6
4	60	NaOH	10	2800	80% cal-sil 20% fiberglass	9
5	60	$Na_2B_4O_7$	8	2400	100% fiberglass	9

* ICET environment most similar to plant. Some plants fit multiple environments.

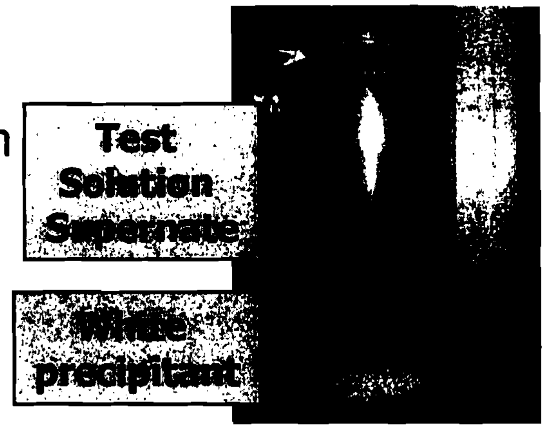
ICET Test Loop

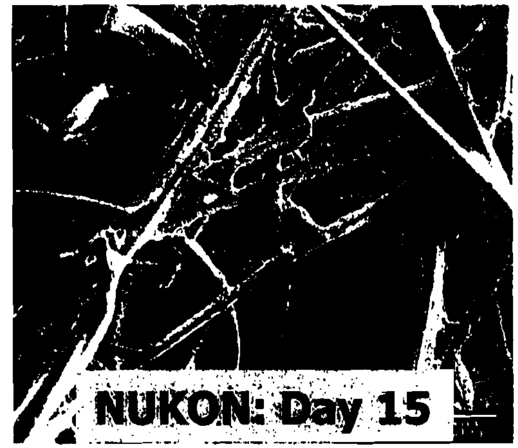


ICET: Significant Results

Test #1: NaOH & NUKON

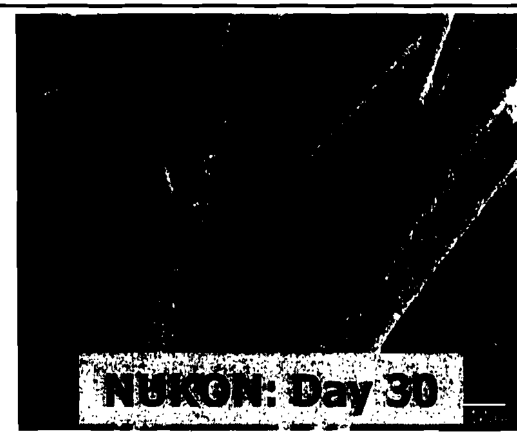
- White precipitate (aluminum oxyhydroxide)
- Insulation deposits
- Significant Al weight loss





Test #2: Na_3PO_4 & NUKON

- Insulation deposits



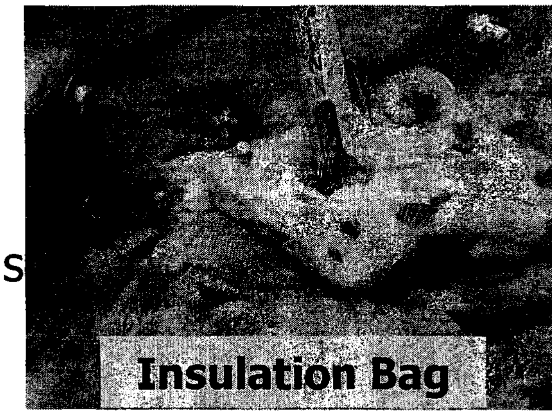
Test #5: $Na_2B_4O_7$ & NUKON

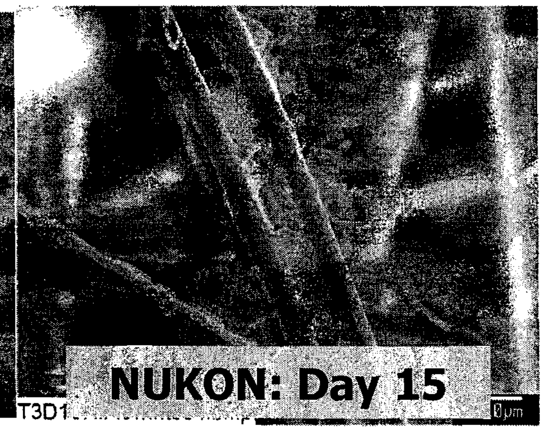
- Similar products as Test #1
- Less quantity and slower to form at lower temps

ICET: Significant Results

Test #3: Na_3PO_4 & Cal-Sil/NUKON

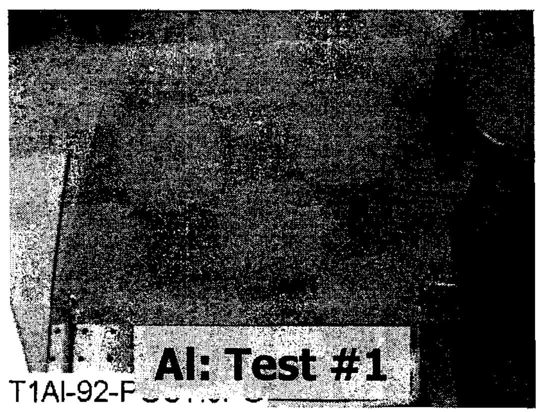
- During test: White flocculent material observed
- Post-Test:
  - White substance $\{Ca_3(PO_4)_2\}$ coating test chamber materials
  - Insulation deposits

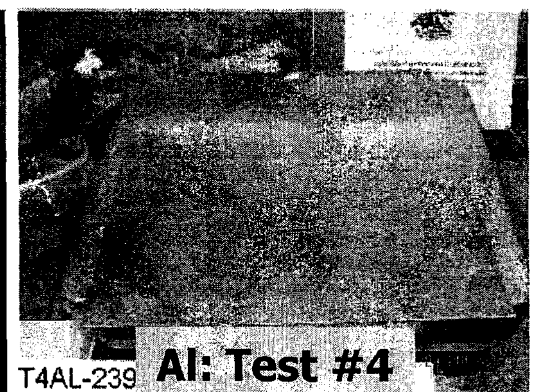




Test #4: NaOH & Cal-Sil/NUKON

- Much less insulation deposits
- Minimal aluminum weight loss
- Thin white coating ($CaCO_3$) on Al specimens

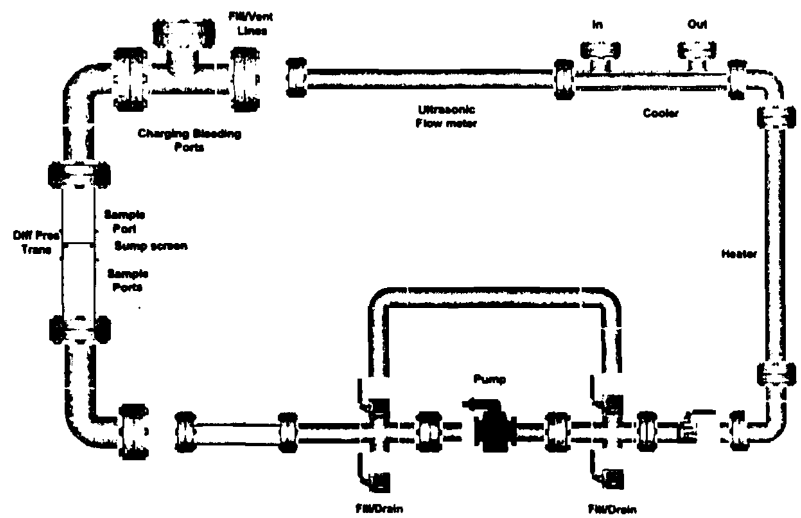




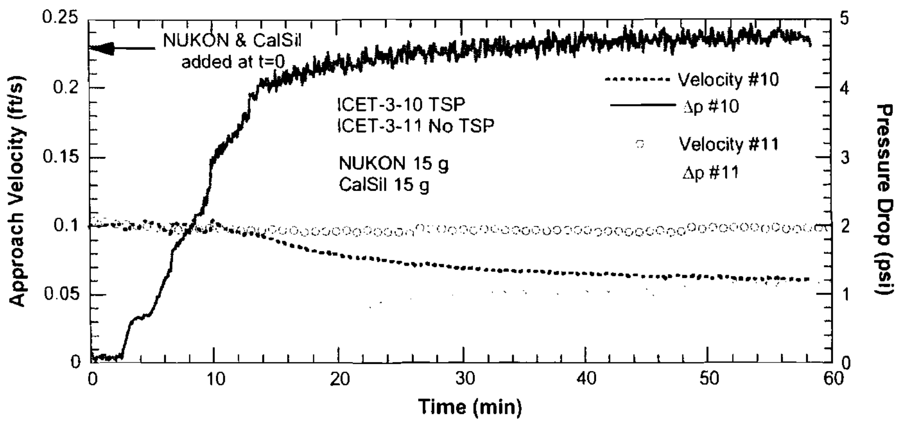
Chemical Head Loss Testing: Approach

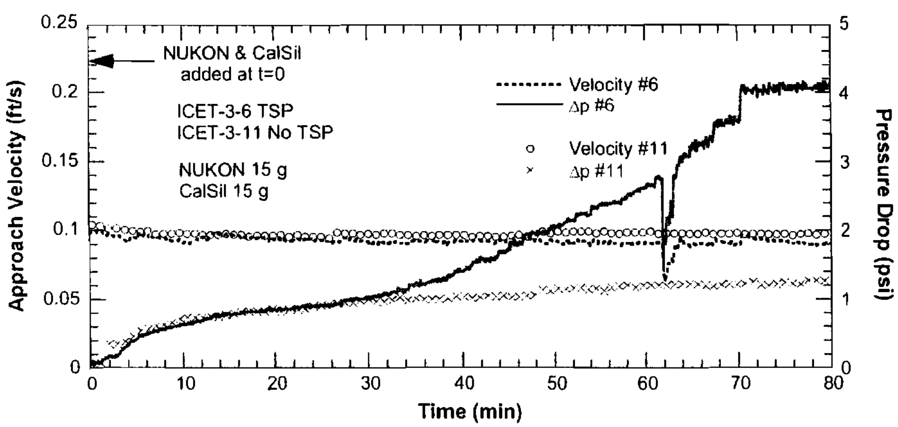
- Simulate chemical products observed in ICET
- Examine effects over a broad, representative range of environmental variables (time, temperature, concentrations, etc.)
- Conduct single effects tests in closed vertical loop instead of integrated tests
- Evaluate plant-relevance using scaling parameters
  - Head loss: mass of chemical product & debris per sump screen area
  - Product formation: mass of chemical product per containment volume

Chemical Effects Head Loss Test Loop



Chemical Head Loss Testing: Significant Results





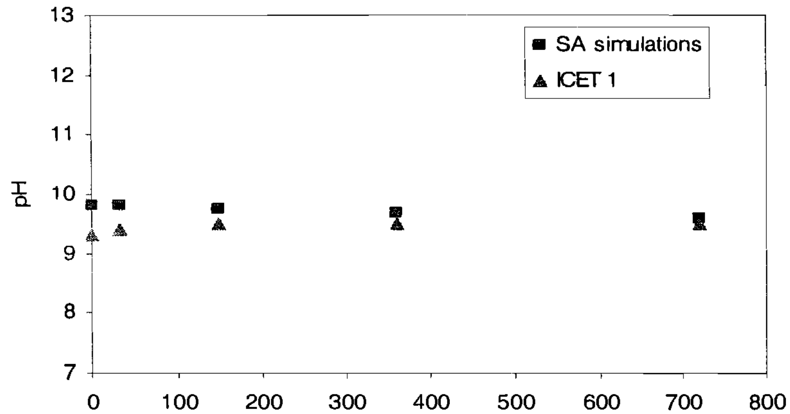
Head Loss in Na_3PO_4 Environments

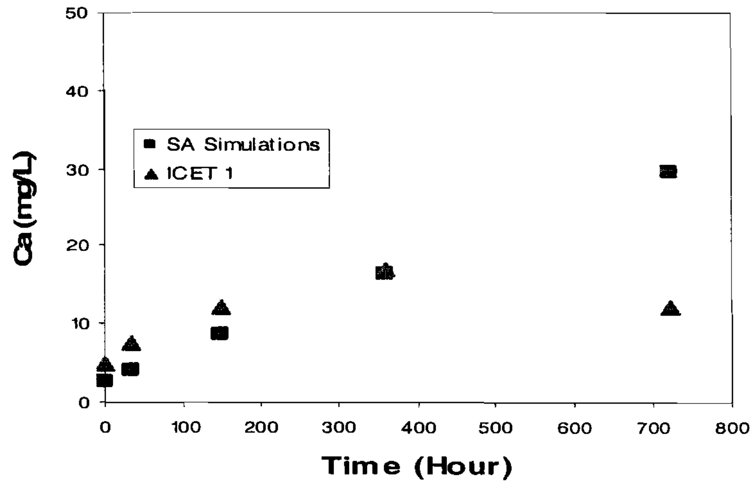
- Head losses with chemical products can be greater than with an corresponding amount of cal–sil
- No significant difference in maximum head loss apparent as a function of cal-sil/Na_3PO_4 dissolution rates
- Relative contribution of $Ca_3(PO_4)_2$ to head loss depends strongly on the debris loading
  - Biggest contribution: Fiber bed saturated with chemical product
  - Similar behavior observed as with particulate loading

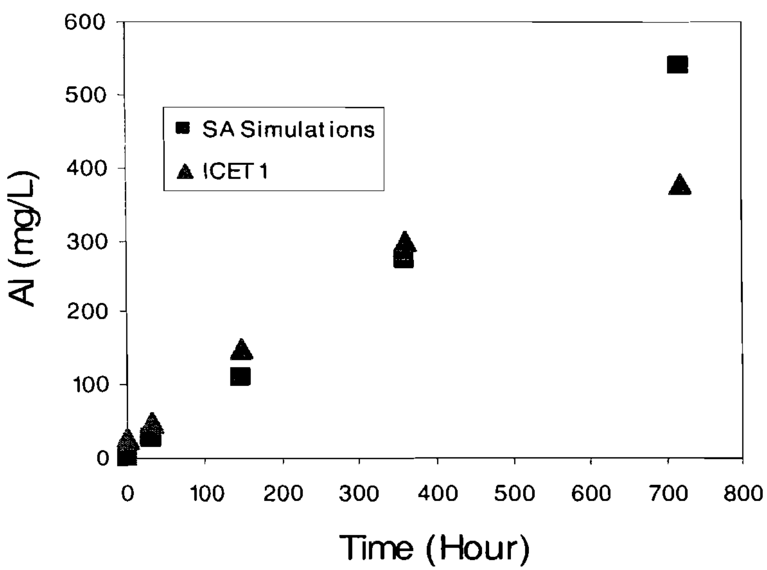
Chemical Speciation Prediction: Approach

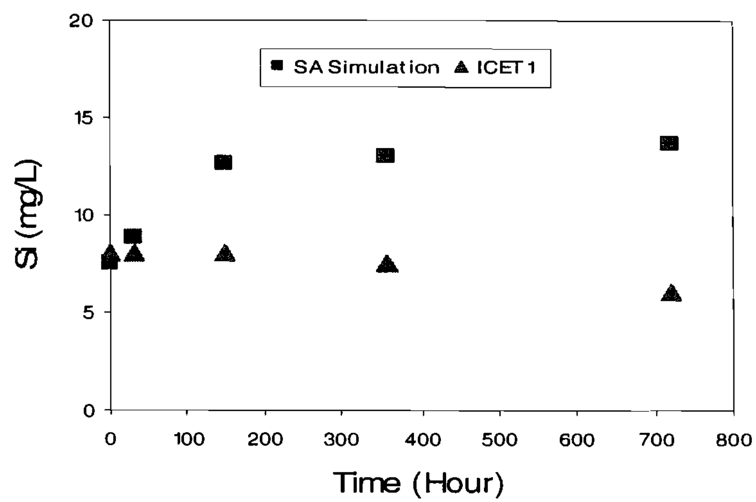
- Evaluate the feasibility of utilizing commercially-available thermodynamic simulation codes for predicting chemical species formation in plant-specific environments
- Measure corrosion rates of important materials: Al, Cu, Zn (galvanized steel), fiberglass, cal-sil, carbon steel, concrete
- Perform initial blind predictions of the ICET experiments to compare the quantity and type of solid species which form
- Conduct follow-on calibrated simulations to omit species not observed in ICET testing

Chemical Speciation Prediction: Calibrated Simulation of ICET #1









Chemical Effects: Initial Conclusions

- Chemical products, precipitants and gelatinous-like materials can form in a representative PWR containment pool.
- Relatively small changes to important variables (e.g., pH, insulation) can significantly affect the quantity, types, and nature of chemical by-products that form.
- Chemical products in the environments examined thus far can contribute significantly to sump screen head loss.
- In Na_3PO_4 environments, small inventories of dissolved Ca may significantly contribute to head loss.
  - Greater than 25 ppm dissolved Ca
  - Depending on fiber loading, greater than 0.5 kg/m^2 of cal-sil screen loading
- Blind predictions using only input corrosion data were not successful.
- Most accurate results achieved by suppressing thermodynamically species not observed in ICET testing.

Particulate Head Loss: Testing & Modeling

- Contractor: Pacific Northwest National Laboratory
- RES Investigator: William Krotiuk
- Objectives
  - Develop improved model to conservatively predict pressure drop across and compression of a debris bed on a sump screen
  - Utilize test data to support model development of empirical constants and independently validate applicability
  - Experimentally investigate important mechanistic parameters affecting head loss in mixed debris beds

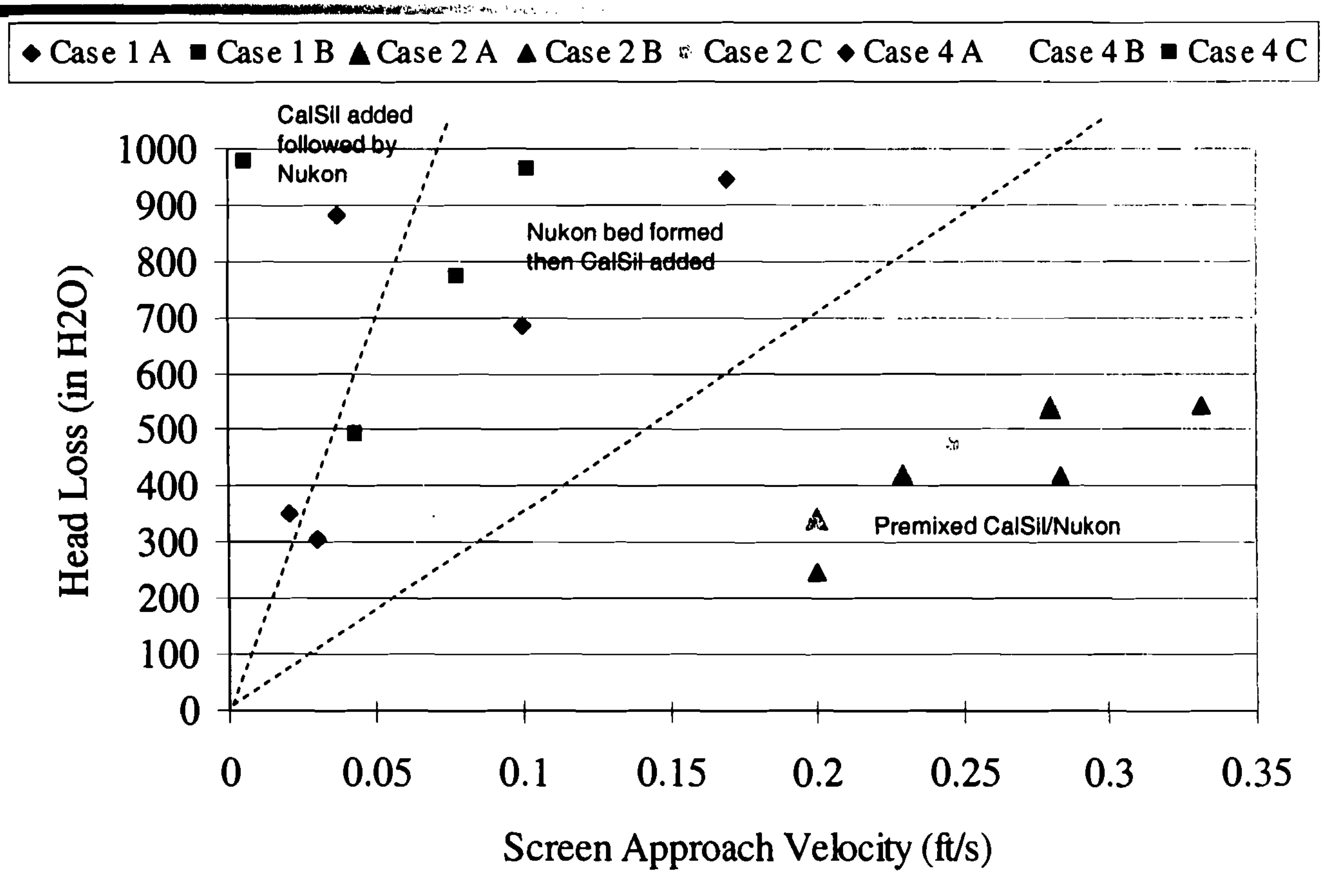
Particulate Head Loss: Modeling Approach

- Base model on classical form of porous medium flow equation (Ergun Equation) accounting for viscous and kinetic flow components
  - Develop improved method to predict debris bed compressibility
  - Develop particulate "saturation" relation for mixed fibrous (NUKON), particulate (cal-sil) debris beds
  - Identify the limiting particulate concentration as a function of NUKON bed characteristics
- Model formulation
  - One homogeneous control volume
  - two homogeneous control volumes through debris bed thickness, each with independent debris concentration distribution
- Evaluate model assumptions and validity with head loss test data from variety of test programs (PNNL, LANL, ANL)

Particulate Head Loss: Testing Approach

- Design closed-loop facility to control of test parameters over a range of relevant conditions
  - Pressurize loop to eliminate gas and two-phase flow conditions
  - Measure bed height in situ
  - Permit separate filtering of suspended particles
- Develop standardized debris preparation so that material with repeatable characteristics can be produced by independent operators.
- Characterize the debris bed after the test
  - Measure mass of individual constituents in bed.
  - Evaluate through-thickness particulate concentration within the debris bed .
- Principal test variables
  - Debris bed mass and relative composition
  - Particulate distribution within bed
  - Debris arrival sequence
  - fluid temperature
  - flow velocity

Particulate Head Loss: Significant Test Results



Target test conditions

- NUKON: 1.01 kg/m^2
- Cal-sil: 0.51 kg/m^2
- Total: 1.52 kg/m^2

- Debris arrival sequence can significantly affect measured head loss
  - Localized bed saturation is likely important contributor
  - Debris bed sectioning being used to investigate bed homogeneity and particulate distribution within the bed

Particulate Head Loss: Initial Conclusions

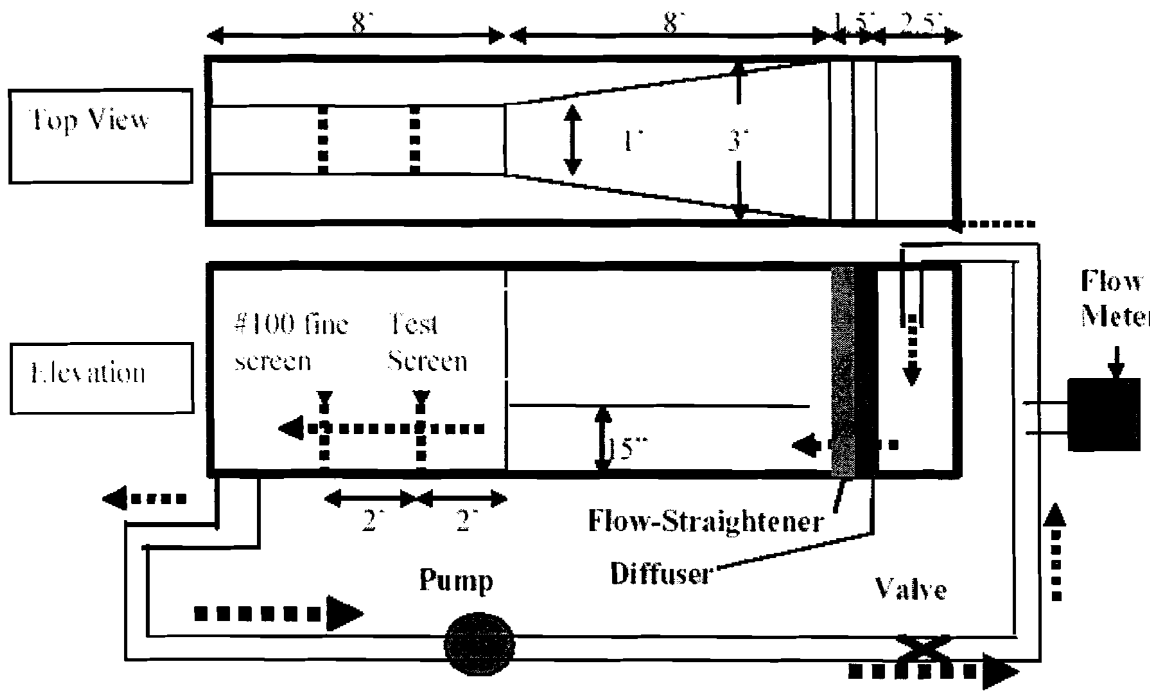
- Significant head loss increases occur when a fibrous debris bed become saturated with small particles either uniformly or locally
- Fibrous debris is required initially to trap finer particulates
- Debris entrapment at the test screen is a function of debris type
  - Most fibrous insulation added accumulates in the debris bed
  - Depending on particulate mass added, significant (as much as 50%) particulate remains suspended during testing

Downstream Effects

- Contractor: Los Alamos National Laboratory
- Objectives
  - Debris Ingestion (Phase I): Examine variables that affect the amount of insulation debris that can pass through a sump strainer screen and become ingested within the ECCS system (NUREG/CR-6885).
  - Throttle Valve Blockage (Phase II): Evaluate effect of ingested insulation debris on blockage of surrogate high-pressure safety-injection (HPSI) throttle valves.

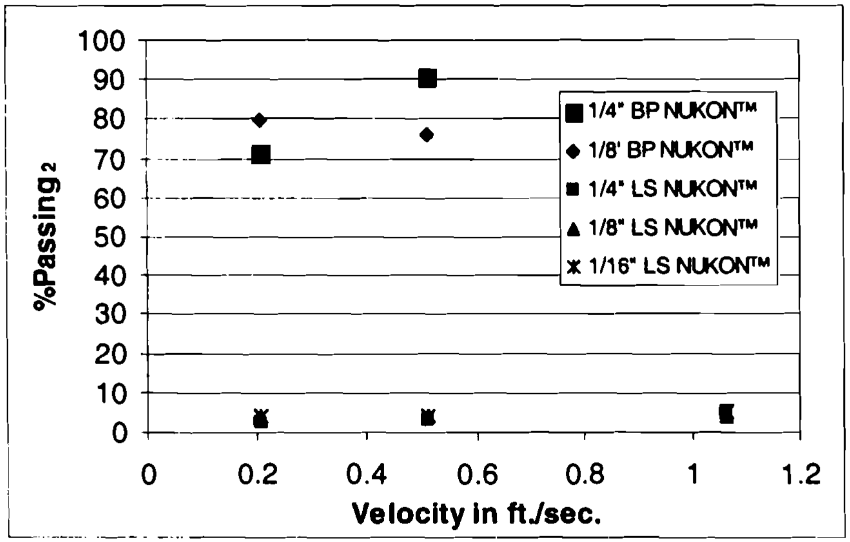
Downstream Effects: Debris Ingestion

- Approach: Phase I
  - Evaluate fiberglass (NUKON), cal-sil, and reflective metal insulation (RMI) debris
  - Conduct constant velocity testing within linear flume
  - Pass individual debris types through clean test screens
- Principal test variables
  - Debris size
  - Debris agglomeration
  - Debris location: floor or within flow
  - Flow velocity

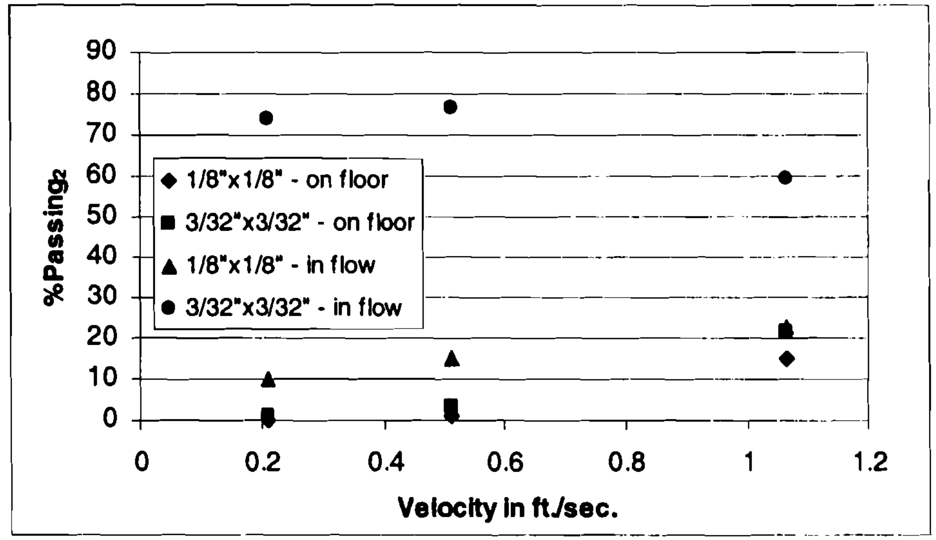


Debris Ingestion: Significant Results

NUKON: 1/16", 1/8", & 1/4" screens



RMI: 1/8" screen

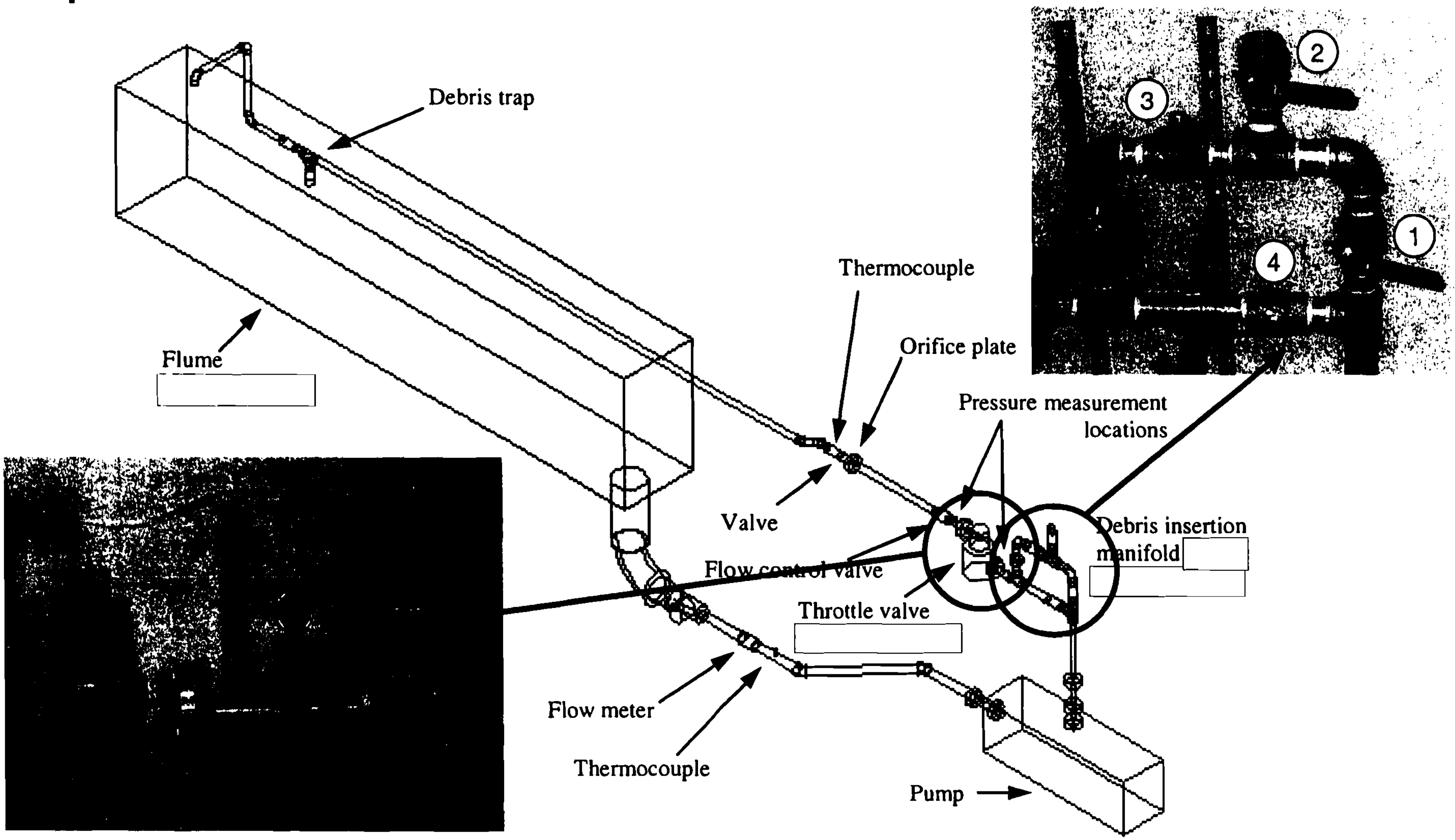


- A significant amount of NUKON debris arriving in finely separated fibers (BP) passed through the test screens while larger, agglomerated pieces (LS) did not.
- Significant percentages (up to 75%) of RMI debris passed through the test screens when the debris was smaller than the screen opening and was introduced directly into the flow at these velocities.
- Virtually all cal-sil insulation particulates passed through any size test screen.

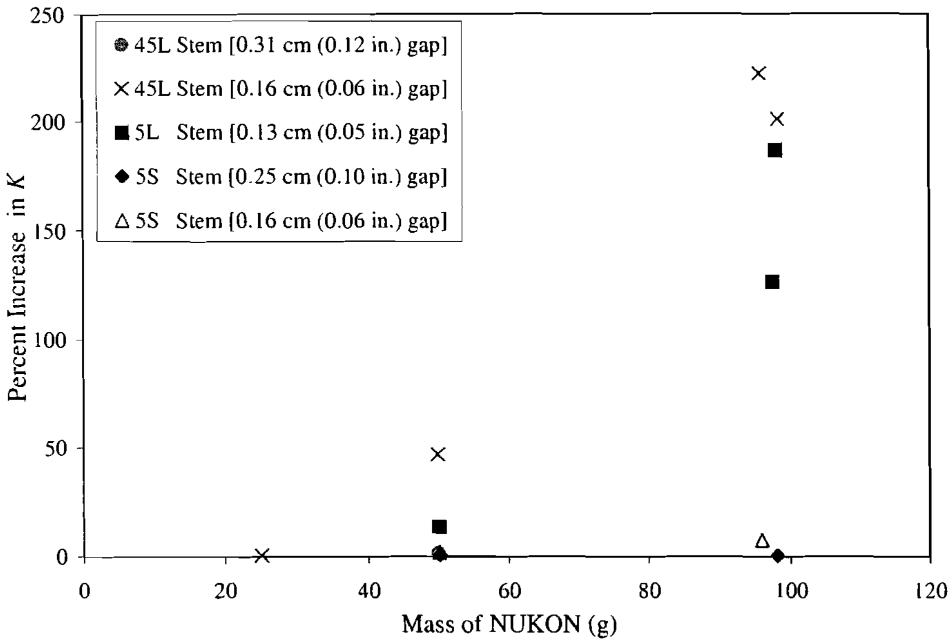
Downstream Effects: Throttle Valve Blockage

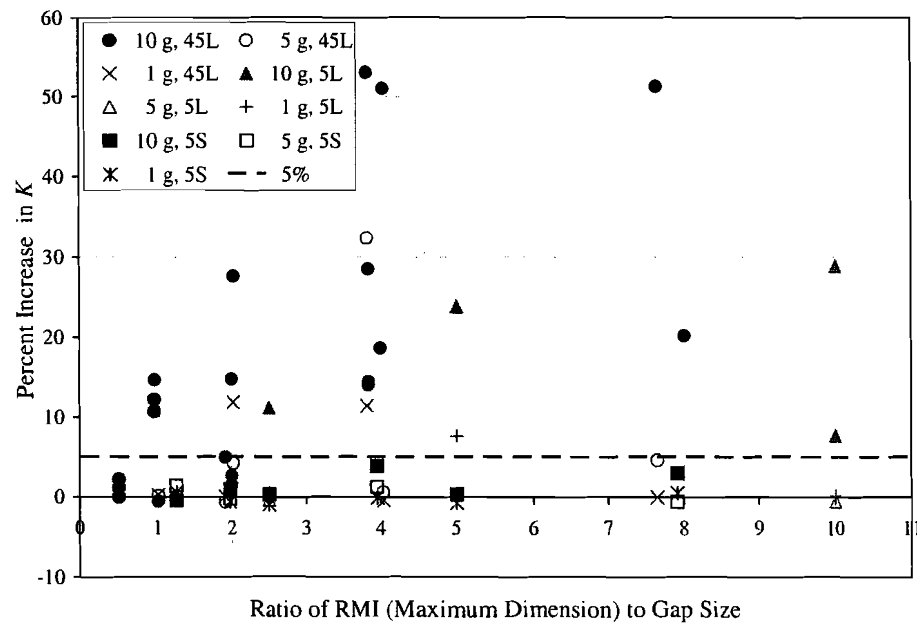
- Approach: Phase II
  - Select ingested debris characteristics for RMI, NUKON, cal-sil based on the phase I study
  - Test a surrogate valve chamber with flexible geometry: 3 configurations with different contact angles and seat diameters
  - Parametrically study important variables to identify plausible debris retention mechanisms
  - Determine relationship between flow area and valve loss coefficient
  - Infer debris retention based on increases in valve loss coefficient
- Principal test variables
  - Valve geometry
  - Debris type and size
  - Valve gap height setting
  - Single input vs. accumulated debris
  - Single vs. mixed debris

Throttle Valve Blockage: Test Apparatus Schematic



Throttle Valve Blockage: Significant Results





- Greatest loss coefficient increases resulted from NUKON loading
- Valve loss coefficient increased as the RMI debris size relative to the gap setting increased
- Measured loss coefficient is a function of valve geometry
- Considerable variability was apparent in results

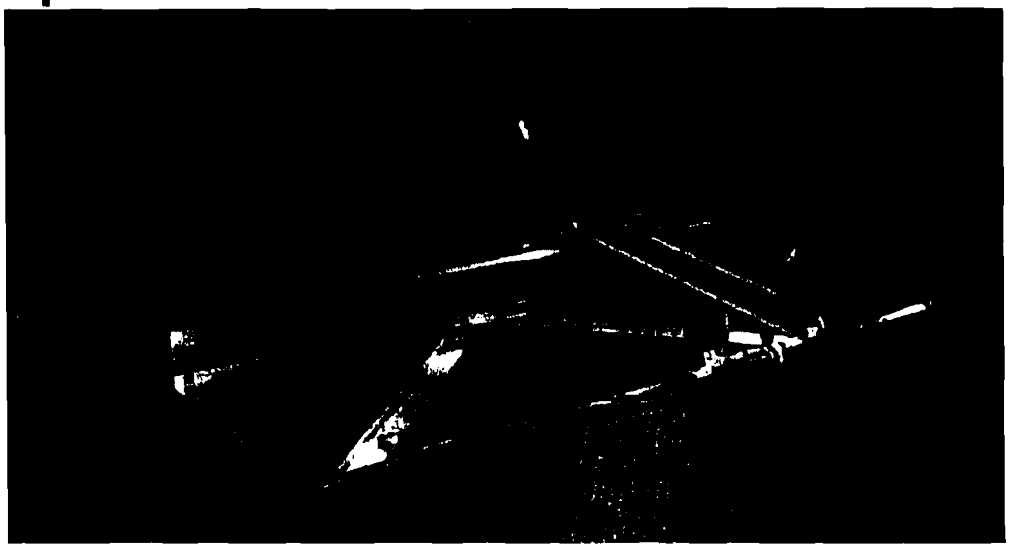
Downstream Effects Testing: Conclusions

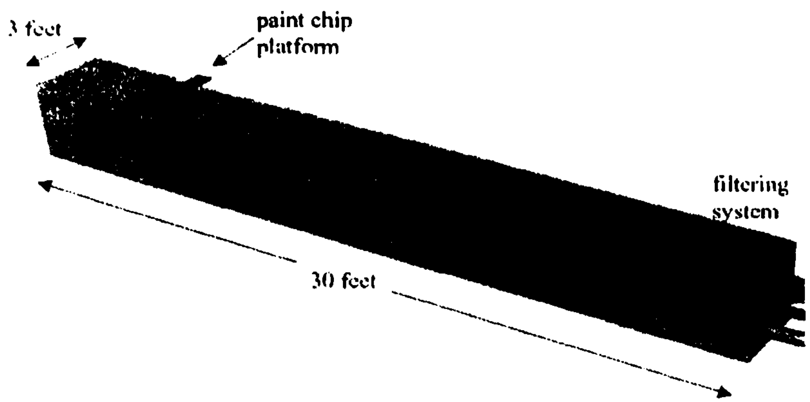
- A significant percentage of finely-divided, suspended debris (NUKON, RMI, cal-sil) can pass through clean screens
- It is important to understand size distribution and timing of debris arriving at screen to determine percentage of debris ingestion
- All debris types (except for finely divided cal-sil) and combinations resulted in valve loss coefficient increases for a surrogate HPSI throttle valve
- Some tests demonstrated that finer debris (cal-sil) could be retained if blockage is initially established with coarser debris (NUKON, RMI)
- Debris accumulation over time was observed, but the effects were not monotonic and self-clearing was observed at certain points

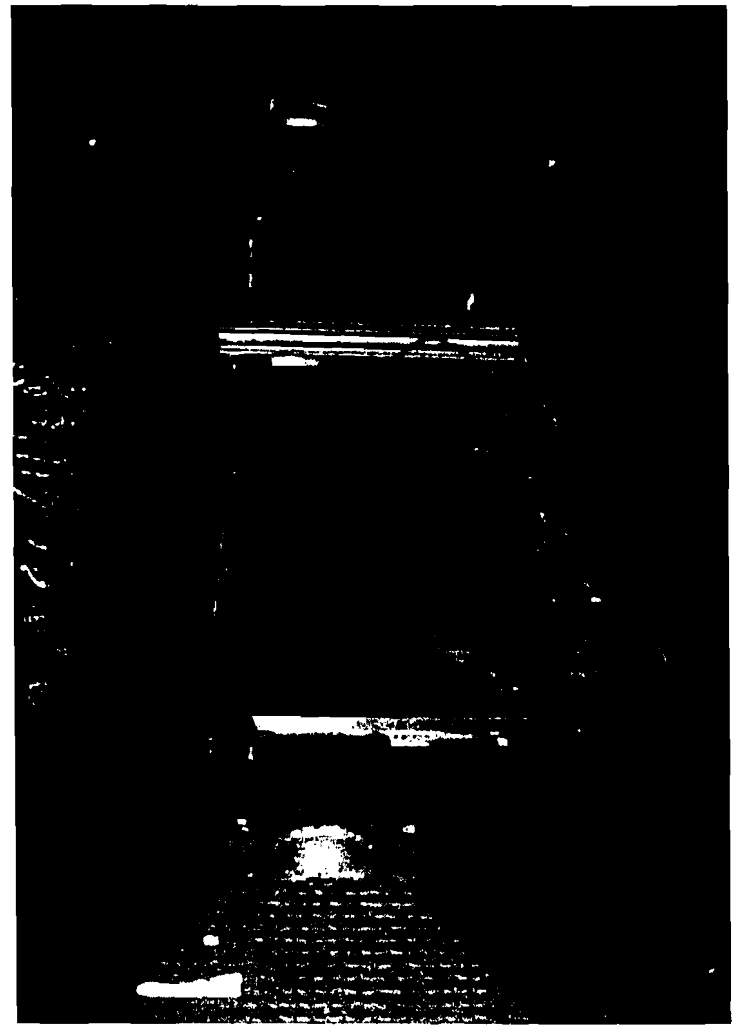
Coatings Transport Testing

- Contractor: Naval Surface Warfare Center, Carderock Division
- Objective: Characterize the transport behavior of coatings debris in water under stagnant and flow conditions.
- Approach
  - Study 5 coating systems representing a range of representative physical characteristics (e. g., specific gravity, thickness, surface roughness)
  - Perform quiescent settling tests: terminal velocity and time-to-sink.
  - Conduct uniform flow transport testing: tumbling steady-state velocity.
- Principal test variables
  - Debris size: 1/64 inch to 2 inch
  - Debris shape: flat and curled
  - Flow velocity

Coatings Transport Testing: Transport Test Apparatus







Coatings Transport Testing: Preliminary Observations

- Time-to-sink is significantly influenced by specific gravity (SG)
  - Alkyd coatings (SG = 1.05) did not sink
  - Heavier coatings typically sank within 1 second
- Transport velocities were influenced primarily by SG and chip shape
  - Alkyd coating (SG = 1.05) injected into the flow transported at 0.2 ft/s
  - Heavier coatings had higher transport and tumbling velocities
  - Curled chips generally had lower tumbling velocities than flat chips

Important Messages

1. NRC's research program is designed to provide basic conceptual understanding about several important technical issues which impact ECCS functionality
2. NRC's primary research role is to provide confirmatory information so the staff can independently evaluate whether licensees satisfy regulatory requirements
3. Several important research findings have been discussed that should be considered in reaching an acceptable resolution of the technical issues raised in Generic Letter 2004-02
4. Thorough understanding and consideration of plant-specific issues is required to assess the implications of research findings and develop acceptable resolution strategies

Advanced Nuclear Power
THE MAGAZINE OF FRAMATOME ANP
N° 1 February 2001

Nuclear Fuel

ATRIUM Fuel for BWRs: More Power at Less Cost

Framatome ANP's ATRIUM™ fuel assemblies for boiling water reactors (BWRs) offer high operational reliability and excellent fuel utilization to reduce fuel-cycle costs. These characteristics help customers remain competitive in today's deregulated power markets.

ATRIUM fuel assemblies are available for all BWR designs and can be manufactured with enriched natural uranium, as well as enriched reprocessed uranium (ERU) or mixed-oxide (MOX) fuel.

High Fuel Reliability at Ever Higher Burnups

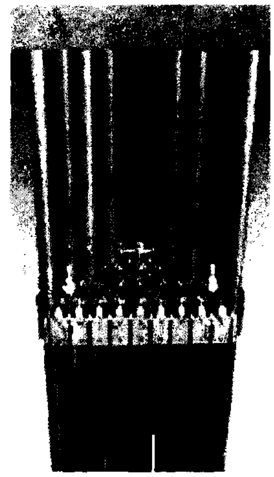
By August 2000, more than 8,300 ATRIUM fuel assemblies have been used in numerous BWR plants all over the world including Germany, Sweden, Finland, Switzerland, the US and Taiwan. The maximum assembly burnup attained so far is 60 MWd/ kgU, just a small step away from the 70 MWd/kgU target that will be reached in 2001. These fuel assemblies, despite the continual increase in burnup, have achieved an average annual fuel failure rate (not counting failures due to debris-induced fretting) of less than $0.5x10^{-5}$.

Advanced cladding materials contribute to a high degree of reliability. For example, fuel assemblies containing Zircaloy-2 clad tubes with iron-enhanced zirconium liners - a cladding variant that was commercially deployed for the first time in 1993 - have suffered only one failure, caused by debris entrained in the fuel assembly from the top.

Serving Customers with Quality and Experience

Framatome ANP's high-quality fuel production minimizes costs, makes licensing easier and forms the cornerstone for reliable and economical reactor operation. It also helps protect the environment and conserve resources. All-inclusive expertise in nuclear systems and processes, know-how from thousands of service projects and in-core fuel management skills gained from hundreds of operating cycles contributes to Framatome ANP's broad-based experience. Customers also benefit from years of experience with licensing procedures worldwide and the comprehensive support provided for all issues related to licensing.

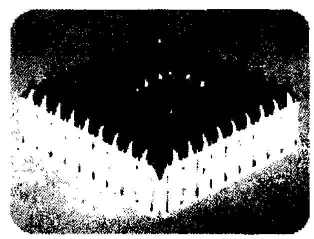
The ATRIUM 10 Concept - A Wide Range of Benefits Founded on Experience and Innovation



The ATRIUM™ Design provides more even power distribution for optimum fuel utilization

Our ATRIUM 10 fuel assemblies contain 91 fuel rods arranged in a 10x10 array, eight of these being part-length rods. The square internal water channel, which occupies nine (3x3) lattice positions, supplies non-boiling water to the center of the fuel assembly. This, along with the part-length fuel rods, results in a more even power distribution across the fuel assembly and thus in optimum fuel utilization. At the same time, the water channel serves as a load-bearing structure, making tie rods superfluous and reducing the mechanical loads imposed on the fuel rods during operation and refueling.

ULTRAFLOW Spacer - Optimum Fuel Rod Cooling for High Operational Flexibility

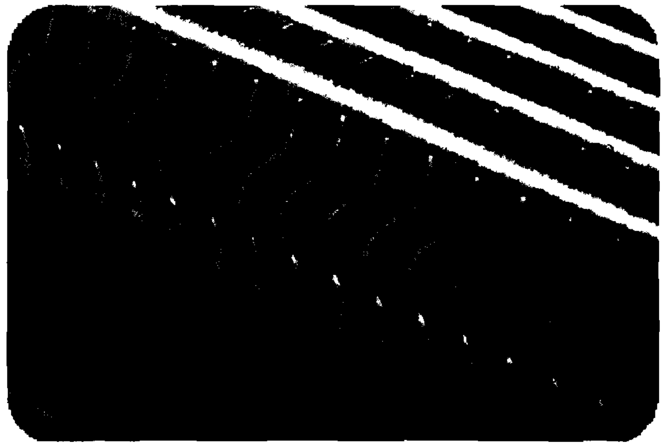


ULTRAFLOW™ Spacer allows greater design margin

In the ULTRAFLOW™ spacer, the swirl vanes protruding into the center of the subchannels between the fuel rods direct water droplets from the two-phase mixture onto the fuel rods, resulting in greater design margins with respect to critical power ratios. A special variant of the ULTRAFLOW spacer made of Inconel (instead of the usual Zircaloy) was developed for the latest generation of ATRIUM

SmallHole and FUELGUARD - Lower Tie Plates That Keep All Debris Out

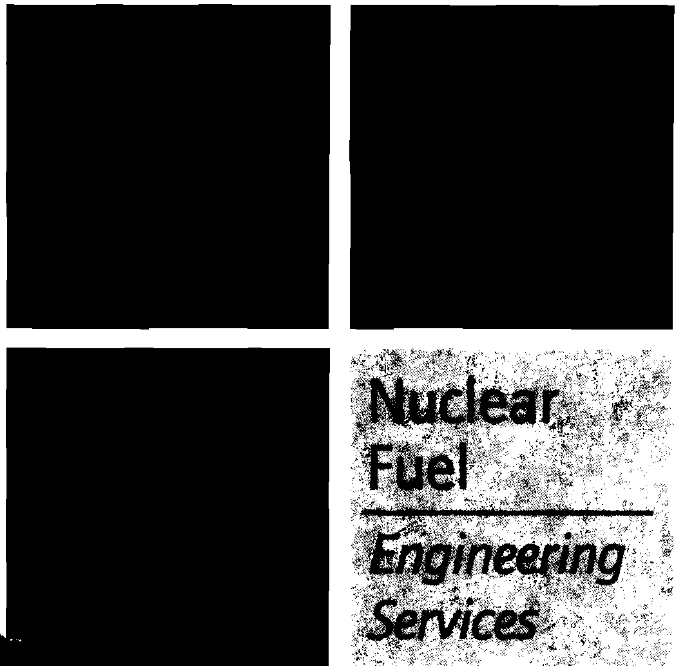
Two different lower tie plates with integral debris filters are available for ATRIUM fuel assemblies: SmallHole, which uses a perforated plate to filter out debris, and FUELGUARD™, in which debris is retained by a parallel array of curved blades. Both designs guarantee an exceptionally high debris-retention efficiency, significantly reducing the probability of fuel damage from debris-induced fretting.



Unique lower tie plates reduce probability of fuel damage from debris-induced fretting

Nuclear Fuel

PERFORMANCE+ Fuel



PERFORMANCE+ fuel helps utilities achieve exceptional fuel reliability and performance in today's challenging operating and commercial environment.

Background

Trends in nuclear fuel operations include efficiency enhancing – but technologically demanding – practices, such as longer cycle lengths into the 18- to 24-month range, increased discharge burnups above 50,000 MWD/MTU, higher primary coolant temperatures and elevated lithium levels. The challenge to nuclear fuel suppliers is to provide product features that support these industry trends without sacrificing fuel reliability or operating margins.

Westinghouse

PERFORMANCE+ fuel offers features which support and implement today's operating plant priorities. It provides utilities with the opportunity to select the combination of features best-suited to their specific goals.

Benefits

Utilities can select among the benefits of these PERFORMANCE+ features:

- A removable top nozzle aligns easily and engages with out external force. It simplifies tooling and takes less time to remove and reattach. Strong, durable inserts reduce inspection requirements. Low-cobalt stainless steel reduces personnel exposure levels during refueling operations. Operating and maintenance costs are reduced and overall reliability increased.

- Axial blankets place more enriched uranium in the inner part of the fuel rod where fuel is used most efficiently; reducing axial neutron leakage by about 50%. Enriched annular axial blankets use neutrons even more efficiently, further reducing fuel enrichment costs.

- Intermediate flow mixer grids enhance the margin to departure from nuclear boiling by enhancing flow mixing and heat transfer in the upper part of the core. Increased margins can be used to help upgrade energy output, or to increase the allowable peaking factor limits for greater flexibility in loading patterns, or to help accommodate longer cycle lengths.

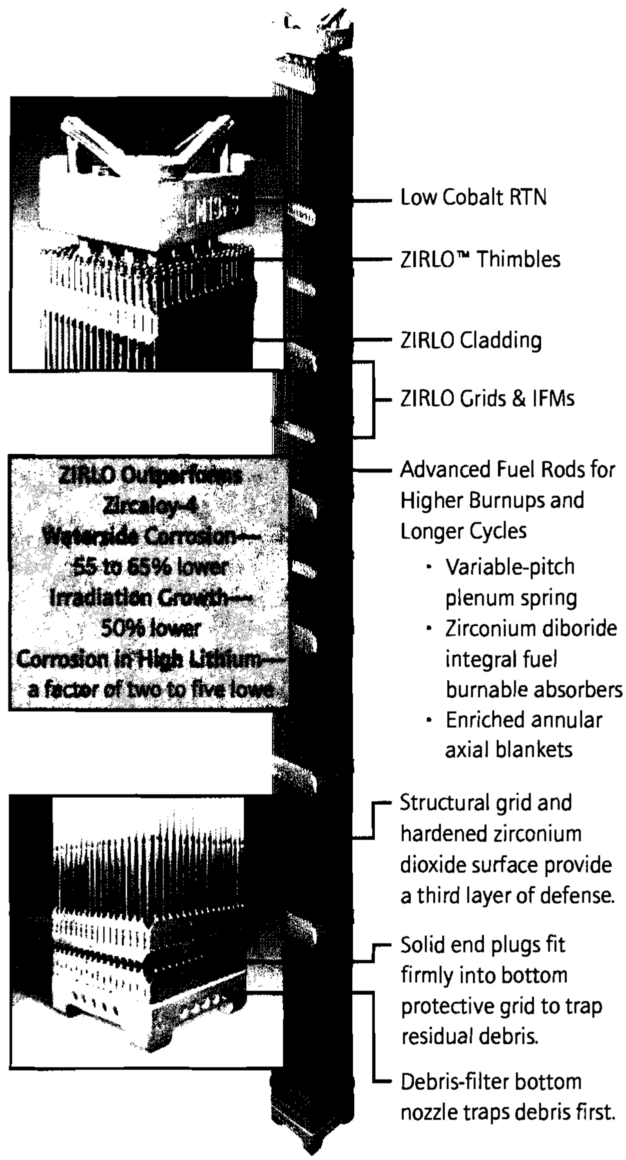
- PERFORMANCE+ fuel utilizes ZIRLO™ material extensively, in the structural grids and intermediate flow mixer grids, as well as in the fuel rod cladding, instrumentation tubes and guide thimbles. ZIRLO cladding provides greater fuel reliability at extended burnups and high fuel duties.

(Continued on back)

For more information, call your local Westinghouse Electric Company sales representative.

It has greater dimensional stability, greater resist ance to corrosion and irradiation growth and creep, and greater resistance to elevated lithium levels than other available cladding materials.

- Zirconium diboride integral fuel burnable absorbers (IFBAs) can help to reduce fuel cycle costs and storage requirements, increase reactor availability and increase loading pattern flexibility. IFBAs provide a smaller residual reactivity penalty than other burnable absorbers, reducing uranium requirements and resulting in up to a 3 percent reduction in fuel cycle costs. Since the absorber is part of the fuel rod, there is no need to handle separate discrete burnable absorbers during refueling or worry about their subsequent disposal.
- Mechanical modifications to the assembly, when com bined with the greater creep resistance of ZIRLO cladding, enhance fuel performance under the more severe duty of extended burnups and high fuel duties. For example, a small, variable-pitch plenum spring provides added space for fission gas. Annular axial blanket pellets, with 25 percent less fuel rod volume, also increase available space for gas release.
- PERFORMANCE+ fuel offers triple protection against debris-induced fuel rod damage. The Debris Filter Bottom Nozzle traps the vast majority of debris before it enters the assembly. A protective grid at the bottom of the assembly reduces the potential for both debris-induced and grid-to-rod fretting damage. And a hard ened layer of zirconium dioxide applied during fabrica tion protects the lower end of the rod and increases wear-resistance early in life when fuel is most susceptible to debris-induced damage. The results: reduced operat ing and maintenance costs, fewer inspections, reduced personnel exposure and less radwaste disposal.



Description

The challenge for nuclear plant operators is to reduce fuel-cycle expenses and lower operating costs while maintaining excellent performance records. PERFORMANCE+ fuel has evolved as a response to customers' needs for increased power plant availability, enhanced reliability and greater operating flexibility. Advances have been made in the three basic elements of the fuel assembly — the fuel cladding, the fuel rod and the fuel assembly skeleton.

Westinghouse Electric Company
Box 355
Pittsburgh, PA 15230

www.westinghouse.com

71412

(9)

NEI

NUCLEAR ENERGY INSTITUTE

Anthony R. Pietrangelo
SENIOR DIRECTOR, RISK REGULATION
NUCLEAR GENERATION

February 28, 2006

Dr. Brian Sheron
Associate Director, Engineering and Safety Systems
Office of Nuclear Reactor Regulation
Mail Stop O5-7
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: NRC Requests for Additional Information to PWR Licensees Regarding Responses to Generic Letter 2004-02

Dear Dr. Sheron:

The Office of Nuclear Reactor Regulation has recently issued requests for additional information (RAI) to each PWR licensee regarding their responses to Generic Letter 2004-02. These RAIs cover a broad range of topics and are intended to support the NRC staff's ongoing review of industry activities to resolve Generic Safety Issue 191. Responses to the RAIs were requested within 60 days of the date of the letter transmitting the information requests.

The timing of the information requests impacts actions currently underway by PWR plants. These impacts are discussed below. The purpose of this letter is to outline an alternative set of actions and schedule that will minimize the impacts while continuing to support the NRC staff review efforts. I request your consideration and acceptance of the alternative set of actions in lieu of PWR licensee responses to the RAIs within 60 days as requested by the NRC staff.

Impacts of Information Requests on Licensee Activities

PWR licensees are fully engaged in activities necessary to resolve GSI-191 on the schedule established by GL 2004-02. As presented during the February 9, 2006 public meeting on GSI-191, schedules call for 35 of the 69 PWR plants in the U.S. to have all necessary strainer modifications installed by the end of 2006. An additional 33 PWR plants plan to install strainer modifications during scheduled outages in 2007. The activities and work necessary to accomplish this are requiring a significant level of dedicated manpower by both the PWR licensees and their contractors. These activities include preparation and review of design packages, plant specific testing, preparation and submittal of license amendment requests where needed, and implementation of design modifications. Normal schedules for finalization of design packages in advance of outages have, in many instances, been compressed in order to accommodate the desire to install strainer modifications as quickly as possible.

The effort necessary to prepare responses to the information requests at this time will divert resources and attention from the plant strainer modification efforts and jeopardizes current strainer modification schedules.

Ability to Provide Complete Responses
The RAIs request detailed information on the results of GSI-191 evaluations. In a number of instances the requested level of detail is not currently available due to ongoing industry and plant activities. While plant-specific evaluations have been completed to the level needed to support planned strainer modifications, evaluation efforts are continuing in parallel with strainer modifications to incorporate plant-specific test results and to resolve areas of uncertainty. The evaluations will be finalized following planned outages to reflect as-built strainer configurations and other design and operational changes.

Alternative Set of Actions and Schedule
Because of the impacts on ongoing industry actions and schedules presented by the plant-specific RAIs, we propose the following actions.

In lieu of responding to the plant-specific RAI letters, licensees will provide a supplement to their GL 2004-02 responses on a schedule defined by the completion of the outage in which strainer modifications are completed. These supplemental responses will update the information previously provided the NRC and will describe the evaluation methodology that was used to address GSI-191 concerns, include evaluation results and will incorporate plant-specific details requested in the RAI letters. The supplemental responses to GL 2004-02 will be provided on the following schedule:

By December 31, 2006	For units that complete their outage to incorporate strainer modifications on or before December 31, 2006.
Within 90 days of outage completion, not to exceed December 31, 2007	For units that complete their outage to incorporate strainer modifications after December 31, 2006 but before December 31, 2007.

Industry Activities to Support Plant Evaluations
Several areas of interest in the information requests (e.g., chemical effects, downstream effects and coatings) are being actively pursued by the WOG, EPRI, NEI and strainer vendors. For areas in which the requested data are independent of plant-specific details, we believe it is appropriate for industry organizations to work directly with NRC staff. Questions and comments can then be efficiently addressed as part of public meetings and may be incorporated as part of plant-specific supplemental responses to GL 2004-02.

Chemical Effects
The WOG has recently completed a comprehensive set of bench-top tests as a follow-up on the NRC-Industry sponsored ICET program. The final report on these tests, WCAP-16530, *Evaluation of Post-Accident Chemical Effects in Containment Sump Fluid to Support GSI-191,* should be provided to NRC staff by March 10, 2006. In addition, the WOG has initiated a research effort to investigate alternative chemical buffer materials for PWRs. This effort is scheduled to be completed by July 2006. Every effort will be made to be responsive to NRC requests for information on the WOG chemical test activities.

The results of the WOG chemical effects tests are currently being incorporated as part of strainer qualification tests by each of the strainer vendor teams supporting licensee strainer replacement activities. These test activities are open to NRC staff and separate meetings between strainer vendors and NRC can be arranged to address specific questions on chemical effects or general questions on strainer qualification activities.

Downstream Effects
The WOG efforts to support industry activities to address downstream effects are documented in WCAP 16406-P, *Evaluation of Downstream Sump Debris Effects in Support of GSI-191.* This document was provided to NRC staff for information in July 2005. The NRC provided comments on this WCAP in a letter to WOG dated October 27, 2005. In a letter to NRC dated February 27, 2006, the WOG requested NRC review of the WCAP. The letter identifies that the WOG is preparing responses to staff comments on the report.

Coatings
NRC staff concerns regarding the treatment of qualified coatings within GSI-191 resolution activities were identified in a letter to NEI dated January 16, 2006. NEI is working with EPRI and the Nuclear Utilities Coating Council (NUCC) to address the concerns. A response to the letter will be provided by March 31, 2006. We would then be prepared to meet with NRC staff to discuss specific coatings concerns as well as broader industry coatings initiatives.

In order to maintain an open dialogue on the above issues and other areas of interest, we support the establishment of periodic meetings between Industry and NRC on GSI-191 resolution activities. These meetings would provide a necessary forum for addressing known, as well as new, areas of interest.

We respectively request your consideration and acceptance of the alternative set of actions and schedule outlined in this letter at your earliest convenience. A timely response to this request is needed to ensure that licensee activities to resolve GSI-191 are properly directed and to ensure that strainer modification schedules are maintained.

Please contact me should you have any questions.

Sincerely,

Anthony R. Pietrangelo

Anthony R. Pietrangelo

c: Mr. Tom Martin, U.S. Nuclear Regulatory Commission

March 3, 2006

Mr. Anthony R. Pietrangelo
Nuclear Energy Institute
1776 I Street, NW , Suite 400
Washington DC, 20006-3708

SUBJECT: NUCLEAR REGULATORY COMMISSION REQUEST FOR ADDITIONAL INFORMATION TO PRESSURIZED WATER REACTOR LICENSEES REGARDING RESPONSES TO GENERIC LETTER 2004-02.

Dear Mr. Pietrangelo:

This letter responds to your letter of February 28, 2006, on the same subject. As you are aware, the Nuclear Regulatory Commission (NRC) staff has reviewed pressurized water reactor (PWR) licensees' responses to Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors." The staff has provided requests for additional information (RAIs) to individual licensees to support the staff's disposition and eventual closure of Generic Safety Issue (GSI)-191, which relates to PWR sump performance. The staff requested in its correspondence that licensees respond and provide the requested information within 60 days. Your letter proposes an alternative approach and timetable for the staff to obtain this information.

The NRC staff has reviewed your letter, and we find your proposed alternative approach acceptable with the clarification of the time-frames provided in the following paragraph. We are agreeing to your proposed alternative because it is consistent with the staff's emphasis on early installation of modifications to address GSI-191. We understand the burden created by the design and installation of modified strainers within the schedule provided by GL 2004-02. In addition, we recognize that much of the information needed to address the RAIs will not be available until ongoing testing activities are completed. We appreciate your commitment to continue to work with NRC staff to resolve ongoing technical issues. Significant progress can be made through staff interaction with industry groups and sump screen vendors such that the staff may gain confidence that licensees are adequately addressing the remaining technical issues related to GSI-191.

We understand from your letter that for units completing their outage to incorporate strainer modifications in 2006, information needed to fully address GL 2004-02 will be provided to the staff by December 31, 2006. For units installing strainers after 2006, information needed to fully address GL 2004-02 will be provided to the staff within 90 days of outage completion but not later than December 31, 2007. The staff encourages licensees to provide the information at the earliest feasible date to provide for early resolution of any issues with the responses. Licensees who provide information later may find greater difficulty in reconciling any concerns the staff may have with the plans for modifying their strainers, including the need for additional plant modification.

In your letter, you state that in lieu of responding to the plant-specific RAI letters, licensees will provide a supplement to their GL 2004-02 responses. The letter also states that these supplemental responses will update the information previously provided to the NRC and will describe the evaluation methodology that was used to address GSI-191 concerns, will include evaluation results, and will incorporate plant-specific details requested in the RAI letters. We expect these supplemental responses will fully address the issues identified in the RAIs. The staff will review these supplements and determine whether they provide adequate information to support issue closure. Should that be the case, we will consider the RAIs to have been addressed. If additional information is still needed, we will request it at that time.

Please also note that licensees requesting license amendments in support of their strainer modifications will need to provide sufficient information for the staff to determine the acceptability of the proposed amendment. Some licensees will also need to provide information to support staff audits of adequacy of the design bases for their modifications to address GSI-191. Such information may be of the same nature as that requested in the GL 2004-02 RAIs.

Your letter indicates you support the establishment of periodic meetings between Industry and NRC on GSI-191 resolution activities. We agree these meetings would provide a necessary forum for addressing known, as well as emerging areas of interest. The staff supports specific technical meetings, particularly those focused on resolving issues associated with the GL 2004-02 RAIs, to help ensure licensees are on a path to successful issue resolution. Your letter also identified several areas of interest (e.g., chemical effects, downstream effects, and coatings) where there is ongoing work by the industry and suggested it would be appropriate for industry organizations to work directly with NRC staff. We agree that this approach provides an initial method for the industry to explore staff questions and comments. The following provides staff's current high level issues in these areas. NRC expects industry will work with the staff regarding these technical issues and ensure an appropriate basis is incorporated into the plant specific evaluations that will be provided in the supplemental responses to GL 2004-02.

Chemical Effects

- The particle generator model from the WOG chemical effects testing should be sufficiently validated over the range of debris materials and containment pool conditions projected for the PWR plants.

- Plant specific testing and analysis should provide a technical basis for evaluation of chemical effects in areas not addressed by NRC and general industry test programs.

- A sound technical basis should be established for the use of "chemical surrogates" or chemical products to be used in plant specific testing by screen vendors.

- Justification should be provided for chemical effects testing performed in an environment (e.g., tap water) not representative of postulated plant specific post-LOCA containment pool.

- Uncertainties associated with chemical effects should be assessed sufficiently and accounted for with design margin.

Coatings

- Justification should be provided to show how coatings zone of influence (ZOI) test conditions simulate or correlate to actual plant conditions and ensure representative or conservative treatment in the amounts of coating debris generated by the interaction of coatings and a two-phase jet.

- Justification should be provided to show how coating assessment techniques demonstrate that qualified/acceptable coatings remain in compliance with plant licensing requirements for DBA performance.

- Exceptions taken to the staff approved methodology for sizing coating debris for transport analyses should be adequately justified.

Downstream Effects

- Debris interactions and potential concerns with chemical products should be addressed for their effects on downstream components, such as heat transfer surfaces.

- References, such as screen vendor test results and evaluations relevant to screen penetration, should be made available for staff review.

- The analytical approach described in WCAP-16406-P should be validated by correlation to testing or demonstrated acceptable through an established engineering practice in order to determine the need for additional confirmatory testing or analysis.

- Potential non-conservative assumptions should be validated or margin added to compensate.

If you would like to discuss the contents of this letter further, please contact me at (301) 415-1274.

Sincerely,

/RA/
Brian W. Sheron, Associate Director
for Engineering and Safety Systems
Office of Nuclear Reactor Regulation

March 3, 2006

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Brian W. Sheron, Associate Director
for Engineering and Safety Systems
Office of Nuclear Reactor Regulation

DISTRIBUTION:

DSS r/f	J Grobe, NRR	M Cunningham, RES
J Hannnon, NRR	T Martin, NRR	MScott, NRR

Package No.: ML060610153
Letter No.: 060620050
Ticket No.: 020060047 **NRR-106**

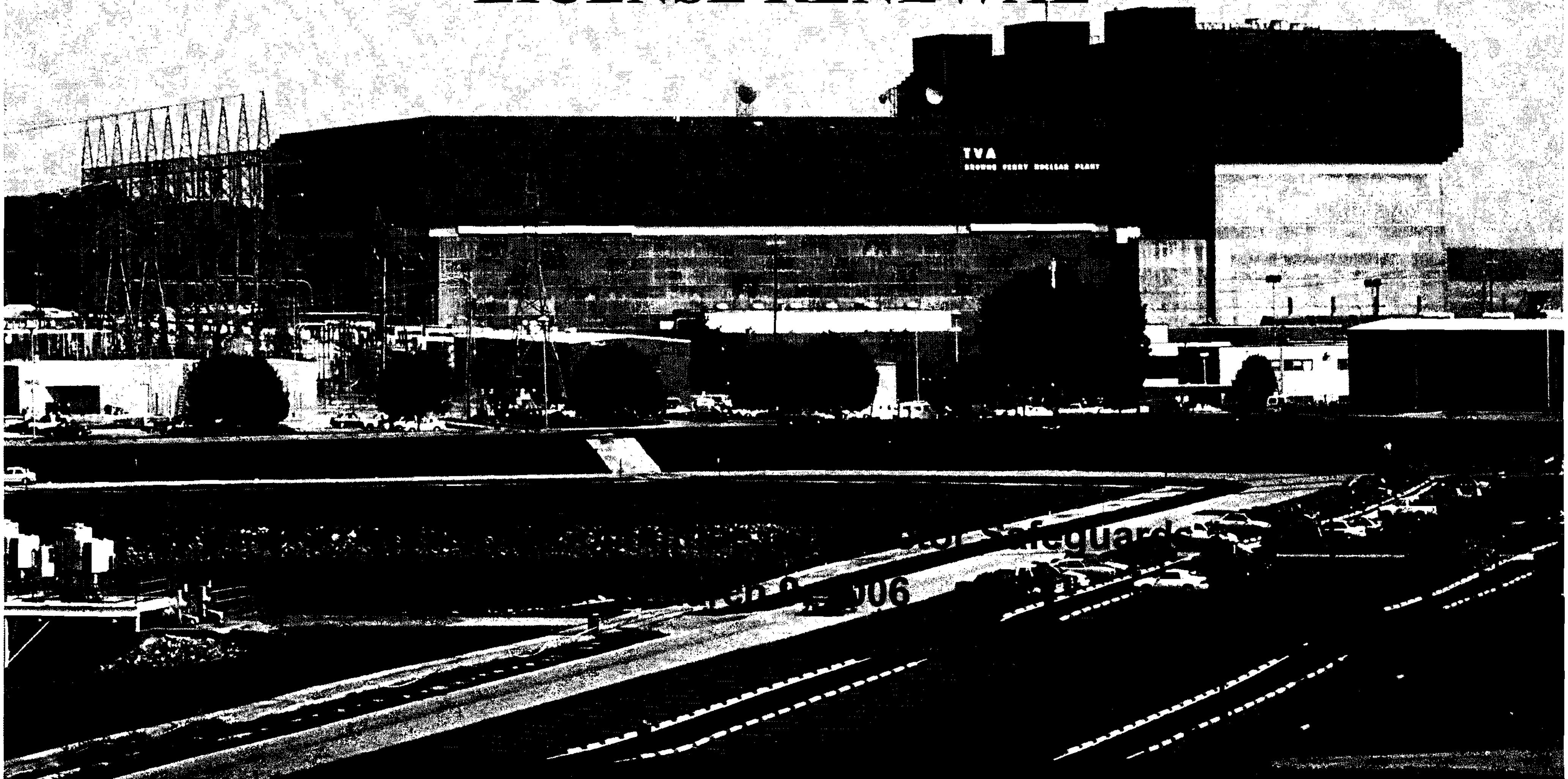
OFFICE	NRR/SSIB	NRR/DCI	NRR/DSS	NRR/ADES
NAME	M Scott	J Grobe	T Martin	B Sheron
DATE	3/3/06	3/3/06	3/3/06	3/3/06

OFFICIAL RECORD COPY

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT

TVA

LICENSE RENEWAL



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Agenda

• Opening Remarks	Masoud Bajestani
• Description of Browns Ferry	Bill Crouch
• BFN License Renewal Application	Bill Crouch
• Unit 1 Major Equipment Replacement / Repair	Joe Valente
• Operating Experience Applicable to Unit 1	Bill Crouch
• Unit 1 Periodic Inspection Program	Joe Valente
• Major Exceptions to GALL Report	Ken Brune
• Corrective Action Program	Rich DeLong
• License Renewal Commitments	Rich DeLong
• Status of AMP Implementation	Rich DeLong
• Unit 1 Maintenance Rule Implementation	Bill Crouch
• Summary	Bill Crouch

Description of Browns Ferry

- All Three BFN Units are General Electric BWR 4 Reactors with Mark I Containments
- Designed and Constructed Materially and Operationally Identical Including Systems, Components, Materials and Environments
- Approximate Years of Operation
  - Unit 1 – 10
  - Unit 2 – 23
  - Unit 3 – 18
- NRC Performance Indicators Green
  - Operating at High Capacity Factor
- Unit 1 on Schedule to Restart in May 2007
- Unit 2/3 Operating at 105% Original Licensed Thermal Power

BFN License Renewal Application

- Three-Unit Application Submitted December 31, 2003
- Original License Expiration
  - Unit 1 – December 20, 2013
  - Unit 2 – June 28, 2014
  - Unit 3 – July 2, 2016
- License Renewal Application at Current Licensed Thermal Power for each Unit (Unit 1 – 3293 MWt, Units 2 and 3 – 3458 MWt)
- Appendix F Describes the Current Licensing Basis Differences Between Unit 1 and Units 2/3
  - These Differences will be Eliminated Prior to Unit 1 Restart (May 2007)
  - Modification and Program Changes in Progress to Eliminate These Differences
  - Current Licensing Basis Same at Restart
- Prepared Using Generic Aging Lessons Learned Report (Rev. 0, 2001)

Unit 1 Major Equipment Replacement / Repair

- Reasons for Replacement / Repair – Examples Provided Below
  - Fidelity with Units 2/3 and Reliability
    - Recirculation Pump Variable Frequency Drives
    - Install Digital Feedwater Control System
    - New Drywell Coolers
    - RHR Heat Exchanger Floating Head
  - Regulatory Issues (Nuclear Performance Plan, GLs and Bulletins)
    - Replace piping subject to Intergrannular Stress Corrosion Cracking
    - Drywell structural steel and electrical penetrations
    - Environmental Qualification
  - Dose Reduction
    - Replace valves due to stellite content

Unit 1 Major Equipment Replacement / Repair

- Reasons for Replacement / Repair
  - Maintenance Reduction
    - Large pump and motor refurbishment
    - Turbine refurbishment
    - Valve replacement / refurbishment
  - Lessons Learned from Unit 3 Layup and Recovery
    - Residual Heat Removal Service Water Piping Replacement in the Reactor Building
    - Extraction Steam Piping (FAC) Replacement
    - Raw Cooling Water Piping Replacement
  - Extended Power Uprate
    - Feedwater Pump and Turbine Modifications
    - Additional Condensate Demineralizer

Unit 2/3 Operating Experience Applicable to Unit 1

- Identical GE BWR4 Reactors with Mark I Containments
- Designed and Constructed Materially and Operationally Identical Including Systems, Components, Materials and Environments
- Unit 3 Shutdown for 10 Years
  - All Units Used Same Layup Philosophy, Processes and Conditions
    - Aging Effects Monitored and Addressed Prior to Unit 3 Restart
    - No Layup Induced Aging Effects During 10 Years of Ensuing Operation
  - Extensive Layup Experience from Unit 3 Directly Applicable to Unit 1
    - Other than Duration, Same Effects
- Anticipated Piping Replacements as a Result of Layup Experience from Unit 3 Incorporated into Unit 1 Recovery Plan
  - RHR Service Water Piping Replacement (A and C Loops)
  - Raw Cooling Water Small Bore Piping

Unit 2/3 Operating Experience Applicable to Unit 1

- Planned Replacement of IGSCC Susceptible Piping in Reactor Recirculation, Residual Heat Removal, Reactor Water Cleanup and Core Spray Systems
- Did not Credit Unit 1 Layup Program as Sole Means to Establish Acceptability of Piping and Components for Restart or License Renewal
  - Visual and UT Inspections Performed to Establish Condition
  - Piping and Component Replacements
- Implementing Same Restart Programs and Modifications as were Completed on Unit 2 and 3
- Implementing Same Aging Management Programs for Duration of Original License Period and Period of Extended Operation
- Compensatory Periodic Inspection of Unit 1 Non-Replaced Piping

Unit 1 Periodic Inspection Program

- Periodic Inspections will be Performed to Verify No Latent Aging Effects are Occurring in Non-Replaced Piping
- Supplements Other Aging Management Programs
- Baseline Inspections Before Restart
- 95/95 Confidence Level Samples for each Group in Accordance with NUREG 1475
- Samples Grouped by Common Material Types and Environments
  - Stainless Steel/Treated Water
  - Stainless Steel/Raw Water
  - Carbon Steel/Treated Water
  - Carbon Steel/Raw Water
  - Carbon Steel/Treated Water Closed Cooling Water System

Unit 1 Periodic Inspection Program

- Sample Points
  - From Non-Replaced Piping In-Scope for License Renewal
  - From Piping not in Operation During Unit 1 Lay-up
  - Includes Areas Where There is Potential for Degradation as well as Areas Where Degradation is not Expected
- First Round of Periodic Inspections will be Performed After Several Years of Unit 1 Operation and Prior to Period of Extended Operation
- Additional Inspections will be Performed during the Period of Extended Operation Prior to Completion of 10 Years of Extended Operation
- Subsequent Inspection Frequency will be Determined Based on Inspection Results

Unit 1 Periodic Inspection Program

- The Periodic Inspection Sample Locations will be a Subset of Non-Replaced Piping Locations in
  - Residual Heat Removal Service Water (A and C loops)
  - Fire Protection
  - Emergency Equipment Cooling Water
  - Raw Cooling Water
  - Control Rod Drive
  - Core Spray
  - Feedwater
  - High Pressure Core Injection
  - Main Steam
  - Reactor Core Isolation Cooling
  - Residual Heat Removal
  - Reactor Building Closed Cooling Water
  - Turbine Drains and Miscellaneous Piping
  - Radiation Monitoring
  - Radwaste
  - Containment Inerting
  - Reactor Water Cleanup
  - Rector Recirculation
  - Containment
  - Standby Liquid Control
  - Sampling and Water Quality
  - Gland Seal
  - Reactor Vessel Vents and Drains
  - Heater Drains and Vents
  - Condensate and Demineralized Water

Major Exceptions to GALL Report

- No Major Exceptions to Generic Aging Lessons Learned Report
- 39 Aging Management Programs
- 8 Aging Management Programs Have Taken Minor Exceptions to GALL
- Each Aging Management Program is Adequate to Manage the Aging Effects for Which it is Credited.

Exceptions to GALL Report

- Aging Management Programs with Exceptions to GALL
  - Electrical Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements used in Instrumentation Circuits Program
    - LPRM cables use calibration results of surveillance program
  - Chemistry Control Program
    - Used updated EPRI guidelines for water chemistry
  - Bolting Integrity Program
    - Other AMPs were used for some bolting
  - Inspection of Overhead Heavy Load and Light Load Handling Systems Program
    - Crane fatigue was addressed by TLAA analysis
  - Fire Protection Program
    - CLB requirements used for inspection and testing
  - Fire Water System Program
    - CLB requirements used for inspection and testing
  - Fuel Oil Chemistry Program
    - Different industry standard used
  - ASME Section XI Subsection IWE Program
    - Some inspection and testing requirements based on approved relief requests

Corrective Action Program

- License Renewal Commitments Tracked with Onsite Commitment Tracking System and Corrective Action Program
- TVA Corrective Action Program Applies to all TVA Units
- Requires All Personnel to Promptly Document and Report Problems and Adverse Conditions for Evaluation and Corrective Action
- Ensures Immediate Action, Operability Evaluation, Reportability Determination, Determination of Severity for Root Cause and Extent of Condition (if required), Management Review, Evaluation, Corrective Action Tracking and Trending
- Condition Identified on any BFN Unit Reviewed for Generic Implications to Other Units and Other TVA Sites
- Internal and External Plant Operating Experience Incorporated into Corrective Action Program

License Renewal Commitments

- Commitments made Through Application and Requests for Additional Information
  - 110 Commitments made to Date
  - Revise Existing Aging Management Programs to Include License Renewal References
  - Enhance Existing Aging Management Programs
  - Implement New Aging Management Programs
  - Completion of Open Items from Draft SER
  - Unit 1 Specific - Appendix F Current Licensing Basis Differences Between Unit 1 and Units 2 and 3
- License Renewal Commitments Tracked Through Onsite Commitment Tracking System and Corrective Action Program

Status of AMP Implementation

- 39 Aging Management Programs Total
  - 11 Existing Aging Management Programs Revised Only to Include Unit 1
    - Complete Revisions in 2006
  - 11 Existing Aging Management Programs Requiring No Enhancement
    - Complete Revisions in 2007
  - 11 Existing Aging Management Programs Require Enhancement for all Units
    - Complete Revisions in 2008
  - 6 New Aging Management Programs
    - Develop by 2009
- Aging Management Program Implementation Packages Have Been Developed for All 39 Programs
- Implement Unit 1 Periodic Inspection Program Prior to Restart

Unit 1 Maintenance Rule Implementation

- Underlying Purpose of Maintenance Rule is to Ensure SSCs are Maintained so that they will Perform their Intended Function when Required
- Because of Defueled Condition Most Unit 1 Systems do not Perform Functions Required to be Monitored by Rule
- Because of Layup Status Most Unit 1 Systems cannot Perform Functions
- Unit 1 Systems that Perform Required Function in Defueled Status or Support U2/3 Operation are Operated and Maintained under Applicable Technical Specifications and Included in Maintenance Rule Program
- Temporary Exemption Created to Resolve Issue Raised in 1997 NRC Initial Maintenance Rule Inspection – Eliminated When System Required to be Operable by Technical Specifications

Summary

- Three-Unit Application at Current Licensed Thermal Power
- Prepared using Generic Aging Lessons Learned Report
- Unit 2/3 Operating Experience Applicable to Unit 1
- Unit 1 Periodic Inspection Program for Non-Replaced Piping to Verify No Latent Aging Effects are Occurring as a Result of Layup Duration
- Aging Management Programs Established to Manage the Effects of Aging so that BFN can be Operated Safely in Accordance with Current Licensing Basis for Period of Extended Operation
- License Renewal Commitments Tracked Through Onsite Commitment Tracking System and Corrective Action Program

Browns Ferry Nuclear Plant
Units 1, 2, and 3
License Renewal
Safety Evaluation Report

Staff Presentation to the ACRS Full Committee

Ram Subbaratnam, and
Yoira Diaz Sanabria, Project Managers
Office of Nuclear Reactor Regulation

March 9, 2006
3:15 - 5:15 PM (EST)

Review Highlights

- **License Extension Request – December 31, 2003**
  - Unit 1: December 20, 2013
  - Unit 2: June 28, 2014
  - Unit 3: July 2, 2016

- **SER with Open and Confirmatory Items issued on August 9, 2005**

- **Final SER issued on January 12, 2006**
  - Two (2) CIs and Four (4) OIs were resolved
  - March 6, 2006 letter - Applicant certified CLB differences in Unit 1 satisfied 10 CFR 50.59 criteria and ready for audit (TI 2509/001)
  - Supplemental SER will provide details/clarifications on Unit 1 Periodic Inspection Program and resolution of OI for Drywell Shell Corrosion

- **Open Items (OIs)**
  - **Two (2) OIs (Closed)**
    - Time-limited aging analysis: OI 4.7.7
    - Unit 1 Periodic Inspection: OI 3.0-3 LP (lay up)
  - **One (1) OI AMP Inspection**
    - RHRSW piping

- **Open Item 2.4-3: Drywell Shell Remains Unresolved**

ACRS Interim Report Letter

Highlights

- Interim Report Letter – October 19, 2005
- Response to Letter – November 28, 2005
- Four major recommendations
  - The final SER included:
    - Resolution of four OIs
    - Discussion of Units 2 and 3 operating experience and its applicability to Unit 1
    - Description of Unit 1 Periodic Inspection Program attributes
    - Evaluation of operating experience at up-rated power level that incorporates lessons learned into the AMP prior to the PEO

SER – OI 4.7.7 (Closed)

OI 4.7.7: Stress Relaxation Core Plate Hold-Down Bolts

- Applicant committed to perform plant specific analysis per BWRVIP-25

- Analysis will be submitted for staff's review and approval two years prior to entering the PEO

SER – OI 3.0-3 LP (Closed)

OI 3.0-3 LP (lay up): Unit 1 Periodic Inspection Program

- Staff requested and evaluated program that was included in final SER Section 3.0.3.3.5
  - Plant specific program to monitor latent aging effects of left in place / lay up components in Unit 1
  - Assures level of safety of Unit 1 left in place / lay up components equivalent from those components in Units 2 and 3
  - Staff's reviewed subsequent sampling methodology to confirm consistency with NUREG-1475
- The program will be fully developed and implemented prior to Unit 1 restart

RHRSW Piping Confirmatory Item

Inspection report – November 7, 2005

- RHRSW suction side: Three 24-inch diameter cast iron pipes, cast into concrete of the intake structure, have never been inspected
- On February 14, 2006 letter, the applicant committed to perform one-time inspection by using a remote method before entering the PEO
- Staff asked the applicant to confirm no blockage path through pipes by using the buried piping inspection program and tanks as recommended by GALL
- Applicant agrees with the staff and will provide this as a commitment
- Pending on formal submittal this is a confirmatory item
- No additional safety issues were identified, therefore aging management inspection is closed as documented in letter dated March 1, 2006.

SER – OI 2.4-3 (Open)

- Earlier, the applicant indicated that no significant degradation observed in normally in-accessible areas of the drywell
- Staff accepted a one-time UT inspection based on the understanding that the degradation is insignificant
- Inspection will be done prior to restart for Unit 1 and before entering the PEO for Units 2 and 3

SER – OI 2.4-3 (Continued)

On March 9, 2006 new information verbally provided by applicant on Unit 1 drywell UT data

- Found a small localized area of wall thinning
- Applicant is evaluating issue to provide impact on drywell integrity for all three units
- Evaluation will be provided to the staff for review
- Staff evaluation will be documented in Supplemental SER
- Therefore, OI 2.4-3 remains open

ACRS Interim Report Letter

Operating Experience Applicability

- Applicant claims:
  - Unit 1 Environment was maintained consistent with those of Units 2 and 3
  - Unit 1 experienced the same aging mechanisms and rates
  - Water chemistry within Unit 1 piping systems maintained in Service met operating purity requirements
  - Effective portions of certain systems in areas where OE from U 2 & 3 showed adverse effects from uncontrolled lay up were replaced for all three units
- Staff questions the applicant's statement of OE applicability
- Unit 1 Periodic Inspection Program will be an acceptable mitigating action for the lack of applicable OE

ACRS Interim Report Letter

AMR and AMP evaluated at EPU level

- Applicant to evaluate BFN operating experience at the up-rated power level and incorporate lessons learned into their aging management programs for the PEO

- Applicant committed to implement operating experience and aging management program reviews before entering PEO

Conclusion

- On the basis of its evaluation of the license renewal application, the NRC staff concluded that the requirements of 10 CFR 54.29(a) have been met, pending resolution of OI 2.4-3.

Regulatory Guide 1.97, Revision 4
"Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"

Advisory Committee on Reactor Safeguards Meeting
March 10, 2006

George Tartal, I&C Engineer

Instrumentation and Electrical Engineering Branch

Division of Fuel, Engineering and Radiological Research

Office of Nuclear Regulatory Research

gmt1@nrc.gov 301-415-0016

(2)

OVERVIEW

- BACKGROUND
- REGULATORY GUIDE 1.97, REVISION 3
- IEEE STANDARD 497-2002
- REGULATORY GUIDE 1.97, REVISION 4
- PUBLIC COMMENTS AND STAFF RESPONSES
- CONCLUSION

BACKGROUND

- Instrumentation required to monitor variables and systems under accident conditions
  - 10 CFR Part 50, Appendix A, Criteria 13, 19, 64
- Reg Guide 1.97 Rev. 1 issued in August 1977
- Lessons learned from TMI
  - NUREG-0737
  - 10 CFR Part 50.34(f)
- Reg Guide 1.97 Rev. 2 issued in December 1980
  - Endorsed ANSI/ANS-4.5-1980
  - Implemented via NUREG-0737 Supp. 1
- Reg Guide 1.97 Rev. 3 issued in May 1983
  - Endorses ANSI/ANS-4.5-1980 (withdrawn and inactive)

REGULATORY GUIDE 1.97, REV. 3

- Each accident monitoring variable is assigned a variable type and a category
  - Variable type is selected based on function
  - Category is selected based on required quality level
- Organizes accident monitoring variables by variable type
  - Type A are for planned manual actions with no automatic control
  - Type B are for assessing plant critical safety functions
  - Type C are for indicating breach of fission product barriers
  - Type D are for indicating safety system performance and status
  - Type E are for monitoring radiation levels, releases and environs
- Design and qualification criteria applied by category
  - Cat 1 is for indicating accomplishment of safety function (~SR)
  - Cat 2 is for indicating safety system status (~AQ)
  - Cat 3 is for backup and diagnostic variables (~NSR)

IEEE STANDARD 497-2002

- Consolidates and updates criteria from ANSI/ANS-4.5-1980, IEEE Std 497-1981 and Reg Guide 1.97 Rev. 3
- Technology-neutral approach intended for advanced design plants
- Performance-based, non-prescriptive approach to accident monitoring variable selection
  - Prescriptive tables of variables are replaced by criteria for selection based on the accident mitigation functions
  - This is the most significant difference from Reg Guide 1.97 Rev. 3
- Selected variable type determines the applicable performance, design, qualification, display and QA criteria
- Categories are no longer used

IEEE STANDARD 497-2002 CRITERIA

- Selection
  - Defines variable types A, B, C, D and E and lists typical sources
- Performance
  - Range; Accuracy; Response Time; Duration; Reliability
- Design
  - Single & Common Cause Failure; Independence; Separation; Isolation; Power Supply; Calibration; Portable Instruments
- Qualification
  - Environmental; Seismic
- Display
  - Characteristics; Identification; Display Types; Recording
- Quality Assurance

REGULATORY GUIDE 1.97, REV. 4

- Responds to User Need Request NRR-2002-017
- Endorses IEEE Standard 497-2002 with exceptions and clarifications
- Intended for new nuclear power plants
- Conversion to the new criteria by current operating plants may be done on a comprehensive, voluntary basis
- Issued for public comment as DG-1128 in August 2005
- Eight regulatory positions

REGULATORY POSITIONS

1. How might current operating plants using Rev. 2 or 3 of Reg Guide 1.97 convert to IEEE Std 497-2002 criteria?
   - "The guidance provided in this standard may prove useful for operating nuclear power stations desiring to perform design modifications or design basis modifications."
   - Some interest in applying Rev. 4 to current plants
   - IEEE Std 497-2002 provides no guidance in translating from type and category to type only
   - <u>Generally</u>: Type A,B,C = Cat 1, Type D = Cat 2, Type E = Cat 3
   - New criteria may be more or less stringent than existing criteria
   - The staff recommends conversion to be comprehensive and is strictly voluntary
   - Partial conversion could result in an incomplete analysis and is not endorsed

REGULATORY POSITIONS (cont.)

2. Calibration during an accident
   - IEEE Std 497-2002 requires maintaining instrument calibration by means of recalibration, interval specification, equipment selection or cross-calibration
   - Of these means, only recalibration can satisfy the requirement
   - Modifies IEEE requirement to <u>validating</u> instrument calibration instead of <u>maintaining</u> instrument calibration
3. Severe accidents
   - IEEE Std 497-2002 does not directly address severe accidents
   - IEEE Std 497-2002 requires Type C variables to have extended ranges
   - Clarifies the need for extended ranges based on current regulatory requirements

REGULATORY POSITIONS (cont.)

4. Contingency actions
   - "Alternative actions taken to address unexpected responses of the plant or conditions beyond its licensing basis"
   - IEEE Std 497-2002 excludes all contingency actions from the scope of potential Type A variables
   - Applied as if all contingency actions are to mitigate accident conditions beyond the licensing basis of the plant
   - Recommends considering all operator actions within the licensing basis during the selection process
5. Number of points of measurement
   - IEEE Std 497-2002 does not address this topic, but was addressed in RG 1.97 Rev. 3
   - Recommends the number of points of measurement be sufficient to adequately indicate the variable value

REGULATORY POSITIONS (cont.)

6. Codes and standards referenced
   - Guidance is provided for references codified in regulations, endorsed in regulatory guides, or neither codified nor endorsed
7. Type C variable operating time
   - IEEE Std 497-2002 requires at least 100 days operating time for type C variables
   - Recommends an optional operating time as specified in the plant licensing basis
8. Replace “post-event operating time” with “operating time”
   - The new language is consistent with the title change from “post accident monitoring” to “accident monitoring”
   - Operating time should encompass the full accident duration

PUBLIC COMMENTS AND STAFF RESPONSES

Seven sets of comments received

- NEI
- NUGEQ
- IEEE
- BWROG
- Westinghouse
- TVA
- Exelon

Each comment has been addressed and responses made publicly available in ADAMS

ADAMS accession number ML053640161

Only significant comments will be highlighted

PUBLIC COMMENTS AND STAFF RESPONSES

RP#1: Voluntary conversion to Rev. 4 for current plants

- Should recognize acceptability of plant's current licensing basis
- Unnecessarily restrictive requirement to convert the entire plant's accident monitoring system to Rev. 4
- "Not intended for current operating reactor licensees" language is confusing
- Should provide guidance for performing digital upgrades

→ RP#1 revised to clarify it is intended for new plants

RP#2: Calibration during an accident

- Not clear that requirements are relaxed
- Only during post-event operating time
- Change "maximum extent" to "extent practical"

→ RP#2 revised "maintain" calibration to "validate" calibration

PUBLIC COMMENTS AND STAFF RESPONSES

RP#3: Type C variable extended range requirements

- Should address IEEE section 5.1 instead of section 4.3
- Should add current alternative source terms

→ RP#3 revised to reference section 5.1

RP#4: Contingency actions

- BWR contingency actions extend beyond design basis
- No limitations to contingency actions considered
- Contingency actions are, by definition, beyond design basis
- Should exclude beyond design basis actions from contingency action criteria

→ RP#4 revised to recommend consideration of contingency actions within the plant's licensing basis

PUBLIC COMMENTS AND STAFF RESPONSES

RP#5: Number of measurement points

- No comments

RP#6: Referenced codes and standards

- Should allow use of codes & standards within current licensing basis

→ RP#6 not revised

RP#7: Type C variable instrument duration

- Should give option for use of licensing basis documents as a source for type C variable instrument duration

→ RP#7 added as a result of this comment

RP#8: Clarification of operating time

- Post-event vs. full accident duration

→ RP#8 was added as a result of this comment

CONCLUSION

- Regulatory Guide 1.97, Rev. 4 endorses current IEEE Standard 497-2002 with exceptions and clarifications
- Public comments have been received and staff responses are publicly available in ADAMS
- Intended for new nuclear plants, with current operating plant conversion on a comprehensive, voluntary basis
- No backfit issues
- Final Comments or Questions?

BRIEFING OF THE ACRS:
EVALUATION OF ACCIDENT SEQUENCE PRECURSOR DATA TO IDENTIFY SIGNIFICANT OPERATING EVENTS

March 10, 2006

Patrick W. Baranowsky, Deputy Director for Operating Experience & Risk Analysis
Douglas W. Weaver, Acting Branch Chief
Gary M. DeMoss, ASP Program

Operating Experience and Generic Issues Branch
Division of Risk Assessment and Special Projects
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

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Outline of Presentation

- Introduction & Background (P. Baranowsky)
  - Purpose
  - Highlights

- Program Status (G. DeMoss)
  - Progress of analyses
  - Recent events

- ASP Trends & Insights (G. DeMoss)

- Summary (G. DeMoss)

Purpose of the Presentation

- To provide a brief overview of the status of the ASP Program
- To describe our analysis of trends in ASP-analyzed events

ASP Program Background

ASP has been a part of NRC events analysis activities for about 25 years, and it has a variety of internal and external users.

- The primary objective of the ASP Program is to systematically evaluate operating experience to identify and document events likely to lead to core damage. Analyses are performed to define and project potential accident scenarios, determine risk exposure, and assess risk mitigation measures.

- ASP analyses are used to support:
  - Performance measures in the Annual Performance and Accountability Report to Congress
  - Industry trends program
  - Decisions to develop generic communications
  - Studies to determine the safety significance of potential regulatory issues
  - A partial check on PRA scenarios / SPAR models

Highlights of the Presentation

- Analysis of FY 2003 & FY 2004 events are substantially complete and included in the trend analyses

- No significant precursors (conditional core damage probability ≥ 1 x 10^{-3}) in FY 2003, FY 2004 or FY 2005

- No trend was identified in the rates of occurrence of all precursors during the period from FY 1993 through FY 2004

- Trending of precursors by bins yielded mixed results. There is no increasing trend in higher risk precursors (>1 x 10^{-5})

ASP Program Accomplishments

- Final Davis-Besse ASP analysis issued in March 2005
- FY 2004 Precursors essentially completed in November 2005
- Preliminary assessments of all FY 2005 events will be available in Spring 2006
- Investigation of trends and insights completed in SECY-05-0192
- Trial application of expert elicitation methodology issued in Palo Verde analysis – awaiting comments (if any)
- Risk-informed review process implemented in December 2005

Status of ASP Analyses

(as of February 28, 2006)

	FY-01	FY-02	FY-03	FY-04	FY-05
Total precursors identified[a]	22	14	22	17	19
Final precursor analyses completed	17	10	20	14	0
Analyses not yet complete	5[b]	4[b]	2[b]	3	19

a) All of the reviews and analyses have not been completed, and therefore, the number of total precursors for these years may change

b) Events involving cracking of control rod drive mechanism housings have not been completed, and therefore, the number of precursors attributable to cracking of CRDM housings may change

Note: The ASP program screens all LERs and rejects 20 to 50 events per year after performing risk analysis.

Interesting 2004 Analyses

- Palo Verde LOOP (9E-6, 4E-5, 9E-6)
  - Grid LOOP complicated with a breaker failure
  - Unit 2 had an unavailable EDG
- Palo Verde ECCS Piping Voids (1E-5 per unit)
  - SDP conservatively assumed that low pressure recirculation would not work for MLOCA
  - ASP used expert panel approach to create probability distribution for system operability – result consistent with SDP
- St. Lucie LOOP during hurricane Jeannee (1E-5 per unit)
  - Salt Spray on switchyard created uncertain recoverability
  - Full power model was adjusted to credit pre-hurricane shutdown procedures. (i.e., remove relief valve lift, RCP seal LOCA & some short term sequences)
- Calvert Cliffs Trip and Potential Overcooling (5E-5)
  - Reactor trip due to low SG level caused by loss of MFW pump
  - Relay failure causes excessive cooldown
  - SPAR models modified to include over-steam demand sequences

Potentially Interesting FY 2005 Analyses

- Flooding vulnerability
- Single failure vulnerabilities due to metering relays
- Initiating events
  - Trips with complications related to low voltage power, RCIC, leakage & safety valves
  - LOOPs complicated by hurricanes and equipment failure
- S/D events with plants in vulnerable conditions
  - Solid Plant
  - Mid-loop

ASP Results, Trends & Insights

Summarized from SECY-05-0192

- No *significant* precursors were identified in either FY 2003, FY 2004 or FY 2005. Davis-Besse was a significant precursor in FY 2002

- Four precursors identified in FYs 2002–2004 had a CCDP greater than $1x10^{-4}$. Includes Davis-Besse, the potential common mode failure of AFW at Point Beach 1& 2, and another potential common mode failure of AFW at Point Beach 2.

- No trend was identified in the rates of occurrence of all precursors during the period from FY 1993 through FY 2004.

- Trending of precursors by CCDP bins yielded mixed results. If a trend is considered statistically significant, it is very unlikely that the trend is a result of chance alone. Trending analysis of precursors in the CCDP bins yielded the following results:

• CCDP > 1×10^{-3}	No trend
• 1×10^{-3} > CCDP > 1×10^{-4}	Decreasing trend - statistically significant
• 1×10^{-4} > CCDP > 1×10^{-5}	No trend
• 1×10^{-5} > CCDP > 1×10^{-6}	Increasing trend - statistically significant

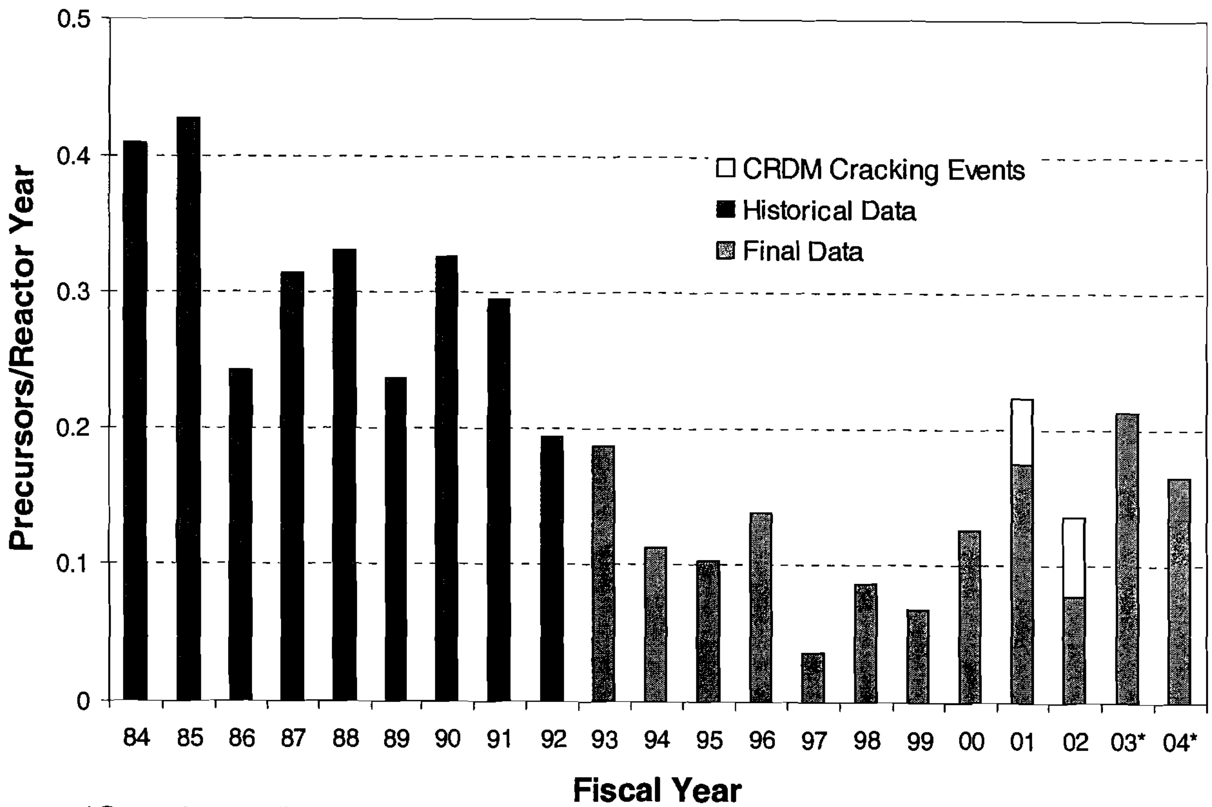
Trending Approach Used in SECY-05-0192

- The SECY-05-0192 contains an expanded trending analysis. We will continue to refine and if necessary expand this section in future SECY papers.
- Uses the p-value approach for determining the probability of observing a trend as a result of chance alone
- A trend is considered statistically significant if the p-value is smaller than 0.05
- Trending starts at 1993 because of the advent of SPAR models.

ASP RESULTS, TRENDS & INSIGHTS

A Historic Perspective

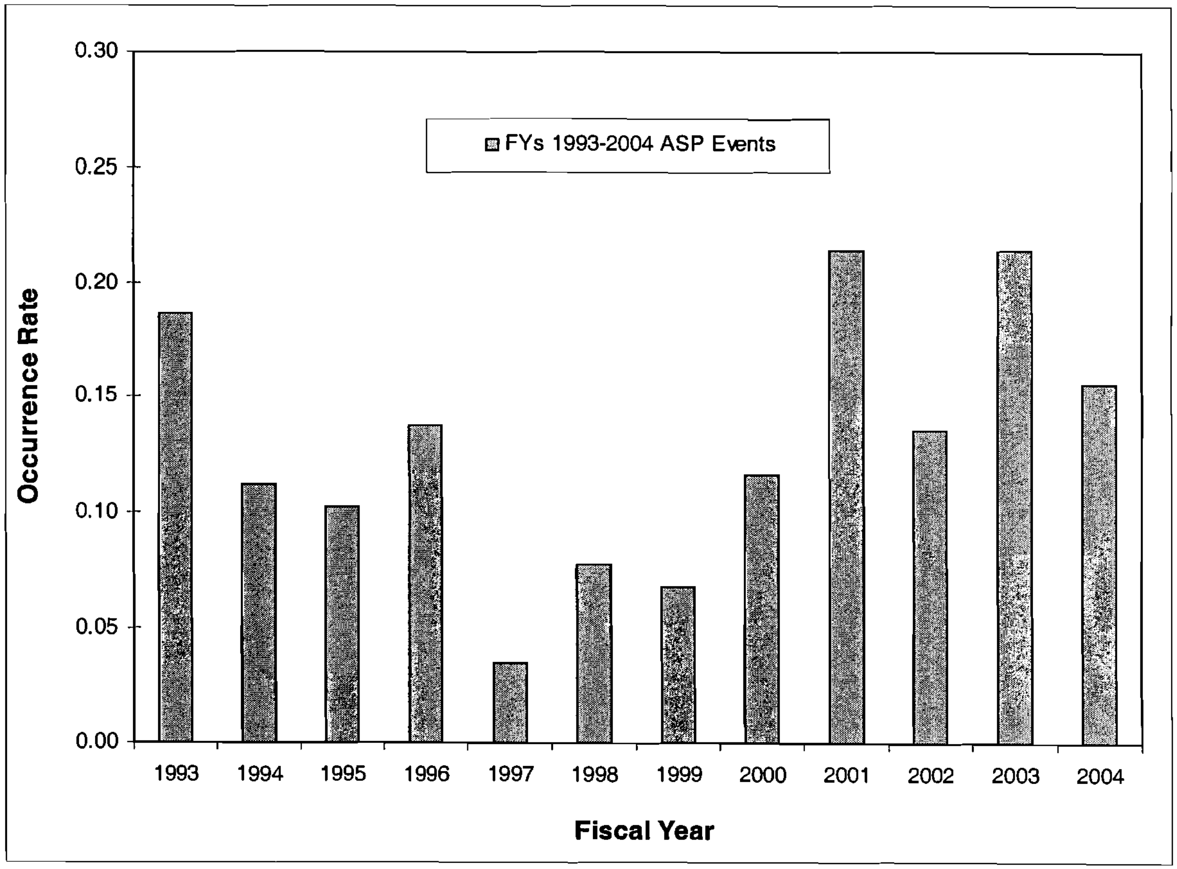
Number of Precursors by Fiscal Year



*Contains preliminary data

ASP RESULTS, TRENDS & INSIGHTS

No trend was identified in the rates of occurrence of all precursors during the period from FY 1993 through FY 2004 (p-value = 0.1016)

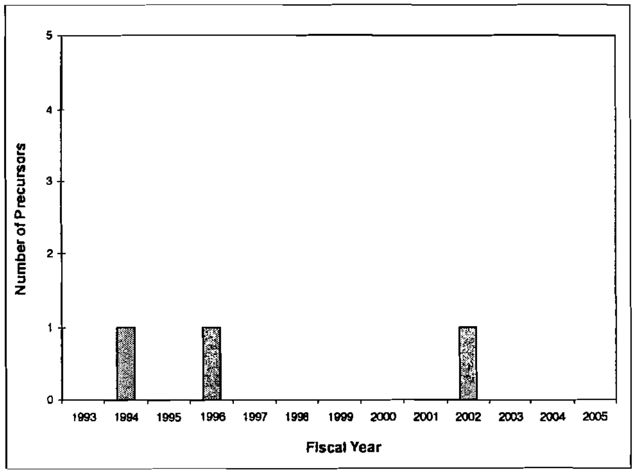


*Years after FY-2001 contain preliminary data

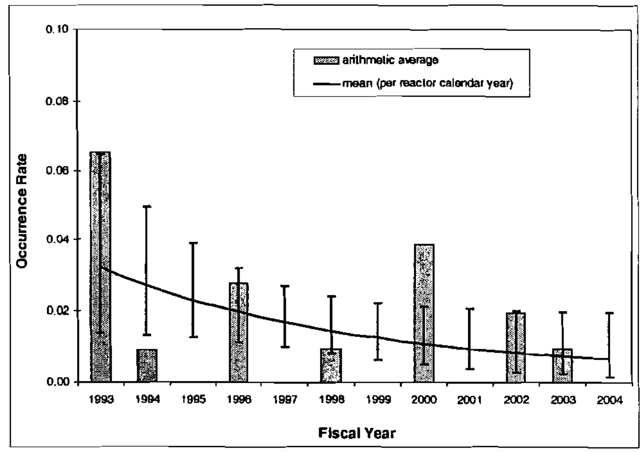
Source: SECY-05-192

ASP Results, Trends & Insights

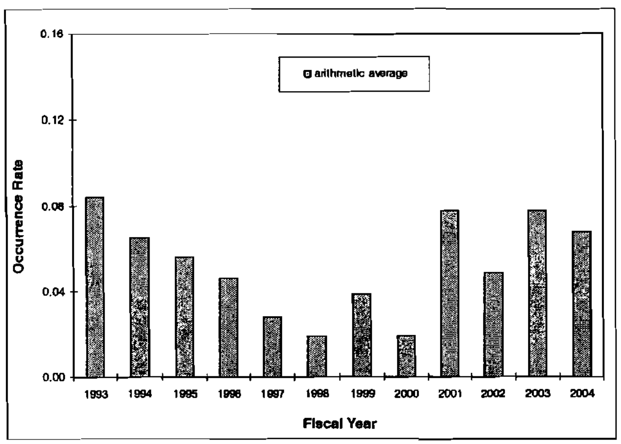
Trending of precursors by CCDP bins



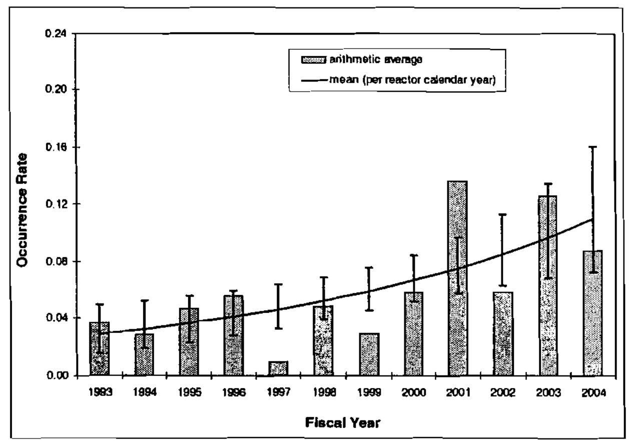
Precursors in CCDP bin 10^{-3}



Precursors in CCDP bin 10^{-4}



Precursors in CCDP bin 10^{-5}



Precursors in CCDP bin 10^{-6}

Source: SECY-05-192

Evaluations of Trends in Precursor Counts

- Trending was done for the FY 1997 – 2004 period and for the FY 2001 – 2004 period

- To ensure consistency of data, post-FY 2001 data was adjusted to reflect changes in event selection criteria. This 'rebaselining' accounts for:
  - Evolving analysis methods and SPAR models allow analysis of complex conditions (i.e., fire, external events, HELB and internal flooding) that were previously screened out.

  - ASP not analyzes all greater than green SDP findings

- Rebaselining removed 23 precursors from the data

Trending Evaluation Conclusions

- Important Precursors (> 1×10^{-4}) – No trend
- FY 1997 – 2004 trend
  - Increase in scope of the ASP program is shown.
  - Increasing trend measured.
  - No trend remains if CRDM or LOOP events are removed.
- FY 2001-2004 (re-baselining not needed) – No trend measured

Additional Trending

- **Initiating Events vs Degraded Conditions**
  - Frequency of degraded conditions is increasing relative to initiating events
  - This would be more pronounced if not for the increase in LOOP precursors

- **LOOP Initiating Events**
  - Statistically significant increasing trend FY 1993 – 2004
  - Would not be statistically significant if not for August 14, 2003 event

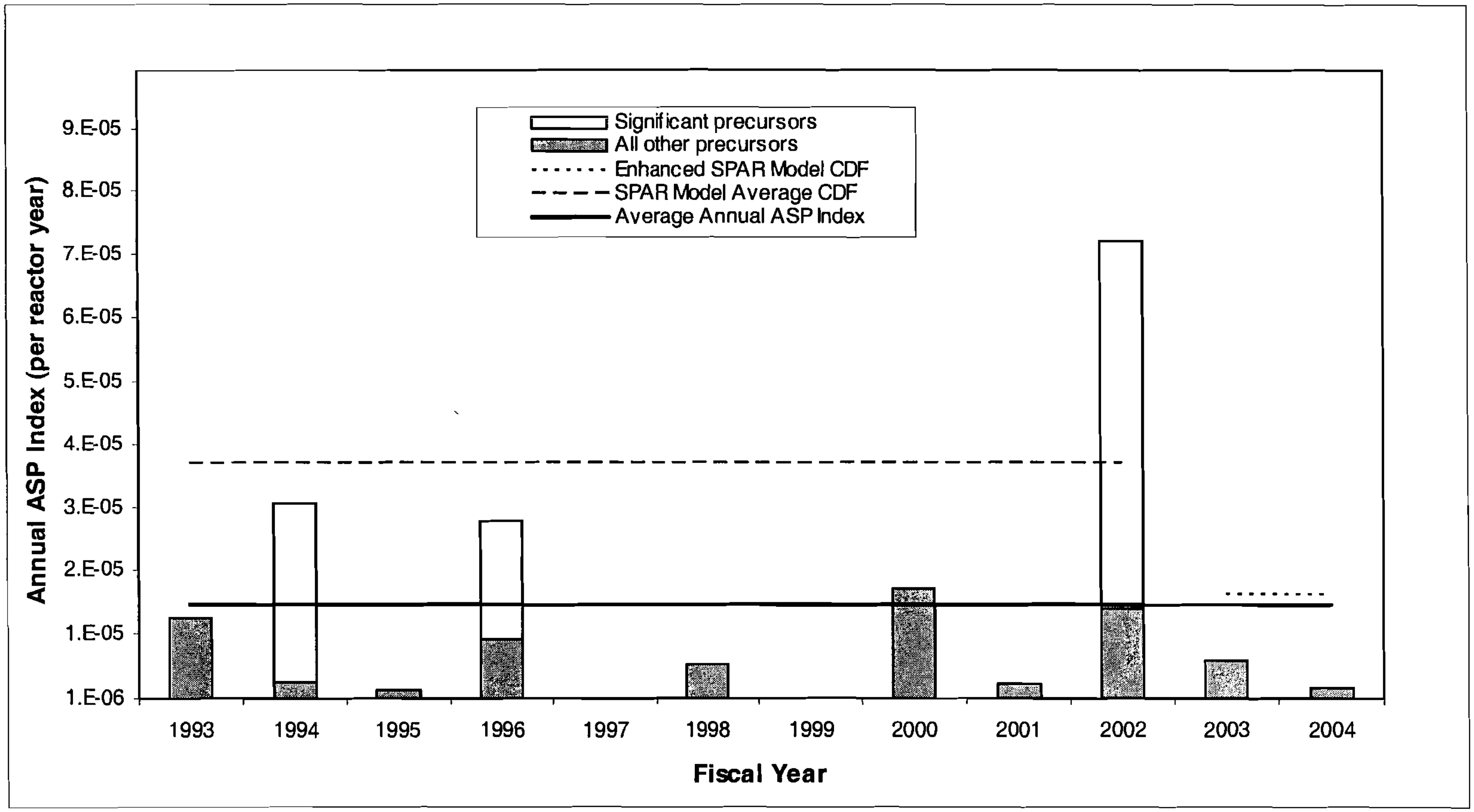
- **Precursors at BWRs vs PWRs**
  - BWR precursors show an increasing trend
  - PWR precursors do not show a trend

Two ASP Indices

- Annual ASP Index
  - Assigns the risk of ASP events to the year in which it occurred
- Integrated ASP Index (new)
  - Assigns the risk of ASP events to the actual duration

Annual ASP Index

(Total CCDP and ΔCDP divided by the number of reactor years)



Results from the Annual ASP Index

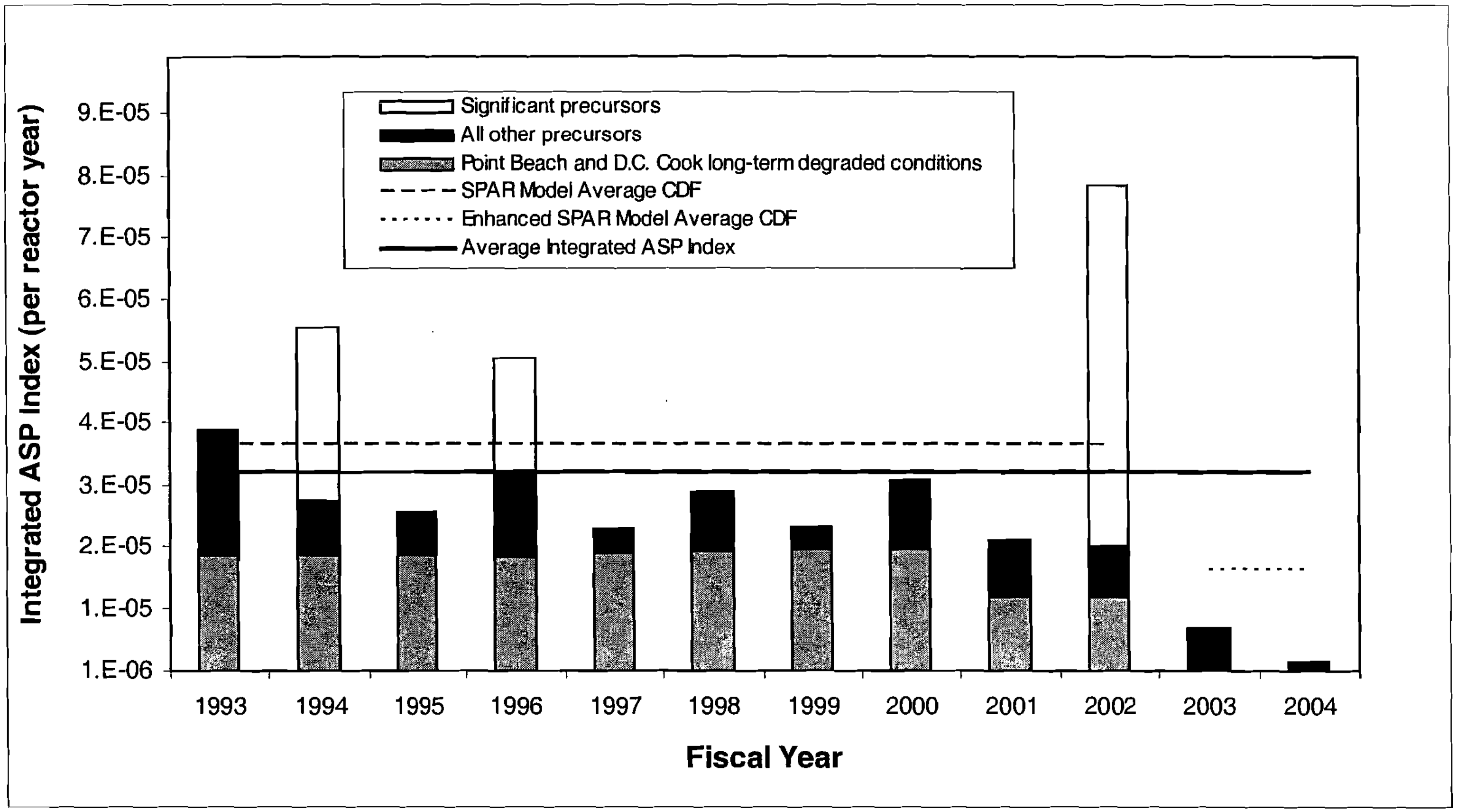
- Average ASP index is consistent with CDF estimates (same order of magnitude) from SPAR models (and therefore with licensee's models)

- Increases in the ASP index in FYs 1994, 1996 and 2002 are attributable to significant precursors

- Limitations to the Index
  - Use of CCDPs and ΔCDPs to estimate CDF is difficult due to imprecise mathematical relationships, sparse statistics and screened events
  - SPAR models only cover internal events

Integrated ASP Index

(The total CCDP of all precursors divided by the number of Rx years)



Source: SECY-05-192

Description and Results of the Integrated ASP Index

(A New Index first published in SECY-05-0192)

- Major feature – includes the risk of a precursor for the entire duration of the condition
- Initiating events are included in the year in which they occurred
- Results are consistent (same order of magnitude) with CDF estimates from SPAR and licensee models
- Insights on total contribution to integrated average CDF
  - Four precursors contribute nearly one-half
  - Three significant precursors contribute over one-quarter
  - The remaining quarter is from 156 precursors

Summary

- **ASP Program Status**
  - The program continues to evaluate the safety significance of operational events
  - FY 2004 analyses are essentially complete, and the preliminary results for FY 2005 events will be available to support the Agency Action Review Meeting (AARM) in April 2006

- **ASP Results**
  - The occurrence rate for higher risk precursors is constant or decreasing
  - The overall risk from ASP events is relatively constant
  - The number of precursors analyzed is affected by the SDP and recent increase in LOOP frequency

(14)

G:PPHandout

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
530	10	10.1

Title: PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES

Authors:
JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

10

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

From Staff Person
JOHN T. LARKINS

March 10, 2006 (8:20am)
G:\PlanPro(ACRS)\ppmins530.wpd

INTERNAL USE ONLY

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

The ACRS Subcommittee on Planning and Procedures held a meeting on March 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 2:40 p.m. and adjourned at 4:00 p.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 March ACRS meeting

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

RECOMMENDATION

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

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through May 2006 is attached (pp. 7-9). 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. 10-11).

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, resulting from the ACRS meeting with the NRC Commissioners on December 8, 2005, the Commission requested that:

> 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.

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 and the associated resource needs.

During the February 2006 ACRS meeting, the Committee authorized the ACRS Executive Director to work with the Planning and Procedures Subcommittee and develop a final response. A final response, which reflects incorporation of comments received from the Planning and Procedures Subcommittee members, was sent to the members in February 2006.

The ACRS Chairman and Executive Director have reconciled comments received from ACRS members and a revised final draft has been prepared and distributed to Committee members for comment. Following the March 2006 full Committee meeting, the ACRS Chairman will forward to the Commission the Committee's proposal for handling the anticipated workload increase. Subsequently, the ACRS Chairman and Executive Director will meet with individual Commissioners to discuss the Committee's proposal.

4) ACRS Conference Room Upgrade

During the February ACRS meeting, members were informed about the upgrade to 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 have been made to hold ACRS Subcommittee and full Committee meetings in the following locations:

Hyatt Regency Bethesda

- Thermal-Hydraulic Phenomena - March 14, 2006
- Power Uprates - March 15-16, 2006
- Plant License Renewal - April 5, 2006

Commissioners' Conference Room

- Planning & Procedures - April 5, 2006
- ACRS Full Committee Meeting - April 6-8, 2006

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director and Jenny Gallo keep the Committee informed of the status of the upgrade to the ACRS conference room.

5) Reappointment of Dr. Powers for a Fourth Term

The Commission took exception to its current policy of maximum three-term limit to the ACRS members and reappointed Dr. Powers for a fourth term.

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

The members interviewed several candidates for membership on the ACRS on March 8-9, 2006. Another candidate will be interviewed by the members during the Thermal-Hydraulic Phenomena Subcommittee meeting on March 14, 2006. Subsequent to interviewing the Candidates on March 8 and 9, the members should provide their feedback to the ACRS Chairman. The ACRS Chairman will provide the members' views to the ACRS Member Candidate Screening Panel during the March meeting.

RECOMMENDATION

The Subcommittee recommends that the members provide their views to the ACRS Chairman on the candidates they have interviewed. The members should provide the rationale for recommending a specific candidate.

7) Quadripartite Meeting Status

Planning for the 2006 Quadripartite Meeting continues as scheduled and we expect full participation from the Member Countries. There will be 15 participants from France's

GPR and IRSN; 18 participants from Germany, including 11 RSK members, three from BMU and four from the RSK secretariat; and we expect to hear from Japan soon. Among the invited participants, Switzerland's KSA will send two attendees and we have not heard from Sweden and Finland yet.

Assignments have been made to ACRS staff engineers on specific topics to assist the ACRS members in preparing the abstracts, which are due on March 31, 2006.

Next major steps include: identifying and inviting key note speakers; formally inviting the Commissioners and selected NRC staff; selecting translators for the Japanese and the French. Additionally, there are a number of other administrative issues being addressed by the ACRS staff.

<u>RECOMMENDATION</u>

The Subcommittee recommends that the members finalize abstracts on their assigned topics and that the Executive Director and Mugeh Afshar-Tous keep the Committee informed of further progress.

8) <u>April ACRS Meeting</u>

During the April 6-8, 2006 ACRS meeting, the Committee is scheduled to write a report on security-related research activities and plant-specific mitigation strategies. Since a large amount of information needs to be discussed, Dr. Bonaca, Chairman of the ACRS Subcommittee on Safeguards and Security, suggests that the April ACRS meeting be started at 1:30 p.m. on Wednesday April 5, 2006 to discuss the safeguards and security matters. Such an arrangement will preclude scheduling another presentation to the full Committee at the April ACRS meeting.

<u>RECOMMENDATION</u>

The Subcommittee recommends that the April ACRS meeting be started at 1:30 p.m. on Wednesday, April 5, 2006.

9) <u>Staff Requirements Memorandum Related to ACRS/ACNW Coordination</u>

In an SRM dated February 9, 2006 (pp. 12-13) resulting from the ACNW meeting with the NRC Commissioners on January 11, 2006, the Commission stated that the ACNW should find, with input from the ACRS, an approach to provide the Commission with a coordinated set of recommendations on how the Center for Nuclear Waste Analyses (CNWRA) might broaden its assistance to NRC, for example, to support NRR programs and/or other new and significant regulatory research activities. Additionally, in an SRM dated February 7, 2006, (pp. 14-15) on the ACNW Action Plan, there are some additional activities that the ACNW has been tasked to perform, such as staying abreast of new approaches to reprocessing technology and fuel cycle that could involve coordination with the ACRS.

<u>RECOMMENDATION</u>

The Subcommittee recommends that the Committee develop a strategy to assist ACNW in responding to the Commission request.

10) Member Issue

Proactive ACRS Initiative in Hydrogen Production Safety

During its retreat on January 26-27, 2006, the Committee discussed a proactive initiative on hydrogen production safety. The American Nuclear Society (ANS) is planning to hold an embedded topical meeting on this subject on June 25, 2007 in Boston (pp. 16). Dr. Powers suggests that interested ACRS members contact the Chairman of the Technical Program Committee to help him organize this meeting by identifying topics and speakers (pp. 17).

RECOMMENDATION

The Subcommittee recommends that those members who are interested in the ANS embedded topical meeting on Hydrogen Production Safety contact Mr. Kevin O'kula, Chairman of the ANS Technical Program Committee.

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	**Draft**
Powers	—	Fischer/Snodderly	Final Review of the Clinton Early Site Permit Application and the Final SER	A	To support staff schedule	**Draft**
		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**
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	**Draft**
	—	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	**Draft**

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ANTICIPATED WORKLOAD
APRIL 5 (1:30 P.M.) - 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 Flack	Safeguards and Security Matters **[CLOSED]** Safety Conscious Work Environment/ Safety Culture	A A	To provide Committee's views To provide Committee's views	— —
Denning	—	Caruso	**SUBCOMMITTEE REPORT** - Ginna Power Uprate Application and the Associated Safety Evaluation - **SUBC. Mtg. 3/15-16/06**	—	—	—

1

ANTICIPATED WORKLOAD
APRIL 5 (1:30 P.M.) - 8, 2006 (Cont'd)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Powers	—	Fischer	Hazards Analysis Associated With Grand Gulf ESP	A	To provide Committee's views	—
Sieber	—	Lamb	**SUBCOMMITTEE REPORT** - Interim Review of the Nine Mile Point License Renewal Application **[SUBC. Mtg. - 4/5/06]**	—	—	—
Wallis	—	Caruso	Application of TRACG Code for Analyzing ESBWR Stability **[TENTATIVE]**	A	To support staff schedule	—

D

ANTICIPATED WORKLOAD
MAY 4-6, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Thornsbury	**SUBCOMMITTEE REPORT** - ESBWR PRA **[SUBC. Mtg - 4/20/06]**	—	—	—
Bonaca	—	Thornsbury	Report on Safeguards and Security Matters **[CLOSED] [IF NOT COMPLETED IN APRIL]**	A	To provide Committee's views	**Draft**
Dennig	—	Caruso	Ginna Power Uprate Application and the Associated Safety Evaluation	A	To support staff schedule	—
		Caruso	Beaver Valley Power Uprate Application and the Associated Safety Evaluation	A	To support staff schedule	—
Maynard	—	Santos	Staff's Evaluation of Licensees' Responses to Generic Letter 2006-01, "Steam Generator Tube Integrity and Associated Technical Specifications" **[INFORMATION BRIEFING]**	Report as needed	To provide Committee's views	—
Kress	Powers	Fischer	Proposed Revisions to 10 CFR Part 52	A	To support staff schedule	—
Powers	—	Fischer	Lessons Learned from the Review of Early Site Permit Applications	A	To provide Committee's views	—
Sieber	—	Lamb	Final Review of the Brunswick License Renewal Application and the Associated Final SER	A	To support staff schedule	—

2

1 **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 staff also described the resolution of this GSI in a memorandum

10.

dated February 15, 2006, from Mark A. Cunningham, Director, Division of Engineering, to John T. Larkins, Executive Director, ACRS.

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

11.

IN RESPONSE, PLEASE
REFER TO: M060111B

February 9, 2006

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

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

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

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

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

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

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

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

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

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

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

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

February 7, 2006

MEMORANDUM TO: Michael T. Ryan, Chairman
Advisory Committee on Nuclear Waste

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

SUBJECT: STAFF REQUIREMENTS - COMSECY-05-0064 - FISCAL YEAR 2006 AND 2007 ACTION PLAN FOR THE ACNW

The Commission has approved the Fiscal Year 2006 and 2007 Action Plan for the Advisory Committee on Nuclear Waste, subject to the comments noted below.

The ACNW should be prepared to advise the Commission on the unique waste management, decommissioning, and environmental protection issues related to the licensing of in-situ leach (ISL) uranium recovery facilities which may arise from a Part 41 rulemaking addressing uranium recovery.

The ACNW should remain abreast of industry, technical and legal developments in the areas of spent fuel storage, disposal and reprocessing to ensure that members will be ready to provide advice in these areas, should the need arise.

The current second tier "Fuel Cycle Facilities" topic may need to move to the first tier, if new approaches to the fuel cycle are proposed. As such, an important design criterion for any new reprocessing effort will be that decommissioning costs be manageable. The ACNW's early thoughts on these issues could prove very helpful.

The committee should broaden its focus on LLW beyond merely risk-informing 10 CFR Part 61. Given that the Department of Energy is engaged in evaluating disposal options for Greater-Than-Class-C waste and the Barnwell LLW facility plans to only accept Class B and C waste from compact members by 2008, the committee should work with staff to determine the adequacy of our regulatory infrastructure to meet future challenges in this area and provide advice to the Commission on potential changes, as appropriate.

The Committee should revise its FY06/07 Action Plan consistent with this SRM and the Commission's SRM on its January 2006 meeting with the Committee. The revision should reprioritize the Committee's activities and address the Committee's resource needs to accomplish the directed work.

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

American Nuclear Society

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June 25, 2007

Annual Meeting

ANS National Meetings and Embedded Topicals

June 24 - 28, 2007 Boston, MA

General Co-Chairs: J. Art Stall (Florida Power & Light)
Richard F. Gil (Shaw Stone & Webster Nuclear)
Technical Program Chair: Dr. Ray Klann

Embedded Topical Meeting:
Nuclear Technology and Safety Aspects of Hydrogen Production, Management and Control (NST-H2-PCM)
General Chair: David Henderson (U.S. Department of Energy)
Program Chair: Kevin R. O'Kula (Washington Safety Management Solutions, LLC)

ANS National Meetings and Embedded Topicals
ANS Topicals and Executive Conferences
ANS Cosponsored Meetings

January 2007

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<< 2006 2008 >>

Questions or comments about the American Nuclear Society web site? Contact the ANS Webmaster.

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From: "Powers, Dana A" <dapower@sandia.gov>
To: "Sam Duraiswamy" <SXD1@nrc.gov>, <Graham.B.Wallis@Dartmouth.EDU>, "WJ Shack" <wjshack@anl.gov>
Date: 3/1/06 11:10AM
Subject: Hydrogen Production Safety

To: Planning and Procedures Subcommittee
From: D.A. Powers

Subject: Proactive ACRS Initiative in Hydrogen Production Safety

At its recent retreat, the ACRS professed a proactive interest in the safety of the production of hydrogen using nuclear power. The American Nuclear Society is planning an embedded topical meeting on exactly this subject. If members of ACRS really do have an interest, they might contact the chairman of the technical program committee:

Kevin O'Kula
kevin.Okula@wsms.com

I know that Kevin is looking for help in defining the scope of the embedded topical meeting.

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(15)

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ACRS MEETING HANDOUT

<table>
<tr><td>Meeting No.

530th</td><td>Agenda Item

11</td><td>Handout No.:

1</td></tr>
<tr><td colspan="3">Title RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS</td></tr>
<tr><td colspan="2">List of Documents Attached

See attached list</td><td>11</td></tr>
<tr><td>Instructions to Preparer
1. Paginate Attachments
2. Punch holes
3. Place Copy in file box</td><td colspan="2">Lead Staff Person
SAM DURAISWAMY</td></tr>
<tr><td></td><td colspan="2"></td></tr>
</table>

SUBJECT	ANALYSIS	EDO LTR.	ACRS LTR.
Vermont Yankee Extended Power Uprate (RSD/RC)	3/06/06 (p. 1)	2/09/06 (pp. 2-3)	1/4/06 (pp. 4-11)

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

March 7, 2006

MEMORANDUM TO: ACRS Members

FROM: R. Caruso, Senior Staff Engineer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER CONCERNING THE VERMONT YANKEE EXTENDED POWER UPRATE

Attached for your information is a copy of the EDO's February 9, 2006 response to the ACRS's letter of January 4, 2006, concerning the Committee's review of the proposed Vermont Yankee Extended Power Uprate (EPU). A copy of the Committee's letter is also attached.

Committee Letter

In its letter, the Committee recommended that (1) the application by Entergy for the EPU should be approved, (2) the requested overpressure credit should be approved, (3) large transient testing is not warranted, (4) the times available to perform critical operator actions remain adequate, (5) the proposed interim resolution of the GE methods issue is accetable, (6) the planned monitoring during power ascension provides adequate assurance that resonant vibrational modes in the steam dryer will be identified, (7) additional expanded inspection in support of the EPU is not warranted, and (8) RS-001 has provided a structured process for review, and its continued use and improvement are encouraged.

EDO Response

The EDO accepted all of these recommendations and conclusions, and noted that the ACRS had expressed some additional comments concerning the methodology used to assess the containment overpressure credit issue. The EDO stated that the staff will consider the ACRS comments as it develops more explicit guidance as part of the ongoing revisions to RG 1.82. The staff is currently developing guidance for a new approach that would include statistical analyses of the uncertainty, as recommended by the ACRS, and it expects to bring the revised RG to the Committee in the future.

Analysis

The EDO's response is satisfactory. The staff has taken the Committee's comments to heart, and has already described some of its plans with the T/H subcommittee, in January 2006. The staff plans to have another version of RG 1.82 ready for the Committee in June 2006.

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

February 9, 2006

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

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION, EXTENDED POWER UPRATE

Dear Dr. Wallis:

On December 7, 2005, during the 528th meeting of the Advisory Committee on Reactor Safeguards (ACRS), the U.S. Nuclear Regulatory Commission (NRC) staff presented its review of the Vermont Yankee Nuclear Power Station (VYNPS) extended power uprate (EPU) application from Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy or the licensee). The VYNPS EPU application was also discussed during meetings of the ACRS Subcommittee on Power Uprates in Brattleboro, Vermont, on November 15 and 16, 2005, and in Rockville, Maryland on November 29 and 30, 2005.

In a letter to Chairman Diaz dated January 4, 2006, the ACRS provided the following conclusions and recommendations regarding the VYNPS EPU application:

1. The Entergy application for the EPU at the VYNPS should be approved.

2. The change in the licensing basis associated with the requested containment overpressure credit should be approved.

3. Load rejection and main steam isolation valve closure transient tests are not warranted. The planned transient testing program adequately addresses the performance of the modified systems.

4. The times available to perform critical operator actions remain adequate under EPU conditions.

5. The margin added to the safety limit minimum critical power ratio is an appropriate interim measure until General Electric obtains additional data to complete the validation of nuclear analysis methods.

6. The monitoring that will be performed during the ascension to uprate power provides adequate assurance that, if resonant vibrational modes are induced in the steam dryer, they will be identified prior to component failure.

7. An enhanced, focused engineering inspection was performed. An additional expanded inspection is not warranted.

8. The review standard for EPUs (RS-001) provides a structured process for the review of applications for EPUs. Its continued use and improvement are encouraged.

In addition to the recommendations and conclusions, the ACRS provided some general comments on the NRC staff's review of the VYNPS EPU application. Regarding the issue of crediting containment overpressure for determination of the available net positive suction head for the emergency core cooling system pumps, the ACRS letter stated that:

> Although we concur with the staff's conclusion to grant credit for containment overpressure, we would have preferred to see the assessment performed and presented in a more coherent manner, with a more complete and rigorous consideration of uncertainties. The staff is developing additional guidance to be used in the consideration of overpressure credit in the future. We look forward to reviewing their proposed approach.

The letter provided some additional comments from several ACRS members which addressed a proposed approach for consideration of uncertainties as part of an assessment of crediting containment overpressure. The NRC staff will consider the ACRS comments as it develops more explicit guidance as part of the ongoing revisions to Regulatory Guide (RG) 1.82. Based on discussions with the ACRS, during NRC staff presentations related to the proposed revisions to RG 1.82, the staff understands that the ACRS would prefer that licensees use a statistical approach for the analysis related to crediting containment ovepressure. The staff is currently developing guidance for this new approach and will bring the revised RG 1.82 to the Committee in the future.

Thank you for your comments. The NRC staff appreciates the Committee's insights concerning the VYNPS EPU amendment review.

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-2174

January 4, 2006

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

SUBJECT: VERMONT YANKEE EXTENDED POWER UPRATE

Dear Chairman Diaz:

During the 528th meeting of the Advisory Committee on Reactor Safeguards, December 7-9, 2005, we discussed the Vermont Yankee Extended Power Uprate (EPU) Application. As part of this review, our Subcommittee on Power Uprates held a meeting on November 15 -16, 2005 in Brattleboro, Vermont to receive input from the public, the applicant, and the staff. A second Subcommittee meeting was held in Rockville, Maryland on November 29 - 30, 2005. During our review, we had the benefit of discussions with the staff, the public, and Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy), the licensee. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The Entergy application for the extended power uprate at the Vermont Yankee Nuclear Power Station (VY) should be approved.

2. The change in the licensing basis associated with the requested containment overpressure credit should be approved.

3. Load rejection and main steam isolation valve closure transient tests are not warranted. The planned transient testing program adequately addresses the performance of the modified systems.

4. The times available to perform critical operator actions remain adequate under EPU conditions.

5. The margin added to the safety limit minimum critical power ratio (SLMCPR) is an appropriate interim measure until General Electric (GE) obtains additional data to complete the validation of nuclear analysis methods.

6. The monitoring that will be performed during the ascension to uprate power provides adequate assurance that, if resonant vibrational modes are induced in the steam dryer, they will be identified prior to component failure.

7. An enhanced, focused engineering inspection was performed. An additional expanded inspection is not warranted.

8. The review standard for extended power uprates (RS-001) provides a structured process for the review of applications for extended power uprates. Its continued use and improvement are encouraged.

BACKGROUND

Vermont Yankee Nuclear Power Station (VY) is a boiling-water reactor of the BWR/4 design with a Mark-1 containment. Entergy has applied for an extended power uprate of approximately 20% from the current maximum authorized power level of 1593 MWt to 1912 MWt. The application is similar to other uprates that have been approved within the last five years at Duane Arnold, Dresden Units 2 and 3, Quad Cities Units 1 and 2, and Brunswick Units 1 and 2. In Constant Pressure Power Uprates (CPPU), except for steam and feedwater flow rates, plant operating conditions are essentially unchanged from the pre-EPU values. The extra power is generated largely by flattening the power distribution across the core, and the fuel design safety limits are met at the proposed extended power uprate conditions.

DISCUSSION

When a large-break design-basis loss-of-coolant accident (LOCA) and anticipated transient without scram (ATWS) were analyzed at VY at the proposed EPU level using current design basis assumptions and methodologies, the available net positive suction head (NPSH) was found to be insufficient to avoid cavitation of the low pressure coolant injection (LPCI) and core spray pumps. The need for increased NPSH occurs because at the higher power level the suppression pool heats up more in both of these scenarios than at the currently licensed power level. In the calculations performed to support VY's existing operating license, containment pressure was assumed to be atmospheric when computing the available NPSH.

In its application, Entergy requests changing its licensing basis methodology to grant credit for containment accident pressure in determining available NPSH for emergency core cooling pumps for these LOCA and ATWS scenarios. Using conservative methods and a containment leak rate consistent with its technical specifications, Entergy has determined a conservative lower bound for the time-dependent pressure in containment that would result from these scenarios under EPU conditions. The incremental pressure credits that are requested for these two scenarios are less than these computed pressures. For the LOCA scenario, the maximum containment pressure credit is 6 psi, and the total time for which some overpressure credit is required is 56 hours. For the ATWS scenario, the corresponding values are 2 psi and 1 hour.

The ACRS has historically opposed a general granting of containment overpressure credit. In determining whether such credit should be granted, one aspect to be considered is whether practical alternatives exist, such as the replacement of pumps with those with less restrictive NPSH requirements. If no practical alternatives are available, important considerations include (1) the length of time for which containment pressure credit is required and (2) the margin between the magnitude of the pressure increment that is being granted and the expected minimum containment pressure. Another consideration is the nature of the containment design and whether it provides a positive indication of integrity, prior to the event, as is the case in subatmospheric and inerted designs.

Because of the plant configuration, extent of modifications required, and worker dose that would be involved, we conclude that there are no practical design modifications that would preclude the need to consider the request for containment overpressure credit. VY has an inerted containment. There is, then, a low likelihood of significant pre-existing containment leakage. For the ATWS scenario, the magnitude of pressure required to show adequate NPSH is small compared to the accident pressure, and the time during which the overpressure credit is required is short. For the LOCA scenario, although the duration for which the containment overpressure credit is required is comparatively long, the overpressure credit requested is smaller than what is conservatively predicted to be available.

Under the EPU conditions at VY, the general design requirements regarding single failures in design-basis accidents do not prevent granting of the overpressure credit for the LOCA scenario of concern. The worst single failure that was identified by the licensee involves loss of one train of heat removal from the suppression pool. Conservative, bounding calculations show that the containment overpressures during this scenario are higher than needed to provide sufficient NPSH. Allowing no credit for containment overpressure is equivalent to assuming an additional failure that causes loss of the overpressure. Thus, for all scenarios involving only a single failure, sufficient NPSH is available to ensure that pump cavitation damage is avoided. To maintain defense-in-depth, however, it has been staff practice to require the assumption that containment overpressure is not available in assessing the potential for pump damage.

In evaluating Entergy's request for containment overpressure credit, the staff included in its decisionmaking process more realistic analyses to determine whether containment overpressure would be needed at the proposed EPU power level to prevent pump cavitation in actual accident scenarios. The staff also considered the results of probabilistic analyses to assess the risk significance of scenarios in which containment overpressure is lost.

Design-basis accidents are typically analyzed using conservative methodologies and input assumptions to ensure safety in spite of uncertainties in input and methodology. An alternative approach is to use realistic analyses with a more complete and explicit consideration of uncertainties. Such a methodology has not yet been fully developed for analysis of the need for containment overpressure credit. The staff and the licensee have instead performed sensitivity analyses to determine the effect of relaxing some of the conservative assumptions. More realistic values were used for a number of input parameters to determine the associated reduction in the predicted temperature of the suppression pool, which is the major parameter in determining whether overpressure credit is necessary. The staff concluded that, on a more realistic but still conservative basis, the temperature of the suppression pool would not become high enough in the LOCA scenario to require a credit for containment overpressure.

Independent risk analyses were performed by the staff and the licensee to determine the potential risk significance of granting credit for containment overpressure. These analyses included the conservative assumption that the emergency core cooling system (ECCS) success criteria would not be met whenever containment overpressure is lost and design-basis analyses would suggest that overpressure credit was needed, although the licensee's sensitivity studies indicated that peak suppression pool temperature would probably not be high enough that containment overpressure credit would be required. The results of the analyses indicate that the overall risk associated with the EPU is small and that the change in risk resulting from allowing the requested containment overpressure credit is also small.

Although we concur with the staff's conclusion to grant credit for containment overpressure, we would have preferred to see the assessment performed and presented in a more coherent manner, with a more complete and rigorous consideration of uncertainties. The staff is developing additional guidance to be used in the consideration of overpressure credit in the future. We look forward to reviewing their proposed approach.

The staff performed an expanded engineering inspection of VY. Such an inspection was requested by the Public Service Board of the State of Vermont. The inspection focused on safety-significant components and operator actions. It was performed under the direction of the NRC Office of Nuclear Reactor Regulation (NRR) and included regional inspectors and contractors who had no recent oversight responsibilities for VY. There were eight findings, but they were of low safety significance. A number of members of the public asked for a more extensive inspection, similar to that performed at the Maine Yankee plant. Based on the results of the inspection that was performed and the performance of VY as determined by the Reactor Oversight Process, such an extensive inspection is not warranted.

Hardware and operational changes are required for the power uprate. In order to achieve the proposed EPU power level, all three feedwater pumps must operate, rather than the two pumps currently required. If one of these pumps fails, the plant will undergo an automatic runback of power so that the two remaining pumps will be sufficient. A new signal has been added to trip a feedwater pump in the event of a condensate pump trip. A concern has been raised about the potential for loss of all feed pumps due to low suction pressure as a result of a condensate pump trip. Consequently, Entergy has agreed to perform a trip of a condensate pump to demonstrate that it will not cause loss of all feedwater. This will also test the integrated response of control systems associated with recirculation flow runback, feedwater level control, and reactor pressure control.

Entergy does not plan to undertake large transient tests, such as a main steam isolation valve closure that would result in a reactor trip. Such tests would not directly address confirmation of the performance of systems changed to support EPU. The ACRS concurs with the staff's assessment that the large transient tests are not warranted.

Only minor changes have been made in the emergency operating procedures to accommodate EPU modifications. One of the impacts of the power uprate is a reduction in available response time for operator actions. The operators respond in essentially the same manner as for the current operating conditions but, in some cases, have less time to take an action. A systematic assessment has been made by Entergy of the maximum time available for critical operator actions. The VY simulator has been modified to represent the EPU condition and operators have been trained for EPU conditions. The simulator exercises have demonstrated the ability of the operators to respond correctly within the required time period.

The reactor operating domain is defined so that: (1) the core will not be operated in an unstable regime, (2) the minimum critical power ratio is low enough to prevent dryout of the fuel pins, and (3) the linear heat generation rate is low enough to assure the integrity of fuel cladding during steady and transient conditions. The boundaries of this operating domain are based on neutronic and thermal-hydraulic calculations performed by GE. The computer codes that are used in these analyses have been reviewed and approved by the staff.

In reviewing the application of these methods to EPU uprates, the staff determined that the operation of the fuel extends into a region where the expected void fraction within the fuel bundle is greater than that for which the codes have been validated. To demonstrate the ability of the code to predict isotopic concentrations in this regime, GE has committed to performing gamma scans on the fuel design that is being used in the power uprate. In the interim, Entergy has undertaken an "Alternative Approach" in which it has performed an uncertainty analysis for the model predictions and, as a result, has added an additional margin of 0.02 to the SLMCPR. We concur with the staff's assessment that the addition of such a margin is an appropriate interim measure. The review of the adequacy of the GE computer codes is a generic activity that is being undertaken by the staff. We will have an opportunity to review the staff's assessment of these codes in more detail when we consider the MELLLA+ topical report in 2006.

Higher steam and feedwater flow rates at EPU conditions may lead to an increase in flow accelerated corrosion for some components. The evidence indicates that current flow accelerated corrosion rates at VY are low. Many of the components that would most likely be affected use chromium- molybdenum alloy materials that are resistant to flow accelerated corrosion, and Entergy has committed to an inspection program that will provide reasonable assurance that degradation will be detected prior to reaching an unsafe condition.

Increased flow rates also have the potential to induce vibrations that could lead to failure of components. Because of the previous experience at Quad Cities, the steam dryer has been the primary focus of attention. A number of cracks have been found in inspections of the VY steam dryer. Two cracks found near the lifting lugs were attributed to the initial fabrication of the steam dryer. These cracks have been ground out and repaired. The other cracks that have been found appear to be superficial and were deemed to be the result of intergranular stress corrosion, not flow-induced vibration. Stiffeners have been added to the dryer to provide additional strength and also to raise its natural frequencies.

Entergy has performed hydrodynamic, acoustic and structural resonance analyses to assess the potential for stimulation of a resonant mode of the dryer. These analyses indicate that there is margin between the magnitude of the potential stresses imposed on the steam dryer and the level at which fatigue failure would occur. However, the state of validation of these methods is poor.

To provide further assurance of the integrity of the dryer, additional strain gages have been added to the steam lines at VY. Experiments performed in a scale-model system by GE indicate that acoustic signals initiated in the region of the steam dryer can be correlated with signals measured by strain gages on the steam lines. A similar correlation has been observed at Quad Cities Unit 2 where both the steam dryer and steam lines have been instrumented.

Entergy has developed a program for power ascension involving holds at a number of power levels. The steam line strain gages will be monitored at the various power levels. Any anomalies will lead to a reduction in power until the issue is resolved. Entergy has also committed to inspections of the steam dryers in the next three outages following the uprate. The additional monitoring, the power ascension program, and the inspections provide confidence that, if excessive excitation does occur in the steam dryer, it will be identified before substantial damage is incurred.

Power uprates are not submitted as risk-informed license applications. Nevertheless, licensees have submitted assessments of risk associated with the extended power uprates and the staff includes consideration of this risk information in its decisionmaking process. The purpose of the staff's risk review as stated in RS-001 is to "determine if there are any issues that would potentially rebut the presumption of adequate protection provided by the licensee meeting the deterministic requirements and regulations." The staff has reviewed Entergy's assessment of risk at the proposed EPU conditions and compared the VY probabilistic risk assessment (PRA) results with the staff's SPAR model results for this plant. The values of core damage frequency (CDF) and large early release frequency (LERF) are low and provide substantial margin to values that raise questions of adequate levels of safety. As we noted previously, the staff also used risk insights in their independent determination of the acceptability of the potential for pump cavitation during long-term core cooling in LOCA and ATWS scenarios.

This was the second application by the staff of RS-001 in the review of an EPU proposed upgrade. RS-001 provides a structured approach to the review.

Sincerely,

/RA/

Graham B. Wallis
Chairman

Additional Comments by ACRS Members Richard S. Denning, Thomas S. Kress, Victor H. Ransom, and Graham B. Wallis

Considering all the evidence, including precedents set at other similar plants, we agreed with our colleagues to approve the proposed 20% EPU for VY.

It seems unlikely that there will be a problem with adequate NPSH of the core spray and residual heat removal (RHR) pumps at Vermont Yankee, with a 20% power uprate. However, we were asked to make a professional judgment that would have been more straightforward if the information supplied to us had been more complete. We suspect that more information already exists that could be reorganized, supplemented as needed, and presented logically to provide a more convincing case in the following way, which would set a better precedent for future applications:

1. Derive sufficient detail of the probability distribution for containment pressure following large LOCA and ATWS sequences, based on realistic analysis of the physical phenomena and the attendant uncertainties.

2. Derive sufficient detail of the probability distribution for suppression pool temperature following these events, based on realistic analysis of the physical phenomena and the attendant uncertainties.

3. Combine the results of steps 1 and 2 with realistic and uncertainty analyses of other phenomena influencing NPSH to derive the probability of successful operation of RHR and core spray pumps. This may provide adequate evidence for a conclusion to be reached, if it can be shown that only a small containment overpressure is likely to be needed for a short time, if at all, and it has a high probability of being available. If further evidence is required, these results can be incorporated into the PRA to derive the realistic contribution, if any, to total plant risk due to insufficient NPSH.

Both Entergy and the staff have shown that relaxing a few of the many conservatisms and using realistic values (for example, of the initial temperature of the suppression pool) removes the need for additional NPSH. Such arguments are insufficiently conclusive. The reason is that when one gives up an element of conservatism, without replacing it by a less stringent assumption that is still demonstrably conservative, there is a finite probability that values of the derived parameter will not bound all possibilities. The proper way to relax the many conservative assumptions is to make (some of) them realistic with the inclusion of uncertainty. This will lead to a probability distribution (or more precisely some aspects of it, such as the 95/95 confidence level) for an output such as pool temperature.

From the analyses that we have seen in presentations by Entergy and by the staff, it appears likely that the realistic contribution to risk from inadequate RHR and core spray pump NPSH will prove to be very small, even essentially zero, for the case of the proposed power uprate at VY, but this could be better demonstrated in a manner which is both physically and logically consistent. The probabilities associated with the governing physical phenomena may be regarded as more secure than some other inputs to the usual PRA assessment. Conclusions based on them may help to convince those who doubt if conventional risk-based arguments alone should allow the relaxation of defense-in-depth that is achieved by the independence of cladding and containment barriers to radioactivity release. In particular, if it can be shown that the probability of needing containment overpressure is sufficiently small, the independence of these barriers would effectively be preserved.

<u>REFERENCES</u>:

1. Memorandum from Ledyard B. Marsh to John Larkins, "Vermont Yankee Nuclear Power Station - Draft Safety Evaluation for the Proposed Extended Power Uprate (TAC No. MC0761)", October 21, 2005

2. Letter from Wayne Lanning to Jay Thayer, "Vermont Yankee Nuclear Power Station, NRC Inspection Report 05000271/2004008", December 2, 2004

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* See previous concurrence.

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