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

October 26, 2005

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

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 526th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, OCTOBER 6-7, 2005, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 526th meeting, October 6-7, 2005, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following letters and memoranda:

LETTERS:

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

- Proposed Recommendation for Resolving Generic Safety Issue-80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments," dated October 18, 2005

- Interim Report on the Safety Aspects of the License Renewal Application for the Browns Ferry Nuclear Plant Units 1, 2, and 3 dated October 19, 2005

MEMORANDA:

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

- Proposed Rule on Safety/Security Interface, dated October 11, 2005.

- Draft Regulatory Guide DG-1120 (DG-1120), "Transient and Accident Analysis Methods," and Standard Review Plan Chapter 15 Section 15.0.2 (SRP 15.0.2), "Review of Transient and Accident Methods," dated October 13, 2005.

OTHER:

- Letter to Carl J. Paperiello, Director, Office of Nuclear Regulatory Research, from William J. Shack, Acting Chairman, ACRS: ACRS Assessment of the Quality of Selected NRC Research Projects - FY 2005

- Letter to Nancy Burton, Connecticut Coalition Against Millstone, from Graham B. Wallis, Chairman, ACRS, regarding the License Renewal of Millstone Power Station, Units 2 and 3, dated October 18, 2005

HIGHLIGHTS OF KEY ISSUES

1. Interim Review of the License Renewal Application for the Browns Ferry Nuclear Plant, 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 for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3 and the related Safety Evaluation Report (SER) with Open Items. 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. TVA described the layup program for Unit 1 and its plans to restart this unit in 2007 such that all three units will be operationally identical. TVA has not taken credit for the Unit 1 Layup Program as the sole basis for determining the acceptability of components for restart. TVA also described its Periodic Inspection Program for Unit 1 components that have not been replaced. As a result of our review, the staff elevated the issue of the Unit 1 Periodic Inspection Program from a confirmatory item to an open item The staff described the three other open items identified from its review of this license renewal application. The regional inspectors found that plant equipment was being adequately maintained but another NRC inspection will be performed after TVA has progressed further in its development of aging management programs.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations on this matter, dated October 19, 2005. The Committee agreed with the staff's interim evaluation of the license renewal application related to BFN Units 2 and 3 and concluded that the plant-specific operating experience for BFN Unit 1, by itself, does not fully meet the intent of the license renewal rule. The Committee recommended that the final SER include a cohesive discussion of the applicability of operating experience from Units 2 and 3 to Unit 1 as well as a description of the attributes of the Periodic Inspection Program for Unit 1 components that will not be replaced. The Committee also recommended that if an extended power uprate is implemented, the staff require that TVA evaluate operating experience of Units 1, 2, and 3 at the uprated power level prior to entering the period of extended operation.

2. Proposed Recommendation for Resolving Generic Safety Issue (GSI)-80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of Boiling Water Reactor Mark I and II Containments"

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the proposed recommendation for resolving GSI-80. Damage to the control rod drive (CRD) hydraulic lines by mechanical impact resulting from a loss-of-coolant accident (LOCA) was raised as an issue by the ACRS in 1978 during operating license reviews of certain boiling water reactors (BWRs). The Office of Nuclear Regulatory Research (RES) performed an assessment to address this issue. The core damage frequencies (CDFs) for the various Residual Heat Removal and Reactor Coolant System pipe break events that could potentially impact Control Rod Drive piping were determined for Mark I and Mark II plants. All of the calculated CDF values were less than the threshold (10^{-6} event/Reactor Year) specified in the Handbook to Management Directive 6.4, "Generic Issues Program." Therefore, RES recommends that GSI-80 be closed with no changes to the existing regulations or guidance.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations on this matter, dated October 18, 2005, agreeing with the staff's proposed recommendation to close GSI-80 without any changes to the regulations or guidance.

3. Resolution of ACRS Comments on the Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants"

The Committee heard presentations by and held discussions with representatives of the staff regarding the resolution of the ACRS comments and recommendations, included in its June 14, 2005 letter, on the draft final Regulatory Guide, "Risk-informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," and NEI document, NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)."

The staff stated that it agrees with all the ACRS recommendations except one regarding definitions of the term maximum expected fire scenario and limiting fire scenario. The staff provided a status of the licensee implementation of National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition. The staff stated the Regulatory Guide and the Nuclear Energy Institute (NEI) 04-02 are being revised to address the ACRS comments. The staff did not seek endorsement of the Regulatory Guide at this meeting, because the staff wants to ensure compatibility with other staff documents, such as 10 CFR 50.69, 10 CFR 50.46(a), and 10 CFR 50.48 (c). The staff said that it will provide the final draft Regulatory Guide and NEI 04-02 document to the ACRS and seek endorsement to issue the Regulatory Guide after all the changes are made.

Committee Action

The Committee did not write a letter at this time at the staff's request. The Committee plans to review the revised Guide and the NEI document in the future.

4. Davis-Besse Reactor Pressure Vessel Head Integrity Calculations

The Committee met with representatives of the NRC staff to discuss calculations performed to support the Accident Sequence Precursor (ASP) analysis of the 2002 Davis-Besse events. Specifically, at the April 2005 meeting, the Committee requested more information regarding calculations that supported the determination of LOCA frequencies for the ASP analysis. The RES staff agreed to provide this briefing on the probabilistic structural mechanics calculations that supported the analysis. The staff discussed the details of the structural mechanics testing program and calculations that assessed the as-found condition of the reactor pressure vessel head, predictions of vessel head failure if the plant had not been taken off-line, and postulated vessel head conditions during the year before the plant was taken off-line. As part of this last discussion, the staff also addressed the expert elicitation results that supported the ASP analysis of potential loss-of-coolant accidents. Several Members praised the staff regarding the quality of this work and the quality of the presentation.

Committee Action:

This was an information briefing. The Committee plans to review the staff's annual report to the Commission on the status of the ASP Program, including quantitative ASP results.

5. Quality Assessment of the Selected NRC Research Projects

The NRC Strategic Plan that was developed in accordance with the requirements of the Government Performance and Results Act (GPRA) requires that RES have an independent evaluation of the quality of its research programs. The Committee has agreed to assist RES in assessing the quality of selected research projects. The Committee completed its report on the assessment of the quality of selected research projects on: Station Blackout Risk Evaluation of Nuclear Power Plants; Steam Generator Tube Integrity Program at the Argonne National Laboratory; and Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag at the Penn State University.

Committee Action

The Committee approved the letter transmitting the report on ACRS Assessment of the Quality of Selected NRC Research Projects to the Director of RES. The Committee anticipates receiving from RES a list of candidate projects for review during the next twelve months.

6. Licensees' Responses to the Bulletin on, "Emergency Preparedness and Response Actions for Security-Based Events"

The Committee met with representatives of the NRC staff to discuss licensees' responses to NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events." During the June 2005 meeting, the Committee considered the proposed bulletin and decided to hold any review until the staff's reconciliation of the licensees' responses. The staff discussed the licensee's responses to the Bulletin. The purpose of the Bulletin was to gather together information on actions licensees have taken in this area in response to the many orders, advisories, lessons-learned, and non-regulatory communications. From the responses, nearly all licensees intend to address the issues raised in the Bulletin in their emergency preparedness plans. The staff noted that several of the issues discussed in the Bulletin were related to previous ACRS recommendations. One issue discussed in detail with the staff related to how to escalate the emergency action levels for security-based events and their relationship to the traditional accident-based emergency action levels. A portion of this session was closed to the public to discuss more sensitive information related to the progression of events during security-related emergencies.

Committee Action:

This was an information briefing. The Committee plans to continue its review of selected security-related issues in the future.

7. NRC Staff's Response to the ACRS Letter on the Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident"

The Committee heard a presentation by the staff concerning its plans to further revise its proposed Regulatory Guide 1.82, Rev. 4, to incorporate more risk-informed practices to granting containment overpressure credit, and to apply those practices for the first time as part of its review of the Vermont Yankee power uprate request. The Committee questioned how this approach would reconcile its concerns about maintaining sufficient defense-in-depth as part of the design of the containment and emergency core cooling systems. The staff stated that it would require licensees that propose to apply this method to demonstrate that the five key principles of risk-informed decisionmaking in RG 1.174 are met. The staff will clarify its requirements and describe its expectations for licensees who submit risk-informed license amendments to credit containment overpressure. The staff intends to provide the ACRS with a revised Regulatory Guide 1.82 for review and comment.

Committee Action

The Committee plans to review the revised Revision 4 to Regulatory Guide 1.92.

8. Format and Content of the NRC Safety Research Program Report to the Commission

During the October 6-8 2005 ACRS meeting, the Committee discussed the format and content of the ACRS biennial report to the Commission on review and evaluation of the NRC safety research program.

Committee Action

The Committee plans to discuss a draft report on NRC safety research program during its November 3-5, 2005 meeting.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee considered the EDO's response of September 12, 2005, to ACRS's July 15, 2005 report responding to an April 25, 2005 Staff Requirement Memorandum (SRM), requesting the Committee to "provide the Commission a list of research projects it intends to review in the short term as part of its assessment of research quality, with an indication of the methodology the Committee will use for the reviews."

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

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from September 8, 2005, through October 5, 2005, the following Subcommittee meetings were held:

- Plant License Renewal and Plant Operations - Browns Ferry Unit 1 - September 21, 2005

The Subcommittees reviewed the license renewal, power uprate, and restart activities associated with Browns Ferry Unit 1.

- Plant License Renewal - Browns Ferry Nuclear Plant - October 5, 2005

The Subcommittee reviewed the License Renewal Application and associated SER with Open Items for the Browns Ferry Nuclear Plant, Units 1, 2, and 3.

- Planning and Procedures - October 5, 2005

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

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee requested an information briefing regarding the FERRET reactor vessel fluence methodology.
- The Committee plans to review the license renewal application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 and the related draft final SER.
- The Committee plans to review the draft final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants."
- The Committee plans to review the staff's annual report to the Commission on the status of the ASP Program, including quantitative ASP results.
- The Committee plans to continue its review of selected security-related issues in the future.
- The Committee plans to review the Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-term Recirculation Cooling Following a Loss-of-Coolant Accident."
- The Committee would like an information briefing on the staff's review of the FERRET reactor vessel fluence methodology.

PROPOSED SCHEDULE FOR THE 527TH ACRS MEETING

The Committee agreed to consider the following topics during the 527TH ACRS meeting, to be held on November 3-5, 2005:

- Final Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2
- Draft Final Generic Letter 2005-xx, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power"
- Economic Simplified Boiling Water Reactor (ESBWR) Design
- Draft ACRS Report to the Commission on the NRC Safety Research Program
- Digital Systems Research Plan
- Status of Rulemaking on Post-Fire Operator Manual Actions

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

CERTIFIED

Date Issued: 11/21/2005
Date Certified: 11/30/2005

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OCTOBER 6-8, 2005

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APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
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IV. Future Agenda and Subcommittee Activities
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MINUTES OF THE 526th MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
OCTOBER 6-8, 2005
ROCKVILLE, MARYLAND

The 526th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on October 6-8, 2005. Notice of this meeting was published in the *Federal Register* on September 22, 2005 (65 FR 55637) (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. William J. Shack (Vice Chairman), Mr. John D. Sieber, (Member-at-Large), Dr. George E. Apostolakis, Dr. Mario V. Bonaca, Dr. Richard S. Denning, Dr. F. Peter Ford, Dr. Thomas S. Kress, Dr. Dana A. Powers, and Dr. Victor H. Ransom. Dr. Graham B. Wallis did not attend this meeting. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

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

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

II. Interim Review of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Open)

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

The Committee met with NRC staff and representatives of the Tennessee Valley Authority (TVA) to perform an interim review of the license renewal application of the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3 and the related Safety Evaluation Report (SER) with Open Items. 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. The license renewal application is based on the currently licensed thermal power levels for each unit.

BFN Unit 1 operated for approximately ten years but is currently shutdown. The Unit 1 Layup Program follows the guidance in EPRI NP-5106 "Sourcebook for Plant Layup and Equipment Preservation." TVA has not taken credit for this layup program as the sole basis for determining the acceptability of components for restart. In addition, lessons learned from the layup and restart of Unit 3 are being applied to Unit 1. TVA plans to restart Unit 1 in May 2007. At restart, Unit 1 will be operationally identical to Units 2 and 3. TVA has a total of 39 license renewal aging management programs (AMPs). The Unit 1 periodic inspection program is the only AMP that is not common to all three units. This program will inspect a subset of piping locations that were not replaced before restart. Inspections will be performed before restart, before entering the period of extended operation, and after entering the period of extended operation. The frequency of subsequent inspections will be determined based on the results of these inspections.

The staff described the four open items related to this license renewal application. As a result of our review, the staff elevated the issue of the Unit 1 periodic inspection program from a confirmatory item to an open item. The other open items deal with potential corrosion of an inaccessible portion of the drywell shell, stress relaxation of core plate hold-down bolts, and inspection of residual heat removal service water piping. The regional inspectors found that plant equipment was being adequately maintained but another NRC inspection will be performed after TVA has progressed further in the development of their aging management programs.

Committee Action

The Committee issued a report to the NRC Executive Director for Operations (EDO) dated October 19, 2005. The Committee agreed with the staff's interim evaluation of the license renewal application related to BFN Units 2 and 3 and concluded that the plant-specific operating experience for BFN Unit 1, by itself, does not fully meet the intent of the license renewal rule. The Committee recommended that the final SER include a cohesive discussion of the applicability of operating experience from Units 2 and 3 to Unit 1 as well as a description of the attributes of the periodic inspection program for the Unit 1 components that will not be replaced. The Committee also recommended that if an extended power uprate is implemented, the staff should require TVA to evaluate the operating experience of Units 1, 2, and 3 at the uprated power level prior to entering the period of extended operation.

III. Proposed Recommendations for Resolving Generic Safety Issue (GSI)-80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of Boiling Water Reactor Mark I and II Containments" (Open)

[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 recommendations for resolving GSI-80.

The staff presented the background on this issue. The issue was identified by the ACRS in 1978 during the operating license reviews of some BWRs. The ACRS posed questions concerning the likelihood and effects of a loss-of-coolant accident (LOCA) which could cause interactions with the Control Rod Drive (CRD) hydraulic lines in such a way as to prevent rod insertion, creating the potential for recriticality when the core is reflooded. The ACRS discussed this conclusion with the staff during its 273rd meeting on January 6, 1983, but remained concerned about MARK I and II containments, which are smaller and more congested than the MARK III containments upon which the staff's analysis was concentrated. Thus, the issue remained open for the MARK I and II containments. Following an analysis of the issue in January 1984, the issue was given a LOW-priority ranking (based on Appendix C of NUREG-0933) and documented in NUREG-0933 (the "original analysis" of this generic issue). Later, it was concluded in NUREG/CR-5382 that consideration of a 20-year license renewal period could change the ranking of the issue to a medium priority. However, further evaluation, using the conversion factor of $2,000/man-rem approved by the Commission in September 1995, resulted in the issue being placed in the DROP category. During site visits associated with GSI-156.6.1, "Pipe Break Effects on Systems and Components," some new piping configurations were discovered that were not considered in the original evaluation of GSI-80. Thus, in March 1998, during a periodic review of LOW-priority GSIs, the Office of Nuclear Reactor Regulation (NRR) indicated that the priority of GSI-80 should be reassessed in light of the concerns of GSI-156.6.1. As a result, a study was conducted by RES to determine the safety significance of the issue.

The staff presented the safety significance of this issue. Recriticality during the course of an accident has no direct effect on the health and safety of the public. However, failure to insert a significant number of control rods could pose two separate safety problems. First, when the core is reflooded by cold emergency core cooling water, the reactor will undergo a cold water reactivity transient if the core is not subcritical. The cold water can insert considerable positive reactivity, which means that portions of the core where control rods failed to insert can return to a significant power level and may even overshoot to power levels considerably higher than those experienced during normal operation. Secondly, the recirculation phase of emergency core cooling is sized to carry away decay heat. If fission heat is not shut off, the Emergency

Core Cooling System (ECCS) may not be sufficient to remove this extra energy, resulting in coolant boil-off, core-melt, and potential containment failure.

The staff's technical assessment described a detailed analysis of the high-energy pipe break interactions documented in preliminary evaluations of Boiling Water Reactor Mark I and Mark II containment power plants for GSI-80. The Core Damage Frequencies (CDFs) for the various Residual Heat Removal and Reactor Coolant System pipe break events that could potentially impact control rod drive piping were determined for Mark I and Mark II plants. All of the calculated CDF values were less than the threshold (10^{-6} event/Reactor Year). Therefore, GSI-80 will be closed with no changes to the existing regulations or guidance.

Committee Action

The Committee issued a letter agreeing with the staff's recommendation to close GSI-80 without any changes to the regulations or guidance.

IV Resolution of ACRS Comments on the Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants" (Open)

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

The Chairman of the Fire Protection Subcommittee provided an introduction to the staff. The Committee had the benefit of presentations and discussions with representatives of the staff and the Nuclear Energy Institute (NEI) regarding the resolution of the ACRS comments in the draft final Regulatory Guide, "Risk-informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," and NEI document, NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)." There was one member of the public from EPM, a fire protection consulting firm, who attended the meeting via teleconference.

In response to the ACRS letter dated June 14, 2005, the staff agreed with all the ACRS recommendations except the one regarding definitions of the Maximum Expected Fire Scenario and Limiting Fire Scenario.

The staff provided a status of the licensee implementation of National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition. Two licensees sent letters of intent to implement NFPA 805. The two licensees intend to transition 12 plants total to NFPA 805. The staff addressed the each of the ACRS six recommendations. The staff stated the regulatory guide and NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)," are being revised to address the ACRS comments.

The staff did not seek endorsement of the regulatory guide at this meeting, because the staff wants to ensure compatibility with other staff documents, such as 10 CFR 50.69, 10 CFR 50.46(a), and 10 CFR 50.48 (c). The staff wants to ensure consistency in probabilistic risk assessment terminology. The staff stated they will provide the final draft regulatory guide and the NEI 04-02 document to the ACRS in December 2005 and the staff will seek endorsement at that time. The Committee told the staff that they would review the regulatory guide in early 2006 provided the staff delivers the revised documents to the ACRS in December 2005.

Committee Action

The Committee did not write a letter at this time based on the staff's request.

V. Davis-Besse Reactor Pressure Vessel Head Integrity Calculations (Open)

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

Mr. John Sieber, the cognizant ACRS member for this issue, introduced this topic to the Committee. Mr. Sieber provided an overview of the topic. He noted that the Committee has reviewed many aspects of the 2002 Davis-Besse events. He discussed three questions that he still had: (1) At what pressure would the vessel head had failed during an accident, (2) How long would the plant have run before the vessel had failed, and (3) What was the likelihood of failure during the year prior to the time of discovery.

NRC Staff Presentation

Mr. Allen Hiser, Office of Nuclear Regulatory Research (RES), discussed the three types of calculations performed. First, the as-found condition was analyzed to determine the margin to failure. Second, the conditions during the year before discovery were analyzed to support the Accident Sequence Precursor (ASP) analysis. Third, the staff performed calculations to support the Significance Determination Process (SDP). He then introduced Dr. Mark Kirk to provide the detailed presentation.

Dr. Kirk reiterated the three analyses that the staff performed (1) as-found, (2) forward-looking, and (3) backward-looking. The as-found analysis was important to benchmark the analytical models, such that they predicted the non-failure of the vessel as of the day of discovery. Dr. Kirk showed several pictures and diagrams of the degradation in the reactor pressure vessel head and cracking in the stainless-steel cladding. He also discussed the measurements and analysis of the hole and the cracks, which indicated no evidence of ductile crack initiation in the cladding.

Dr. Kirk then discussed the methodology for the integrity assessment of the as-found vessel head. As inputs to that assessment, the staff used the geometric configuration of the cavity, the crack sizes and distributions, and several large-scale tests. Material tests on the Davis-Besse vessel head metals indicated that it possessed typical properties of 308 stainless steel. Burst tests performed at Oak Ridge National Laboratory to validate the models indicated that different failure modes would occur with different crack depths. If a substantial crack exists, the material would fail through crack tearing, slowly releasing the pressure. If a very small crack, or no crack, exists the material fails at a higher pressure but with a more severe failure mode.

Dr. Kirk continued by discussing the details of the finite-element model. He stated that three different crack models were used, and focused his discussion on the bounding crack model. Even with that model, the failure pressures at the time of discovery were above both the operating pressure and the relief valve setpoint pressure.

For the forward-looking and backward-looking analysis, the shape of the cavity was simplified to a circle. Dr. Kirk presented analytical calculations to support this assumption. The circular assumption appears to be either realistic or conservative, depending on the failure mechanism. These calculations used statistically distributed toughness and strength properties, some based on engineering judgement. They also used expert opinion to estimate general corrosion properties and LOCA binning rules.

Specifically for the backward-looking ASP analysis, the experts had to predict the state of the cavity one year before the day of discovery. In addition, the staff developed LOCA binning rules based on the capability of the plant's makeup systems and made judgements on how to fit statistical distributions to the expert judgement information. Dr. Kirk then described the analysis to predict LOCA probabilities by size. The small LOCA has the greatest probability due to the forensic investigations that indicate existing cracks and a greater likelihood of failure by crack tearing. The bounding flaw model predicts between 2-22 months existing before failure. The best-estimate median value was five months. The backward-looking analysis predicted a 20% failure probability as of the day of discovery.

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

- Dr. Powers asked about the susceptibility of the stainless steel to concentrated boric acid. In particular, he asked what concentration of boric acid the clad can withstand. Neither Dr. Kirk nor Mr. Hiser could answer but said they would get that information for Dr. Powers.

- Dr. Powers questioned the effects of using a gas-pressurized test versus a hydrostatic test. He asked if a hydrostatic test would have torn a larger hole. Dr. Kirk acknowledged that tendency would exist, but that was not quantitatively examined during the tests. Dr. Kirk also added that calculations indicate that the crack would stabilize and not continue to grow. Dr. Shack predicted that the crack would fishmouth under pressure and create a larger leak. Mr. Hiser and Dr. Kirk acknowledged some growth, but emphasized that the calculations predict a stabilized crack and lower leakage under those conditions.

- Mr. Sieber asked if the actual shape of the cavity had an effect on the calculations. Dr. Kirk indicated that it would probably not make much difference, but the actual geometry was used to ensure an accurate model. Mr. Sieber agreed that the footprint is the most important parameter.

- Several Members discussed the use of expert opinion for corrosion properties. Dr. Powers noted the lack of literature on the stability of borate complexes of iron in solution. Dr. Shack pointed out that the primary uncertainty lies in the unknown temperature and concentration in the cavity. Mr. Sieber added that the temperatures and concentrations are also constantly changing, and therefore so is the corrosion rate. Dr. Kress pointed out that some experts believe that the cavity would not have grown any larger because the hole had reached a size that relieved the pressure and concentration.

- Dr. Ransom asked if the corrosion occurred from the outside-in or the inside-out. Mr. Hiser answered by stating that both options were possible, given that no data exists on the cavity before the day of discovery. Dr. Ransom believes the evidence favors outside-in growth. Mr. Hiser pointed out that other plants did not look closely enough to determine if wastage had begun on the inside. Mr. Sieber added to that point by stating that the other plants had examined well enough to find and repair cracks, thereby stopping any wastage from continuing.

- Dr. Denning asked how the greater likelihood of small LOCAs affected the conditional core damage probability. Mr. Gary DeMoss, RES, answered that the ASP analysis indicated that the risk was dominated by the large break LOCA due to the coincident failure of the sump, though the small LOCA was more likely.

- Dr. Powers and others commented on the overall quality of the work presented, noting that it would score well if subjected to the research quality review.

<u>Committee Action:</u>

This was an information briefing. The Members had requested this briefing during the April 2005 meeting. No further Committee actions are planned on this topic.

VI. <u>Quality Assessment of the Selected NRC Research Program</u> (Open)

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

The NRC Strategic Plan that was developed in accordance with the requirements of the Government Performance and Results Act (GPRA) requires that RES have an independent evaluation of the quality of its research programs. The Committee has agreed to assist RES in assessing the quality of selected research projects. The Committee completed its report on the assessment of the quality of selected research projects on: Station Blackout Risk Evaluation of Nuclear Power Plants; Steam Generator Tube Integrity Program at the Argonne National

Laboratory; and Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag at the Penn State University.

Committee Action

The Committee approved the letter transmitting the report on ACRS Assessment of the Quality of Selected NRC Research Projects to the Director of RES. The Committee anticipates receiving from RES a list of candidate projects for review during the next twelve months.

VII. Licensees' Responses to the Bulletin on "Emergency Preparedness and Response Actions for Security-Based Events" (Open)

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

Dr. Mario Bonaca, the cognizant ACRS member for this issue, introduced the topic to the Committee. Dr. Bonaca provided an overview of the topic. He reminded the Members that they considered the draft bulletin during the 523rd meeting in June 2005, and decided to wait to hear a presentation until responses from the licensees had been received. Dr. Bonaca and Mr. Thornsbury also noted that the bulletin and the presentation were publicly available, but the meeting could be closed if necessary to discuss sensitive information. Dr. Bonaca then asked Mr. Eric Weiss of the Office of Nuclear Security & Incident Response (NSIR) to begin.

NRC Staff Presentation

Before Mr. Weiss began, Mr. Nader Mamish, Director of the Emergency Preparedness Directorate, made a few opening remarks to outline the presentation. Mr. Weiss also introduced Mr. Gregory Casto, the staff member responsible for reviewing the details of the bulletin responses, to help him with the presentation. Mr. Weiss first discussed the background of the bulletin, noting that the emergency planning basis remains valid even after the events of September 11. Though licensees have made many improvements to their emergency response programs in response to NRC actions, additional security-based emergency preparedness (EP) actions could be necessary. Therefore, the agency issued Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events," to gather information in five specific areas: (1) emergency classification levels and emergency action levels; (2) prompt notification to the NRC of security events; (3) licensee onsite protective actions for plant personnel; (4) emergency response organization staff augmentation practices; and (5) security-based event inclusion in the emergency preparedness drill and exercise program.

Mr. Weiss explained that the information in the bulletin does not indicate that additional or earlier radiological protective actions are needed, but it does recognize that a security-based event may not progress in the same way as traditional accidents. All licensees responded promptly and generally provided answers consistent with the information in the bulletin. Mr. Weiss also mentioned that NEI has issued a white paper to the industry with similar information that will be adopted by all licensees.

For the emergency classification and action levels, the changes generally escalate the event to one level higher than existing emergency plans and take into account advance warning information on incoming threats.

Mr. Weiss described the current requirements the licensees have to notify local law enforcement and the NRC. The bulletin additionally requests information regarding licensee plans to briefly notify the NRC earlier in the event to support the notification of other government agencies and licensees. Mr. Weiss also noted that the ACRS made a similar recommendation in late 2003. Rulemaking is being considered to change the regulations such that an earlier notification is required.

Mr. Weiss then discussed the information in the bulletin related to onsite protective actions. These are intended to maximize site personnel safety during emergency conditions through assembly, accountability, and evacuation actions. The bulletin describes additional actions licensees should consider for onsite protective actions during security-based events.

Regarding emergency response organizations (EROs), some licensees indicated that they may not fully activate some elements of their ERO during a security event. The bulletin indicates that it is prudent to have plans for staffing the ERO at an alternate location during such events. Mr. Casto noted that this action is also consistent with a recommendation made by the ACRS in 2004.

For the integration of security-based scenarios into EP exercises, Mr. Weiss described an NEI working group that is organizing the implementation of such drills and exercises. Eventually, security-based scenarios will become part of the regular six-year cycle of licensee EP exercises.

In conclusion, Mr. Weiss described the upcoming activities in this area. A Commission paper was in progress to provide a summary of the licensee responses and recommend regulatory actions. Dialogue will continue with licensees that do not have all the provisions currently in place.

Following the formal presentation and discussion, the meeting was closed to discuss additional details of the emergency action levels and address Members questions which involved sensitive information.

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

- Dr. Powers questioned the trigger for a general emergency when the site was taken over, that it should be sooner - perhaps an imminent takeover. Mr. Casto clarified that takeover of some vital areas is enough to trigger the general emergency, so it may occur before a complete takeover has occurred. Mr. Casto also compared this approach to that taken during traditional accidents, that a general emergency is not declared until control of the plant is lost. In addition, the staff's analysis of the emergency planning basis indicated that the core-melt and release progression remains similar to traditional accidents. Therefore, the staff did not feel that this declaration needed to be stepped up.

- Dr. Denning asked if there was a difference between classification for evacuation purposes versus asking for response dealing with the threat. Mr. Casto confirmed that idea, stating that the response by offsite law enforcement personnel occurs on a different path and is not affected by the event classification.

- Dr. Bonaca added to the discussion on event classification that a general emergency is not declared for traditional accidents until two fission product barriers are lost, with the imminent loss of the third barrier. Therefore, declaration of a general emergency on loss of the plant is still conservative.

- Dr. Apostolakis asked who is in charge during security-based events. Dr. Bonaca and Dr. Powers agreed that the plant people are in charge of the plant. Mr. Mamish verified that the licensee is in charge of actions at the plant, while the Department of Homeland Security would likely be in charge of the off-site response activities.

- Dr. Denning asked if the scope of the drills and exercises is limited to the design-basis threat. Mr. Casto answered that the exercises could go beyond the design basis, consistent with current practices for accident-based scenarios.

- Dr. Bonaca asked about additional actions being taken, such as the staging of equipment, that ACRS had previously discussed. Mr. Weiss noted that other groups within NRC and NSIR are addressing those issues.

Committee Action:

This was an information briefing. No further Committee actions are planned at this time on this specific topic, though the Committee plans to continue its review of selected security-related issues.

VIII. NRC Staff's Response to the ACRS Letter on the Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Open)

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

The Committee heard a presentation by the staff concerning its plans to further revise its proposed Regulatory Guide 1.82, Rev. 4, to incorporate more risk-informed practices to granting containment overpressure credit, and to apply those practices for the first time as part of its review of the Vermont Yankee power uprate request. The Committee questioned how this approach would reconcile its concerns about maintaining sufficient defense-in-depth as part of the design of the containment and emergency core cooling systems. The staff stated that it would require licensees that propose to apply this method to demonstrate that the five key principles of risk-informed decisionmaking in RG 1.174 are met. The staff will clarify its requirements and describe its expectations for licensees who submit risk-informed license amendments to credit containment overpressure. The staff intends to provide the ACRS with a revised Regulatory Guide 1.82 for review and comment.

Committee Action

The Committee plans to review the revised Revision 4 to Regulatory Guide 1.92.

IX. Format and Content of the NRC Safety Research Program Report to the Commission (Open)

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

During the October 6-8 2005 ACRS meeting, the Committee discussed the format and content of the ACRS biennial report to the Commission on review and evaluation of the NRC safety research program.

Committee Action

The Committee plans to discuss a draft report on NRC safety research program during its November 3-5, 2005 meeting.

X. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

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

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

- The Committee considered the EDO's response of September 12, 2005, to ACRS's July 15, 2005 report responding to an April 25, 2005 Staff Requirement Memorandum (SRM), requesting the Committee to "provide the Commission a list of research projects it intends to review in the short term as part of its assessment of research quality, with an indication of the methodology the Committee will use for the reviews."

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

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 October 6, 2005. The following items were discussed:

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through December 2005 was discussed. The objectives were:

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

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

Meeting with the NRC Commissioners

The ACRS is scheduled to meet with the NRC Commissioners between 1:00 and 3:00 p.m. on Thursday, December 8, 2005 to discuss items of mutual interest. A list of topics (noted below) was approved on October 4, 2005, by the Commission:

- Issues Related to New Plant Licensing (including technology Neutral Framework) (TSK/MME)
- Proposed Alternative Embrittlement Criteria in 10 CFR 50.46 (DAP/RC)
- Fire Protection Matters (GEA/JGL)
- Power Uprate Technical Issues (RSD/RC)

In addition to the above topics, the ACRS Chairman will provide an overview.

Proposed ACRS Meeting Dates fo CY 2006

Proposed ACRS meeting dates for CY 2006 were agreed to by the members and are listed below:

Meeting No.	Meeting Dates
-	January (No Meeting)
529	February 9-11, 2006
530	March 9-11, 2006
531	April 6-8, 2006
532	May 4-6, 2006
533	May 31 - June 1-2, 2006 *
534	July 12-14, 2006 *
-	August (No Meeting)
535	September 7-9, 2006
536	October 4-6, 2006 *
537	November 1-3, 2006 *
538	December 7-9, 2006

* Wednesday - Friday

ACRS Retreat in 2006

During the September 2005 meeting, the Committee decided to hold a retreat on January 26-27, 2006. The members were requested to propose topics for the retreat by September 23, 2005. Comments were received from several members.

The ACRS Executive Director supports having a retreat in January 2006. It will be valuable to discuss a number of issues related to Committee operations. These issues include:

(i) How should the ACRS handle any significant workload increase in FY 06 and 07? Adding one or two additional meetings does not seem practical as few members already approach the maximum allowed 130 day limit per year. The Committee should look at various options, including expanding the ACRS to 15 members and adding a new Subcommittee or adding more Saturday sessions.

(ii) Each Subcommittee Chairman should take a few minutes and talk about their forecast of future work for the coming year and whether or not they foresee any emerging issues of significance.

(iii) The Committee should take some time and discuss what technical expertise is needed on the ACRS in the future. Also the ACRS should be more proactive in the search for future members and find ways to have these individuals auditioned prior to recommending for membership on the ACRS. May be there should be a standing Subcommittee for potential new ACRS members.

(iv) The Committee should take some time to discuss the upcoming Quadripartite Meeting in October '06, including the presentations and planned events.

Staff Requirements Memorandum on Policy Issues Related to New Plant Licensing

In a Staff Requirements Memorandum (SRM) dated September 14, 2005, (pp. 34) the Commission stated that the ACRS should provide its views on the two policy issues (SECY-05-0130) related to new plant licensing, including the feasibility of alternatives to the QHOs as technology-neutral risk objectives. The staff should then consider ACRS comments in developing a subsequent notation vote paper addressing these policy issues.

The Committee issued a report to the Commission on these policy issues in September 2005. However, the Committee did not explicitly address the issues raised by the Commission in the SRM.

Candidates for Potential Membership on the ACRS (Closed)

On June 28, 2005, the Screening Panel met to discuss 42 applications received in response to the solicitation for the current vacancies on the ACRS. The Panel selected six applicants in the areas of plant operations and materials and metallurgy. These candidates were interviewed by the members and the Screening Panel during the September ACRS meeting. The Panel is in the process of preparing a report to the Commission recommending a number of candidates to fill the vacancy in the area of materials and metallurgy. The Screening Panel will continue to look for qualified candidates to fill the vacancies on the Committee in the areas of thermal-hydraulics and plant operations.

Response to Ms. Nancy Burton, Connecticut Coalition Against Millstone Regarding Millstone Units 2 and 3 License Renewal Application

In a letter to Dr. Wallis, ACRS Chairman, dated September 7, 2005, Ms. Nancy Burton, Connecticut Coalition Against Millstone, requested that the ACRS defer its decision regarding the Millstone license renewal application until after the State of Connecticut has had an opportunity to provide its input. She also made statements during the Plant License Renewal Subcommittee meeting on April 6, 2005 and the full Committee meeting on September 8, 2005. In her letter, she listed several issues that were not addressed in the staff's final safety evaluation report related to Millstone Units 2 and 3 license renewal application.

It has been the Committee's practice to respond to individuals who sent have letters to the ACRS Chairman. During the September meeting, Mr. Santos, ACRS staff engineer, informed the Committee about sending a response to Ms. Burton. Since she raised other issues related to the adequacy of the staff's final safety evaluation report, it would be appropriate to refer these issues to the EDO for possible action.

Summary Matrix of ACRS Reports and Letters

As directed by the Commission, a summary matrix of ACRS reports and letters issued in FY 05 should be submitted along with the Operating Plan. The Operating Plan and the Summary Matrix are due to the Commission on December 30, 2005. In order to preclude violation of the ACRS Bylaws, the Committee should authorize the ACRS Executive Director and/or his designee to summarize the ACRS reports and letters issued in FY 05.

Member Issues

NRR Office Instruction on Risk-Informed Review Process for Emergent Issues

NRR has recently issued for trial use an internal Office instruction on risk-informed review process for emergent issues. This process was developed in response to the GAO recommendations included in its May 2004 report on NRC's handling of reactor vessel head corrosion at Davis-Besse. In its report, GAO stated that NRC should improve its use of PRA estimates in decisionmaking by:

- Ensuring that the risk estimates, uncertainties, and assumptions made in developing the estimates are fully defined, documented, and communicated to NRC decisionmakers

- Providing guidance to decisionmakers on how to consider the relative importance, validity, and reliability of quantitative risk estimates in conjunction with other quantitative safety-related factors.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 527th ACRS Meeting, November 3-5, 2005.

The 526th ACRS meeting was adjourned at 7:00 p.m. on October 7, 2005.

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

November 30, 2005

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

FROM: Graham B. Wallis
ACRS Chairman

Graham B. Wallis

SUBJECT: CERTIFIED MINUTES OF THE 526th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), OCTOBER 6-8, 2005

I certify that based on my review of the minutes from the 526TH 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.

the scope of matters to be discussed at this public meeting.

At the conclusion of the scoping process, the NRC will prepare a concise summary of the determination and conclusions reached, including the significant issues identified, and will send a copy of the summary to each participant in the scoping process. The summary will also be available for inspection in ADAMS at *http://www.nrc.gov/reading-rm/adams.html.* The NRC staff will then prepare and issue for comment the draft supplement to the GEIS, which will be the subject of separate notices and separate public meetings at a later time. Copies will be available for public inspection at the above-mentioned addresses, and one copy per request will be provided free of charge. After receipt and consideration of the comments, the NRC will prepare a final supplement to the GEIS, which will also be available for public inspection.

Information about the proposed action, the supplement to the GEIS, and the scoping process may be obtained from Dr. Masnik at the aforementioned telephone number or e-mail address.

Dated at Rockville, Maryland, this 16th day of September 2005.

For the Nuclear Regulatory Commission.

Samson S. Lee,

Acting Program Director, License Renewal and Environmental Impacts Program, Division of Regulatory Improvement Programs, Office of Nuclear Reactor Regulation.

[FR Doc. 05–18915 Filed 9–21–05; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on October 6–8, 2005, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Wednesday, November 24, 2004 (69 FR 68412).

Thursday, October 6, 2005, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10 a.m.: Interim Review of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Open)—The Committee will hear presentations by and hold discussions with representatives of the Tennessee Valley Authority and the NRC staff regarding the license renewal application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 and the NRC staff's Safety Evaluation Report with Open Items.

10:15 a.m.–11:45 a.m.: Proposed Recommendations for Resolving Generic Safety Issue (GSI)–80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of Boiling Water Reactor Mark I and II Containments" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the recommendations proposed by the NRC Office of Nuclear Regulatory Research for resolving GSI–80.

12:45 p.m.–2:15 p.m.: Resolution of ACRS Comments on the Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding the changes made to this Guide and to NEI 04–02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," in response to the ACRS comments and recommendations included in its June 14, 2005 letter.

2:30 p.m.–4 p.m.: Davis-Besse Reactor Pressure Vessel Head Integrity Calculations (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the expert elicitation and calculations performed for the reactor pressure vessel head integrity of the Davis-Besse Nuclear Power Plant.

4:15 p.m.–5:15 p.m.: Quality Assessment of the Selected NRC Research Program (Open)—The Committee will discuss the results of the cognizant ACRS panel's assessment of the quality of the NRC research projects on: Standardized Plant Analysis Risk (SPAR) Models Development Program; Steam Generator Tube Integrity Program at the Argonne National Laboratory; and the Thermal-Hydraulic Test Program at the Penn State University.

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

Friday, October 7, 2005, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10 a.m.: Licensees' Responses to the Bulletin on, "Emergency Preparedness and Response Actions for Security-Based Events" (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding licensees' responses to the Bulletin related to Emergency Preparedness and Response Actions for Security-Based Events.

10:15 a.m.–11:15 a.m.: NRC Staff's Response to the ACRS Letter on the Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the staff's response to the ACRS letter on the Proposed Revision 4 to Regulatory Guide 1.82.

11:15 a.m.–12:15 p.m.: Format and Content of the NRC Safety Research Program Report to the Commission (Open)—The Committee will hear a report by and hold discussions with the Chairman of the ACRS Subcommittee on Safety Research Program regarding the format and content of the ACRS report to the Commission on the NRC Safety Research Program as well as assignments for the ACRS members.

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

2:15 p.m.–2:30 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

2:30 p.m.–3 p.m.: Subcommittee Report (Open)—The Committee will hear a report by and hold discussions with the Chairmen of the ACRS Subcommittees on Plant Operations and

Plant License Renewal regarding matters discussed at the September 21, 2005 Subcommittee meeting.

3:15 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, October 8, 2005, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

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

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

In accordance with Subsection 10(d) Public Law 92–463, I have determined that it is necessary to close portions of this meeting noted above to discuss and protect information classified as national security information and safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3).

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: September 16, 2005.

Annette L. Vietti-Cook,

Secretary of the Commission.

[FR Doc. 05–18913 Filed 9–21–05; 8:45 am]

BILLING CODE 7590–01–P

RAILROAD RETIREMENT BOARD

Agency Forms Submitted for OMB Review

Summary: In accordance with the Paperwork Reduction Act of 1995 (44 U.S.C. Chapter 35), the Railroad Retirement Board (RRB) has submitted the following proposal(s) for the collection of information to the Office of Management and Budget for review and approval.

Summary of Proposals

(1) *Collection title:* Evidence for Application of Overall Minimum.

(2) *Form(s) submitted:* G–319, G–320.

(3) *OMB Number:* 3220–0083.

(4) *Expiration date of current OMB clearance:* 1/31/2007.

(5) *Type of request:* Revision of a currently approved collection.

(6) *Respondents:* Individuals or households, Business or other for-profit, Non-profit institutions.

(7) *Estimated annual number of respondents:* 475.

(8) *Total annual responses:* 475.

(9) *Total annual reporting hours:* 170.

(10) *Collection description:* Under Section 3(f)(3) of the Railroad Retirement Act, the total monthly benefits payable to a railroad employee and his family are guaranteed to be no less than the amount which would be payable if the employee's railroad service had been covered by the Social Security Act.

Additional Information or Comments: Copies of the forms and supporting documents can be obtained from Charles Mierzwa, the agency clearance officer (312–751–3363 or *Charles.Mierzwa@RRB.GOV).*

Comments regarding the information collection should be addressed to Ronald J. Hodapp, Railroad Retirement Board, 844 North Rush Street, Chicago, Illinois, 60611–2092, or *Ronald.Hodapp@RRB.GOV* and to the OMB Desk Officer for the RRB, at the Office of Management and Budget, Room 10230, New Executive Office Building, Washington, DC 20503.

Charles Mierzwa,

Clearance Officer.

[FR Doc. 05–18928 Filed 9–21–05; 8:45 am]

BILLING CODE 7905–01–P

SECURITIES AND EXCHANGE COMMISSION

Sunshine Act Meetings

Notice is hereby given, pursuant to the provisions of the Government in the Sunshine Act, Public Law 94–409, that the Securities and Exchange Commission will hold the following meetings during the week of September 26, 2005:

A closed meeting will be held on Thursday, September 29, 2005 at 3 p.m.

Commissioners, Counsel to the Commissioners, the Secretary to the Commission, and recording secretaries will attend the closed meeting. Certain staff members who have an interest in the matters may also be present.

The General Counsel of the Commission, or his designee, has certified that, in his opinion, one or more of the exemptions set forth in 5 U.S.C. 552b(c)(3), (5), (7), (9)(B), and (10) and 17 CFR 200.402(a)(3), (5), (7), (9)(ii) and (10) permit consideration of the scheduled matters at the closed meeting.

Commissioner Atkins, as duty officer, voted to consider the items listed for the closed meeting in closed session.

The subject matters of the closed meeting scheduled for Thursday, September 29, 2005 will be:

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

September 15, 2005

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

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

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

2) 8:35 - ~~10:00~~ 9:40 A.M. Interim Review of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Open) (MVB/CS)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the Tennessee Valley Authority and the NRC staff regarding the license renewal application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 and the NRC staff's Safety Evaluation Report with Open Items.

~~10:00~~ 9:40 - 10:15 A.M. *****BREAK*****

3) 10:15 - ~~11:45~~ 11:20 A.M. Proposed Recommendations for Resolving Generic Safety Issue (GSI)-80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of Boiling Water Reactor Mark I and II Containments" (Open) (JDS/JGL)
- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the NRC staff regarding the recommendations proposed by the NRC Office of Nuclear Regulatory Research for resolving GSI-80.

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

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

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

4.1) Remarks by the Subcommittee Chairman

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

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

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

5.1) Remarks by the Subcommittee Chairman

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

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

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

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

6.1) Remarks by the Subcommittee Chairman

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

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

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

Discussion of proposed ACRS reports on:

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

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

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

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

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

9) 8:35 - 10:00 A.M. Licensees' Responses to the Bulletin on, "Emergency Preparedness and Response Actions for Security-Based Events" (Open/Closed) (MVB/EAT)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding licensees' responses to the Bulletin related to emergency Preparedness and Response Actions for Security-Based Events.

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

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

10) 10:15 - 11:15 A.M. NRC Staff's Response to the ACRS Letter on the Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Open) (VHR/GBW/RC)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's response to the ACRS letter on the Proposed Revision 4 to Regulatory Guide 1.82.

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

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

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

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

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

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

13) 2:15 - 2:30 P.M. 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.

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

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

15) 3:15 - ~~7:00~~ 5:55 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

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

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

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

15.4) Quality Assessment of Selected NRC Research Projects (DAP/GEA/JDS/GBW/HPN/CS/RC)

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

16)	8:30 - 12:00 Noon	Preparation of ACRS Reports (Open) Continue discussion of the proposed ACRS reports listed under Item 15.
17)	12:00 - 12:30 P.M.	Miscellaneous (Open) (GBW/JTL) Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

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

APPENDIX III: MEETING ATTENDEES

526TH ACRS MEETING
October 6-8, 2005

NRC STAFF (10/6/2005)

D. Frumkin, NRR
M. Kirill, RES
A. Hiser, RES
N. Chokshi, RES
G. Demoss, RES
A. Lee, RES
C. Ader, RES
T. Mintz, RES
D. Marksberry, RES
J. Mitchell, RES
J. Muscara, RES
C. Julian, RII
A. Pal, NRR
T. Nazario, RII
D. Jay, NRR
K. Tanabe, NRR
P. T. Kuo, NRR
M. Marshall, NRR
J. Zimmerman, NRR
M. Hartzman, NRR
R. Auluck, NRR
F. Gillespie, NRR
G. Cheruvenki, NRR
M. Chernoff, NRR
S. Mitra, NRR
Y.C. Li, NRR
R. McNally, NRR
B. Rogers, NRR
K. Parczewglei, NRR
A. Hodgdan, OGC
S. Dinsmore, NRR
B. Elliot, NRR
G. Galuki, NRR
N. Iqbal, NRR
A. Lee, NRR
E. Brown, NRR
H. Vandermolen, RES
J. Rosenthal, RES
T. Navedo, RES
J. Ibarra, RES
S. Ali, RES
T. Le, NRR
J. Dixon-Herrity, OEDO
A. Sheiku, RES
S. Weerakkody, NRR
B. Radlinski, NRR
R. Gallucci, NRR
P. Lavin, NRR
J. Lyons, NRR
G. Parry, NRR
J. Hyslop, RES
D. Szware, RES
S. M. Wong, NRR
M. Tschultz, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

R. Rucker, First Energy
S. Dost, First Energy

APPENDIX IV: FUTURE AGENDA

November 18, 2005

REVISED
SCHEDULE AND OUTLINE FOR DISCUSSION
528th ACRS MEETING
DECEMBER 7-10, 2005

WEDNESDAY, DECEMBER 7, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 1:00 - 1:05 P.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)
   - 1.1) Opening statement
   - 1.2) Items of current interest

2) 1:05 - 3:00 P.M. Final Review of the Vermont Yankee Extended Power Uprate Application and the Associated Safety Evaluation (Open) (RSD/GBW/RC)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the Entergy Nuclear Operations, Inc. and the NRC staff regarding the 20% power uprate application for the Vermont Yankee Nuclear Plant and the NRC staff's associated Safety Evaluation.

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

3:00 - 4:00 P.M. ***BREAK***

3) 4:00 - 5:45 P.M. Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN/SD)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Discussion of the draft ACRS report to the Commission on the NRC Safety Research Program.

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

4) 6:00 - 6:45 P.M. Preparation for Meeting with the NRC Commissioners (Open) (GBW, et. al/JTL, et. al)
   Discussion of the following topic scheduled for discussion with the NRC Commissioners on December 8, 2005:
   - I Overview (GBW)
     - License Renewal
     - Early Site Permits
     - Future ACRS Activities
   - II Issues Related to New Plant Licensing (including Technology-Neutral Framework) (TSK)

III Proposed Alternative Embrittlement Criteria in 10 CFR 50.46 (DAP)
IV Fire Protection Matters (GEA)
V Power Uprate Technical Issues (RSD)

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

5) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)

6) 8:35 - 10:15 A.M. Early Site Permit Application for the Grand Gulf Nuclear Station and the Associated Final Safety Evaluation Report (Open) (DAP/MME)

6.1) Remarks by the Subcommittee Chairman

6.2) Briefing by and discussions with representatives of the System Energy Resources, Inc. and the NRC staff regarding the early site permit application for the Grand Gulf Nuclear Station and the associated final Safety Evaluation Report prepared by the NRC staff.

10:15 - 10:30 A.M. ***BREAK***

7) 10:30 - 11:45 A.M. Draft Final Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations" (Open) (RSD/JGL)

7.1) Remarks by the Subcommittee Chairman

7.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final Generic Letter on "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations" and a summary of the NRC staff's resolution of public comments received on the public comment version of this Generic Letter.

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

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

8) 1:00 - 3:00 P.M. Meeting with the NRC Commissioners, Commissioners' Conference Room, One White Flint North, Rockville, MD (Open) (GBW, et. al/JTL, et. al)
Meeting with the NRC Commissioners to discuss the topics listed under Item 4.

3:00 - 3:30 P.M. ***BREAK***

9) 3:30 - 5:00 P.M. Proposed Program Plan and Advance Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50 (Open) (WJS/GEA/MRS/EAT)

9.1) Remarks by the Subcommittee Chairman
9.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed Program Plan and the Advance Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50, and related matters.

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

5:00 - 5:15 P.M. *****BREAK*****

10) 5:15 - 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

10.1) Final Review of the Extended Power Uprate Application for the Vermont Yankee Nuclear Plant (RSD/GBW/RC)
10.2) Final Review of the Early Site Permit Application for the Grand Gulf Nuclear Station (DAP/MME)
10.3) Draft Final Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations" (RSD/JGL)
10.4) Proposed Program Plan and Advance Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50 (WJS/GEA/MRS/EAT)

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

11) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)

12) 8:35 - 10:00 A.M. Staff Activities Associated with Responding to the Commission's Staff Requirements Memorandum (SRM) related to Safety Conscious Work Environment and Safety Culture (Open) (MVB/GEA/JHF)

12.1) Remarks by the Subcommittee Chairman
12.2) Briefing by and discussions with representatives of the NRC staff regarding staff activities associated with responding to the Commission's SRM related to safety conscious work environment and safety culture, and related matters.

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

	10:00 - 10:15 A.M.	*****BREAK*****
13)	10:15 - 11:15 A.M.	Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD) 13.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings. 13.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.
14)	11:15 - 11:30 A.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.
15)	11:30 - 12:00 Noon	Election of ACRS Officers for CY 2006 (Open) (JTL/SD) Election of Chairman and Vice Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee.
	12:00 - 1:30 P.M.	*****LUNCH*****
16)	1:30 - 3:30 P.M.	Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN/SD) Discussion of the draft ACRS report to the Commission on the NRC Safety Research Program.
	3:30 - 3:45 P.M.	*****BREAK*****
17)	3:45 - 6:45 P.M.	Preparation of ACRS Reports (Open) Discussion of proposed ACRS reports listed under Item 10.

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

18)	8:30 - 12:30 P.M.	Preparation of ACRS Reports (Open) Continue discussion of the proposed ACRS reports listed under Item 10.

19) 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.**
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APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
526th ACRS MEETING
October 6-8, 2005

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

MEETING HANDOUTS

AGENDA ITEM NO.	DOCUMENTS
1	Opening Remarks by the ACRS Chairman
	1. Items of Interest, dated October 6-8, 2005
2.	Interim Review of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3
	2. Tennessee Valley Authority Browns Ferry Nuclear Plant - License Renewal
	3. Browns Ferry Nuclear Plant Units 1, 2, and 3 License Renewal Safety Evaluation Report
3.	Proposed Recommendations for Resolving Generic Safety Issue (GSI)-80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of Boiling Water Reactor Mark I and II Containments"
	4. Generic Issue 80 Pipe Break Effects on CRD Hydraulic Lines in BWRs
	5. [Viewgraphs]
4.	Resolution of ACRS Comments on the Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants"
	6. Regulatory Guide for NFPA 805 Rule - Paul Lain
	7. Regulatory Guide for NFPA 805 Rule - Bob Radlinski
	8. Regulatory Guide for NFPA 805 Rule - Sunil Weerakkody
	9. [Viewgraphs]
5.	Davis-Besse Reactor Pressure Vessel Head Integrity Calculations
	10. An Assessment of the Structural Integrity Challenge Posed by Boric Acid Wastage in the Davis Besse RPV Head
	11. Revised Proposed Schedule
	12. Memorandum from A. Thadani to W. Travers (handout)

6. <u>Quality Assessment of the Selected NRC Research Program</u>

9. Licensees' Response to the Bulletin on, "Emergency Preparedness and Response Actions for Security-Based Events"

    13. Status of BL 2005-02 Responses
    14. Revised Proposed Schedule

10. NRC Staff's Response to the ACRS Letter on the Proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident"

    15. Regulatory Guide 1.82 Revision 4

11. Format and Content of the NRC Safety Research Program Report to the Commission

12. <u>Future ACRS Activities/Report of the Planning and Procedures Subcommittee</u>
    16. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - October 5, 2003 [Handout #16]

13. <u>Reconciliation of ACRS Comments and Recommendations</u>
    17. Reconciliation of ACRS Comments and Recommendations [Handout #1]

Appendix IV
526th ACRS Meeting

MEETING NOTEBOOK CONTENTS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
524th FULL COMMITTEE MEETING

October 6-8, 2005

TODAY'S DATE: October 6, 2005

NRC STAFF - PLEASE SIGN BELOW

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NAME	NRC ORGANIZATION
Stephen Dinsmore	NRR/DSSA/SPSB
BARRY ELLIOT	NRR/DE/EMCB
Greg Galetti	NRR/DIPM
Naeem Iqbal	NRR/DSSA/SPLB
Arnold Lee	NRR/DE/EMEB
Eva Brown	NRR/DLPM/LPD2B
Harold Vandermolen	RES/DSARE/ARREB
Jack Rosenthal	RES/DSARE/ARREB
Tania Martinez Navedo	RES/DSARE/ARREB
Jose Ibarra	RES/DSARE/ARREB
SYED ALI	RES/DET
Tommy Le	NRR/DRIP
JENNIFER DIXON-HERRITY	EDO
ABDUL SHEIKH	RES/DET/ERAB
Sunil Weerakkody	NRR/DSSA/SPLB
Bob Radlinski	NRR/DSSA/SPLB
Ray Gallucci	NRR/DSSA/SPLB
Paul Lain	NRR/DSSA/SPLB
Tim Lyons	NRR/DSSA
Gareth Parry	NRR/DSSA
J S Hyslop	RES/PRAB
Dariusz Szwarc	RES/PRAB
See Meng Wong	NRR/~~DSSA~~/SPSB
Michael Tschiltz	NRR/DSSA/SPSB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
524th FULL COMMITTEE MEETING

October 6-8, 2005

TODAY'S DATE: October 6, 2005

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NAME	NRC ORGANIZATION
CAUDLE JULIAN	NRC REGION II
Amar Pal	NRR/DE/EEIB
Tony Nazario	Region II
[illegible]	NRR/DE/EMEB
John Hannon	NRR/DSSA/SPLB
Kiyoto Tanabe	NRC/LR/Japanese Assignee ~~NSC~~ MSA
P T KUO	NRR/DRIP/RLEP
Michael Marshall	NRR/DLPM.
Jake Zimmerman	NRR/DRIP/RLEP-B
MARK HARTZMAN	NRR/DE/EMEB
Roger B Rucker	First Energy
Steve Dort	First Energy
R. Auluck	NRR/RLEP/
Frank Gillespie	NRR
Ganesh cheruvenki	NRR/DE
Margaret Chernoff	NRR/DLPM
S. K. MITRA	NRR/DRIP/RLEP
Y. C. (Renee) Li	NRR/DE/EMEB
R. McNally	NRR/DE/EMEB
Bill Rogers	NRR/DIPM
K. Parczewski	NRR/DE/EMEB
Ann Hodgdon	OGC/RP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th FULL COMMITTEE MEETING
October 6-8, 2005

October 6, 2005
Today's Date

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NAME	AFFILIATION
Kevin L. Groom	TVA
Henry L. Jones	TVA
Joe Valente	TVA
Russell Jansen	TVA
Ken Brune	TVA
Lenny Beller	Progress Energy
Roger Jennings	TVA
Don Arp	TVA
David Bunne	TVA
Mark Grantham	Progress Energy
Michael Heath	Progress Energy
Robert J. Moll	TVA
Richard A DeLong	TVA
Mickey R. Hamby	TVA
Bill Crouch	TVA
Joe McCarthy	TVA
Kazunobu Sakamoto	JNES
Kathryn Sutton	Morgan Lewis
Alex Marion	NEI

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th FULL COMMITTEE MEETING
October 6-8, 2005

October 6, 2005
Today's Date

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NAME	AFFILIATION
Biff Bradley	NEI
John Biechman	NFPA
Steven Dolley	[illegible] NRC
Deann Raleigh	US Scietech
Daniel HORNER	McGraw-Hill

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th FULL COMMITTEE MEETING

October 6-8, 2005

TODAY'S DATE: October 7, 2005

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NAME	NRC ORGANIZATION
ERIC WEISS	NSIR/DPR/EPD
Nader Mamish	NSIR/DPR/EPD
Tad Marsh	NRR/DLPM
Cornelius Holden	NRR/DLPM
MARTY STUTZKE	NRR/DSSA/SPSB
BRIAN SHERON	NRR/ADPT
Jim Lyons	NRR/DSSA
John Lehning	NRR/DSSA
Margie Kotzalas	NRR/DSSA
Lynn Mrowka	NRR/DSSA
Rich Ennis	NRR/DLPM
See Meng Wong	NRR/DSSA/SPSB
Hanry Wagage	NRR/DSSA/SPLB
DAVID SOLORIO	NRR/DSSA/SPLB
SHERWIN TURK	OGC
Mark Kowal	NRR/DSSA/SPLB
Mark Rubin	NRR/DSSA/SPSB
ROB TREGONING	RES/DET/ERAB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th FULL COMMITTEE MEETING

October 6-8, 2005

TODAY'S DATE: October 7, 2005

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th FULL COMMITTEE MEETING

October 6-8, 2005

TODAY'S DATE: October 7, 2005

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th FULL COMMITTEE MEETING
October 6-8, 2005

October 7, 2005
Today's Date

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NAME	AFFILIATION
Benton Ives-Halperin	Reporter - CQ
Jennifer Dixon-Herrity	EDO
Tamara Bloom	EDO
Sarah Hofmann	Vermont Dept. of Public Service
William Sherman	" " " "
Kenneth Frederick	First Energy Corp.
Colin Keller	First Energy Corp
Craig Nichols	Entergy Vermont Yankee
Scott Vance	Pillsbury Winthrop Shaw Pittman
~~[illegible]~~ Steven Dolley	~~[illegible]~~ NRC
~~Rob Tregoning~~	
Kathryn Sutton	Morgan, Lewis & Bockius

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th FULL COMMITTEE MEETING
October 6-8, 2005

October 7, 2005
Today's Date

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th FULL COMMITTEE MEETING
October 6-8, 2005

October 7, 2005
Today's Date

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ITEMS OF INTEREST

526th ACRS MEETING

OCTOBER 6-8, 2005

ITEMS OF INTEREST

526th ACRS MEETING

OCTOBER 6-8, 2005

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th MEETING
October 6-8, 2005

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ON-LINE NEWS ARTICLES

NRC NEWS

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

No. S-05-012

The Role of Risk Management in Regulation
(Where we are and where we should be going.)

International Topical Meeting on Probabilistic Safety Analysis
(PSA '05)
September 12, 2005
Nils J. Diaz
Chairman, U.S. NRC

Good morning, it is a pleasure to be here among so many believers in Probabilistic Risk Assessment, and hopefully a few skeptics. Everybody is needed. I do appreciate the opportunity to reach out to a group of experts, surely with diverse views but with many common interests.

I want to start by thanking Dr. John Garrick, Bernard Fourest, and Professor Kondo, who are serving as the General Chairs of the meeting, for their invitation to speak here today. I also want to acknowledge all those who have worked to make this topical meeting possible. I am particularly pleased at the high-level of interest and participation in this meeting by the international nuclear community. My remarks today represent my personal views on the progress that has been made and a path that lies before us for broadening and accelerating the incorporation of risk analysis and risk insights into the regulation, design, operation, and maintenance of nuclear power reactors. This is why I used "Risk Management" as a marquee. The use of risk analysis and risk insights is already a common decision-making tool. I believe it has to go beyond and become an important management tool. I am confident in its worth for achieving safety and reliability as a day-to-day management tool; moreover, its full potential can be realized when it becomes a cornerstone of strategic management decisions.

Ten years ago, in 1995, the Nuclear Regulatory Commission issued a Policy Statement supporting the increased use of Probabilistic Risk Assessment (PRA), in the words of the policy statement, "in all regulatory matters." That was a significant milestone in the history of reactor regulation because the word "all" was added to the statement by the Commission. Since that time, much progress has been made and important steps have been taken, yet the vision of a broadly re-focused risk-informed regulatory program, permeating all the important safety issues for nuclear reactors, is yet to be achieved. I believe this is the right time to expand and accelerate the implementation of the 1995 Commission Policy. Therefore, I am proposing the full implementation of the Commission's Policy Statement; it should result in a predictable and timely regulatory approach, one that integrates and optimizes reactor safety, security, and preparedness through risk management.

It must use the best available information from operating experience and research, the best available techniques, including risk-informed and performance-based regulation; and it must resolve the relevant issues in the right progression and be realistic and implementable. And, I would expect a strong debate on how to implement and communicate the changes needed to achieve an effective risk-informed regulatory framework.

Let me state the obvious. I am sure no one wants, and I certainly do not want, to put resources in dispositioning risk-insignificant issues. We must use our resources in resolving and then managing the risk-significant issues. But the obvious needs to be driven by a commitment to achieve results.

If today's safety, security, and preparedness issues, a new triplet, are to be addressed in the most effective and efficient manner, the NRC must shift focus and resources in order to enable corresponding changes in our licensees. We cannot afford to remain captive to those out-dated issues and approaches that experience has proven to be unimportant or ineffective. There is a better path and it needs to be traveled.

I have spoken and acted on the use of realistic-conservatism for nuclear regulation: "conservative in the sense of preserving safety margins and realistic in the sense of being anchored in the real world of physics, engineering, and experience." I added: "I see realism and conservatism as enabling factors for safety and reliability," and I see the use of PRA enabling all of the above. I have been surprised by the wide-spread acceptance of the need to adopt realistic-conservatism as a mode of operations. It is a simple yet powerful approach to regulatory decisionmaking. There is no larger, or more obvious need for a realistically-conservative mode of operations, than a fully risk-informed regulatory framework. It would move us a long way toward achieving effective and efficient regulatory operations.

In fact, making use of this opportunity, let me ask some loaded questions:

Is a nuclear power plant using a state-of-the-art, full scope PRA safer and more capable of reliable operation?

What are the risks and operational safety limitations of not using a state-of-the-art PRA? What are the benefits?

I frequently hear opposition to a risk-informed framework because of the uncertainties in PRA. Granted, we need to work at it. But, what will the overall uncertainties be without it?

PRA/PSA is an integral technique to propagate safety and reliability. It can address safety, security and preparedness, and the issues and uncertainties in those areas, and it should.

Both the NRC and the industry have many decisions to make and make them in a dynamic environment, where change is expected and sound results are demanded. We must recognize and accept that we all will have to think long and hard about over where to draw the line between the important and the unimportant, between appropriate margins and wasteful margins, and between preserving defense-in-depth in a risk-significant domain, and abandoning it. In fact, PRA/PSA is the tool to provide balanced decision-making for all of the above.

I believe that there are compelling safety arguments for change. Forty years of operating experience; thirty years of probabilistic risk-assessment; recent electrical grid problems; challenging hurricanes; and terrorist threats present compelling arguments for change because they paint the same picture.

They show us that:

- Station Blackouts,
- Small Loss of Coolant Accidents,
- Feedwater Transients,
- Steam Generator Tube Ruptures,
- Fires and External Events <u>are important</u>

but:

- Large LOCAs,
- Locked Rotors,
- Rod Ejections,
- Steam Line Breaks and Loss of Flow, <u>are not</u>.

Experience and risk-assessment upon risk-assessment have shown the importance of:

- diesel generator and electrical bus reliability,
- common cause failure potential,
- reactor protection system reliability,
- turbine-driven systems, auxiliary feedwater, RCIC and HPCI,
- switch-over to ECCS recirculation,
- service water and other support systems,
- severe accident management capabilities,
- reactor coolant pump seal performance, and last but not least
- materials degradation.

Furthermore, PRA has a large role to play in resolving the safety and security interface. Since September 11, 2001, we have dedicated substantial effort and resources to studying terrorist threats, and we have taken many actions. I cannot provide the details of these studies and actions because they involve Safeguards and/or Classified Information that we do not want our adversaries to obtain. But I can repeat what I said in a speech last year:

> " ... security concerns, including terrorist threats, raise many of the same issues involved in avoiding and mitigating reactor accidents. Potential initiating events, safety functions, safety (and often non-safety) equipment and procedures, and design basis and severe accident management guidelines all converge to a simple postulate: shut down the reactor, cool the core, and maintain barrier integrity. These are things we know how to do well and should be able to do regardless of the initiating event." We know how to do them better because of the use of PRAs.

If fact, last Friday in response to Commission directions, the Secretary of the Commission issued an SRM, a Staff Requirements Memorandum, on the safety and security for new reactors, one aimed at bringing the design-related security issues to the forefront of the design phase. To be a little more specific, optimizing safety with respect to reactor accidents, with emphasis on Station Blackouts, Small Loss of Coolant Accidents, Feedwater Transients, Fires and External Events will also optimize safety and security for addressing terrorist threats. The Commission-issued SRM on the safety and security interface for new reactors incorporates many of the lessons learned in this arena.

Each one of these insights provides a good basis for change but does not individually represent a compelling reason for change. The compelling reason for change emerges when the inter-relationships among the requirements, issues and safety needs are fully understood. Safety is well-served when the requirements and constraints on systems and components stem from realistic, safety-focused analyses, enabling resources to be applied to the most important and safety-focused areas.

Important milestones have been reached lately, with the 50.69 rulemaking and the proposed 50.46 rulemaking. In my view, these two are essential to day-to-day operational safety and to the progression of the NRC and the industry to a risk- and safety-focused regulatory framework. Moreover, the Commission has now endorsed risk-informing Part 50, in a progressive but comprehensive manner. Since I know I am preaching to the choir, I am going to ask you to sing and sing loudly for a sound and effective 50.46a and a risk-informed Part 50. It is a battle worth winning, for safety's sake.

Therefore, I support the issuance of an Advanced Notice of Proposed Rulemaking (ANPR) to develop a new risk-informed and performance-based Part 50. This will have special importance for the review of non-light water reactors, for which many of the existing elements of Part 50 are not applicable and for which many important issues may not be in Part 50 at all. An Advanced Notice of Proposed Rulemaking will establish a forum for discussing potential actions and opportunities that are beyond the scope of 50.69 and 50.46, and it will demonstrate the Commission's commitment to a broad application of risk-informed and performance-based regulation.

Let me now outline some of the actions involved in achieving an integrated and optimized approach to reactor safety, security and preparedness. In the near term, we do not need new programs, new policies or significant additional resources. We do need to "manage the risk," internally and externally, focusing on effective implementation and integration of on-going programs, namely, risk-informed rulemaking and exemptions (e.g. 50.46 ECCS requirements, 50.48 (the NFPA-805 alternative), and the GDC). These programs need to be managed to optimize safety, and not on minimizing changes. There is no doubt in my mind that we should take an aggressive approach to change, not a timid and minimalist approach where preservation of the status quo inhibits the enhancement of safety. Through an aggressive approach, we will seize this opportunity to set safety, via risk-management, in its proper place for every important issue.

To put the above in perspective, I will be very blunt. The TMI-2 failure was unacceptable and nothing comparable to it must be allowed to occur again. There is no doubt that the NRC (and its predecessor the AEC), with its preoccupation with Large Break LOCAs, contributed to the TMI-2 accident. We cannot allow the unrealistic conservatism of the past to constrain our ability to manage the real challenges of today and tomorrow. No, we cannot allow an event like the TMI-2 accident to go forward. There must be a healthy, rigorous, and constructive dialog about how to achieve improved safety, pragmatically and realistically, and then there must be a willingness to make the changes that implement it. That is what our responsibility to the American public demands.

Ladies and gentlemen of the probabilistic-risk and safety-assessment round table we know what to do. Let's do it!

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No. S-05-013

Regulatory Perspectives on the Nuclear Fuel Cycle

Platts
Nuclear Fuel Strategies:
Securing Fuel Supply, Managing Transportation and Storage
Washington, D.C.
September 22, 2005

Peter B. Lyons
Commissioner, USNRC

Chairman Nils Diaz was originally scheduled to join you today. I'm pleased to be with you today in his place. I'm hoping that my eight months of experience as a Commissioner will be sufficient to answer most of your questions, albeit not as well as the Chairman's extensive knowledge.

My goal for today will be to share with you my views on the Nuclear Regulatory Commission's (NRC) role in the future of nuclear energy utilization in the U.S. In keeping with the theme of your Conference, I will touch on all aspects of the nuclear fuel cycle.

There is no doubt in my mind that our nation will be challenged in meeting current and future needs for electricity generation. As we strive to meet this challenge, I believe that we should encourage fuel diversity in order to minimize pressure on limited supplies of natural gas, and to reduce our dependance on foreign energy sources.

For this new electricity generation, we need to tap renewables as much as possible. However, the intermittent character of solar and wind systems means that they can never play a dominant role in supplying baseload electricity, unless we invent new, very low cost energy storage systems. Our large coal reserve provides another opportunity for expanded baseload generation, but significant expansion of that resource will depend on development of cost effective, low emission plants. The only other source of significant new electricity generation within the next few decades is nuclear energy.

It's very evident that the nuclear power industry enjoys strong support from recent Administration and bipartisan Congressional actions. The recent visits 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, help 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 current versions of the House and Senate Appropriations Bills both provide strong support for nuclear energy, including increased funding for the NRC to perform security and new reactor licensing activities.

Support for nuclear power is also increasing from the environmental community. The founder of Greenpeace, Patrick Moore, recently testified before Congress that:

> *"I believe the majority of environmental activists, including those at Greenpeace, have now become so blinded by their extremist policies that they fail to consider the enormous and obvious benefits of harnessing nuclear power to meet and secure America's growing energy needs."*

Other noted environmentalists, like James Lovelock and Bishop Hugh Montefiore, have made similarly strong statements.

While this support for nuclear energy certainly enhances the prospects for new domestic plant construction, any applicant taking that step will have to weigh a range of financial risk factors. Some of those financial risk factors involve the regulatory framework for nuclear power and thus directly involve the NRC. With that in mind, I'll turn to discussion of NRC roles in the fuel cycle.

I'll start with the mining and milling of uranium. NRC provides regulatory oversight to 16 uranium recovery licensees in various stages, including operational in-situ, remediation, and standby. Our key challenge in this area is resolving complex groundwater issues.

Despite the decommissioning and remediation activities that NRC regulates, I note an optimism in this sector which wasn't evident before. There are many reports of increased domestic and global exploration for new uranium deposits. While U.S.-mined uranium is now a very small fraction of our annual usage, this improved climate for new mining may lead to increased domestic interest in mining.

NRC licenses eight special nuclear materials facilities, including six fuel fabrication facilities and two gaseous diffusion enrichment facilities. While only the Paducah site is actively enriching product, NRC is experiencing a high level of activity in support of the future domestic fuel supply.

The first gas centrifuge application was received in February 2003 for USEC's Lead Cascade Facility. That application was approved in 2004. This facility is intended to provide a demonstration facility for USEC's American Centrifuge design. The Facility will consist of a number of centrifuges with a total possession limit of 250 kg of UF_6. The only uranium withdrawals from the cascade will be in the form of samples.

We are reviewing licenses for two gas centrifuge applications and have issued a construction authorization for a mixed oxide fuel fabrication plant. In December 2003, the NRC received an application for the National Enrichment Facility to be built in Eunice, New Mexico. This facility is being designed for a capacity of three million SWU per year. NRC developed a 30-month schedule for making a final determination. The Final SER (Safety Evaluation Report) and the EIS (Environmental Impact Statement) were both issued in June 2005. The Atomic Safety and Licensing Board hearing on safety issues is scheduled for October 2005 with completion scheduled for February 2006. We expect a licensing decision in June 2006.

In August 2004, NRC received the license application for USEC's commercial-scale gas centrifuge facility, the American Centrifuge Plant, which will include the Lead Cascade Facility and is being designed for 3.5 million Separative Work Units (SWU) per year. In October 2004, NRC accepted USEC's license application and environmental report for detailed technical review. Currently, the NRC staff is on schedule to issue the final EIS and SER in early 2006 with an expected licensing decision to follow about one year later.

In March 2005, NRC issued a construction authorization for a Mixed Oxide (MOX) Fuel Fabrication Facility to be located at the Savannah River Site. This facility is to disposition 34 metric tons of excess weapons-grade plutonium through irradiation of MOX fuel in domestic commercial nuclear power plants. In addition, NRC has approved use of MOX lead test assemblies in the Catawba plant.

Nuclear Plant Safety and Future

The next step of the fuel cycle is power reactors. I already noted the support for new reactors from several sectors. There is considerable optimism now that several utilities are seriously considering new plant construction. But, in any discussion of the potential for new nuclear plant construction, we must always remember that the entire nuclear power industry has a vital job to attend to first – safe and secure operations of existing plants.

The public needs to be confident of this before they will support new nuclear plants. I want to further emphasize the roles of both NRC and the industry in providing adequate protection of public health and safety and the environment with safe and secure operations. In addition, NRC must provide regulatory stability into the future.

First, the industry must maintain a clear focus on safe operations as a means to assure a large margin is maintained against any harmful release of radioactivity from a commercial nuclear plant in the United States. Furthermore, with this focus, and under the watchful oversight of the NRC, the industry must constantly guard against another serious incident like the reactor vessel head degradation encountered at Davis-Besse.

This focus on safety must extend to natural phenomena that could challenge safe operation of plants. For example, I am proud of NRC's extensive proactive planning for Hurricane Katrina to assure that safety was never compromised by the terrible conditions near several of the nuclear plants.

Second, NRC needs to monitor and report on industry's continued safety performance through our various methods, including the reactor oversight process and the generic issues program. In general, industry's safety trends have shown improvements in decreasing the number and severity of events and safety system failures. The reactor oversight process now uses more objective, timely, and safety-significant criteria in assessing performance, while seeking to more effectively and efficiently regulate the industry.

While assuring safety, NRC must also strive for stability of the regulatory environment, that is, maintaining consistency and predictability. Although this can be a challenge, NRC has demonstrated through programs like the reactor oversight process that it makes predictable regulatory decisions.

Third, security was a key focus of the NRC before 9/11 and has been substantially enhanced since then. Some of the security enhancements are obvious as one approaches any plant perimeter, such as intrusion barriers. 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. In addition to actions of NRC and the licensees, airborne threats are primarily addressed through the operations of the Department of Homeland Security and the North American Aerospace Defense Command, more commonly referred to as NORAD. With these many enhancements, our nuclear plants are even more secure today.

Several vital provisions of the new Energy Bill further enhance plant security. Guards at power plants may now carry more powerful weapons. Federal criminal statutes were expanded to further protect key nuclear facilities and our ability to demand fingerprinting and criminal history checks was expanded.

Fourth, in addition to public assurances on safety and security, nuclear power will not advance unless the industry and the public have confidence that NRC's licensing procedures are well understood, incorporate significant public input, and operate on predictable time scales. NRC's performance on license renewals, power uprates, and new licenses are evidence of our success in this area.

New Plant Construction

At one time, 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 leader in these areas. Today, we have enough remaining infrastructure, both human and industrial, to recover, but may be in danger of losing these capabilities in the not too distant future.

My own view is that the time frame within which we must determine our nation's future capabilities in nuclear energy is at most the next couple of decades. Unless near-term progress is demonstrated in the United States within that time window, which includes construction of new plants, our nation may lose much of our technical capability to support new construction projects using domestic resources. There is no question that today there is more interest in new nuclear power plant construction than at any time in recent history, and a number of companies are now seriously discussing such possibilities.

Historically, the licensing process for nuclear plants allowed construction to start even as technical safety questions were still being addressed, often resulting in extended and costly delays in approving the operating license. In 1989, the NRC established 10 CFR Part 52 which provides for a combined construction and operating license, referred to as a COL. The process also includes the Early Site Permit or ESP process and the Standard Design Certification, which is intended to ensure all safety questions has been addressed prior to major construction. Both the ESP and the design certification may be referenced to simplify an application for a COL.

The overall goal of the COL process is to provide a stable, efficient, and a predictable regulatory framework for utilities that might wish to pursue a new reactor license. At the same time, the NRC has been careful to include appropriate opportunities for public input throughout the parts of the ESP, design certification, and COL processes. I would like to briefly describe each process and give an update on industry interest.

The ESP process allows early resolution of site-related issues and allows an applicant to "bank" a site for future construction. The three key factors that determine site suitability are site safety, emergency preparedness, and environmental protection. The initial permit is issued for 10 - 20 years with renewals issued for an additional 10 - 20 years. Applications have been received for the North Anna, Clinton, and Grand Gulf sites, and the NRC is on track to issue final decisions in 2006 and 2007 for these cases. Southern Company has announced their intent to submit an ESP application.

The standard design certification process allows a vendor to submit a plant design to the NRC for review and certification. The application is independent of a site and the safety reviews are completed based on an essentially complete reactor design. Initial certifications are issued for 15 years with renewals issued for 10 -15 years.

The first standard design certification was issued for the General Electric Advanced Boiling Water Reactor (ABWR) system in 1997. Today three advanced designs are certified, the GE ABWR, System 80+ and AP600. A certification review for the AP1000 is in progress and out for public comment; the ESBWR application was just received for review; and other applications are expected to be filed soon. The NRC has estimated times for completion of a design certification to range from 42 to 60 months depending on the complexity of the design and the extent of 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 design certification are included in the COL application, the review and hearing process for the combined license is anticipated to require less than 30 months.

Currently, several utilities have expressed interest in submitting COL applications, for example:

- A consortium led by Dominion Resources is considering the GE ESBWR design at the North Anna site.
- A consortium led by TVA was scheduled to complete a feasibility study in August for construction of two advanced BWRs at the Bellefonte site. Based on the results of the study, TVA will decide on submitting a COL application.
- The NuStart Energy consortium is considering both the Westinghouse AP1000 and GE ESBWR designs. They have selected six potential sites and currently plan to submit COL applications sometime in 2008.
- Duke, Southern Company, South Carolina Electric and Gas, and Progress Energy have all recently expressed interest in possible COL applications.

One aspect of the COL process which is getting a lot of industrial attention involves verification of Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC). I've heard concerns that this verification step could lead to lengthy delays in the final operation of a site, defeating the whole intent of the Part 52 approach.

In my view, as long as the ITAACs are carefully developed and appropriately focused, and as long as the constructed plant fully meets the agreed upon ITAACs, this verification step should not represent any serious delays. But I also recognize that this is an untested aspect of the new Part 52 process that may cause concerns just because it is new. The inclusion of regulatory delay insurance in the Energy Bill should address this concern.

Storage and Transportation Activities

Spent nuclear fuel storage and transportation activities are extremely important to support the overall national fuel cycle. At the moment, the NRC regulates 30 operating independent spent fuel storage installations. This number has more than doubled from five years ago. Based on current projections, there could be approximately 50 independent spent fuel storage installations by 2010. One indication that this projection is accurate is the continued industry interest in new cask designs. The dry cask storage industry is a maturing industry which is producing robust and safe products.

To date, we have certified 14 cask designs, submitted by five vendors, that are approved for storage of spent fuel. Some of these designs are dual-purpose and are approved for transportation as well as storage. New cask designs are evolving and pushing the technical envelope. Some examples of issues in this area that the staff continues to address are: high burnup fuel thermal issues, allowance for burnup credit, and moderator exclusion for transport.

High Level waste

Finally, let me turn to the back end of the fuel cycle. There should be no doubt that if we receive the License Application for a repository from the DOE, it will be one of the greatest challenges in the history of NRC. NRC has been preparing for this potential challenge for many years. As an Agency, we believe we will be well positioned to respond within the times specified in the High Level Waste Act.

NRC recently issued a proposed rule for public comment that would amend the regulations governing the disposal of high level waste to be consistent with revised EPA environmental standards for Yucca Mountain high level waste repository. Another possible near-term action may be DOE's certification that the Licensing Support Network has been populated. This certification must precede submission of a license application by at least six months.

NRC's staff is working to ensure that we have the appropriate infrastructure in place to support a potential review. Once the potential application is docketed, NRC would conduct extensive technical reviews, as well as public hearings which would be overseen by the Atomic Safety and Licensing Board. After completion of the hearings, the Board would forward its initial decision to the Commission for review.

NRC is preparing for the anticipated legal proceedings, if a license application is received. One major step we have taken is to establish the Commission Adjudicatory Technical Support Program. This division consists of technical experts that will advise the Commission during its review of the Atomic Safety and Licensing Board's initial decision on the application. These staff members will be independent from that staff performing the initial review of DOE's application. This is necessary to guarantee that the Commission's final decision on the application is impartial and untainted by improper communications between the Commission and the staff conducting the review of the application.

Summary

In summary, retaining the nuclear energy option in the future requires continued safe performance of the current operating reactors and continued strong and independent NRC oversight. In addition, it 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.

While the industry has demonstrated a strong track record in recent years, it has not been without challenges and opportunities to learn. As an example, both the industry and NRC's staff must learn and institutionalize the important lessons from the Davis-Besse corrosion event – and not just the technical aspects – but more importantly avoiding the underlying complacency and failure to maintain a questioning attitude.

Another challenge for both the industry and NRC is 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. We must all commit to actively mentoring our less experienced employees to pass on the important values that are essential to continued safe use of the nuclear energy option.

Overall, the industry's performance, as well as NRC's regulatory oversight, will be carefully observed by the public. Only if both the industry and NRC demonstrate strong performance can public confidence be maintained sufficiently to permit an objective and reasoned public dialogue on the future of nuclear energy in this country.

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No. S-05-011

Remarks by

Dr. Nils J. Diaz, Chairman
Unites States Nuclear Regulatory Commission

at the

Joint Meeting of the National Organization of
Test, Research, and Training Reactors
and the International Group on Research Reactors
September 14, 2005

Thank you, Brian [Brian Thomas, Chief of the NRC Research and Test Reactor Section], for your introduction, and a special thank you to Tawfik Raby and Sy Weiss, Co-Chairs of the Organization of Test, Research, and Training Reactors, for inviting me to chat with you. I am going to state the obvious: I am here because I care about this community in a very special way, and because, as Chairman of the NRC, I have direct responsibility for the regulatory oversight of your facilities. I will try to make the best of these two facts in my remarks in the context of the realities of 2005 and beyond.

The research and test reactor community is an important national asset that makes significant contributions in many arenas to the benefit of the people of our great country. It is important that you continue to do so; in fact, it is important that you fulfill the larger role that I believe is around the corner.

The role that you play, and the larger role that you should play, require that you devote the time, the intellect, and the resources to assure the safe and secure operation of your reactor facilities. It is in the area of safe and secure operations that the NRC's licensing and oversight responsibilities interact with your programs to ensure protection of the public and the common defense and security.

And you have done just that: you have protected public health and safety and national security. NRC-licensed TRTRs have a distinctive and laudable record: no member of the public has been injured from the operation of your facilities, no hazardous release of radioactivity has occurred, and you have secured and accounted for nuclear fuel and materials important to national security. You have done all this for many many years, while serving a unique educational and research role for our country.

Yet, today, you and every American are asked to do more. The events and the consequences of the 9/11 terrorist attacks have changed America, and indeed, the entire world. We are all asked to be more vigilant, to take that extra careful step to prevent malevolent events, and specifically, to ensure that there are no gaps in the safety, security, and emergency preparedness in nuclear reactor facilities and radioactive materials users. More has been asked by our country and more has been given by the NRC and its licensees, including TRTRs. You operate in the midst of close-knit communities, in the heartland of America, because you can and you should. The visibility of research and test reactors, and the need to educate, place special demands on your efforts, acknowledging the established facts of low inventory of radioactivity at your reactors and the systems and barriers designed to prevent accidents and minimize potential radioactivity dispersal.

The TRTR community must discharge its responsibilities and establish its "value-added" within the regulatory framework of the NRC, based on the realities of a demanding, yet forgiving, nuclear technology. This is a technology that is always in the public spotlight. You must discharge your responsibilities in an unforgiving environment for nuclear or radiological events, no matter how small the consequences. Someone asked me the other day to place the risks and benefits of TRTRs in perspective. Easy. TRTR facilities, operated within the framework of safety, security, and emergency preparedness that you are required to implement, are safe and have minimal risks for affecting public health and safety. This is true for all types of credible events considered, malevolent or not, regardless of the cause. Furthermore, the risk of diversion of nuclear material with national security implications is low, and getting lower.

On the beneficial side, you bring value to our country in your educational, training, research and service activities. I personally expect that you will continue to do so, even more so as the Nation asks you, the industry, and the NRC to prepare for and fulfill new and growing expectations for energy security, including nuclear power, and as we continue to ensure national security.

Before I discuss future directions for the TRTR community, I need to repeat and emphasize key facts on security. The NRC and the research and test reactor community have worked closely to further improve security in recent years. We must continue to do so. These security improvements have appropriately considered that research and test reactors pose low risks of radiological exposure to the public. Furthermore, within our presently existing national security programs, the risk from the theft of radioactive materials from TRTRs is also low. This does not mean that the NRC and the TRTRs take safety and security for granted; on the contrary, it means that we take it very seriously and will continue to do so.

Enhanced nuclear safety, security, and emergency preparedness are cornerstones for the protection of the people of America from radiological hazards. We have taken all three to new levels of performance. As we go forward, the NRC will continue to be vigilant, cognizant of the threat and of the need to ensure that every one of our licensees is performing at the level needed to protect the public.

With regard to research and test reactors, in response to the terrorist attacks of September 11, 2001, a number of Commission-directed security initiatives were begun. Compensatory security measures were developed and licensee plans for implementing these measures were reviewed and approved. Field assessments were made at most reactors to confirm the effective implementation of the compensatory security measures. The Commission also directed the NRC staff to perform additional safety and security assessments of research and test reactor licensees. Significant NRC resources were directed at completing the security assessments and security-related inspections. Licensees also devoted significant resources to enhance safety, security, and preparedness.

I mentioned a larger role ahead for your communities. The prospect of licensing and constructing new nuclear power plants is squarely in front of us. The President's agenda, as well as the recently enacted Energy Bill, encourage new nuclear power plant licensing and construction. To facilitate this objective, the Energy Bill also contains several provisions that are intended to enhance science and engineering education, including NRC scholarship and fellowship programs for fields that are critical to the NRC's mission, and authorization to provide financial assistance to institutions of higher education to promote the development of academic offerings in subject areas that relate to NRC's mission. The Department of Energy was also directed to undertake similar initiatives to improve the state of science and technology education and training for the Nation's energy workforce. You must play a key role in this endeavor.

The NRC is prepared to objectively review new reactor applications, conduct new reactor construction inspections, and implement effective oversight of any new reactors. In fact, we are currently working on early site permits, a design certification, and a number of pre-application reviews. Strong rumblings about applications for combined licenses (or COLs) are being made. As I am sure you know, there will need to be a good deal of preparation on the part of applicants, utilities, and the NRC to support and oversee new reactor licensing and construction activities.

All this creates an urgent need to train new nuclear professionals. The NRC will substantially increase its recruiting efforts to hire approximately 350 new entry-level and experienced employees by the end of next year. You heard right: 350. That amounts to more than 10% of the agency's current staffing level. In one year, no less. We need to do this to offset expected retirements and to increase staffing levels in anticipation of potential new reactor license applications in 2007 and 2008. We have worked hard over the years to make the NRC an attractive place to work. The American University has identified the NRC as one of the 10 best Federal agencies to work for based on the results of the Office of Personnel Management's 2004 Federal Human Capital Survey. We are well up in the top 10, and there is not a regulatory agency that is ahead of us. Programs such as the Student Career Experience Program, the Nuclear Safety Professional Development Program, and the Graduate Fellowship Program are expected to continue to attract highly qualified and motivated employees to the agency. All that having been said, the Commission, and senior agency management, are well aware that we will have to compete with the regulated industry for qualified people. We will need your facilities to train many of our future employees.

The research facilities and educational institutions supporting TRTRs should play essential roles in technology development and education and training. Key aspects of the safety cases and technical basis for reactor applications will be supported by research and data developed in the research and test reactor community. Students will be trained at research and test reactors. Many of the engineers, health physics technicians, nuclear physicists, and other nuclear professionals will also be educated and trained at facilities associated with research and test reactors. Many new technologies, including innovative new fuel designs, will be tested at research and test reactors. A commitment to operational safety and the supporting know-how by new professionals is founded in your teaching.

The Commission is now using a terminology that I strongly endorse: the NRC's strategic objective is to enable the use and management of radioactive materials and nuclear fuels for beneficial civilian uses, in a manner that protects public health and safety, the environment, and national security. It could have easily been written for you.

The point is clear: you have your work cut out for you, and we have our work cut out for us. The challenges facing us over the next few years are great, yet they are very "do-able." We need to do them timely and right. Let me stop there, and take your questions.

September 28, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - COMSECY-04-0068 - USE OF INSURANCE AS A METHOD TO PROVIDE FINANCIAL ASSURANCE FOR DECOMMISSIONING NUCLEAR POWER REACTORS

The Commission has disapproved the draft supplement to the Standard Review Plan (SRP) relating to the use of insurance for decommissioning funding for nuclear power reactors.

The NRC's regulations allow for the use of insurance for decommissioning funding purposes, and the Commission is considering how the NRC's guidance in this area should be supplemented. In particular, it is the Commission's intent to develop a regulatory guide to provide broad instruction for external stakeholders and the staff related to complex insurance issues that have arisen and might arise again. Taking into account possible resource implications, the Commission is considering whether this guidance document should be prepared using a phased approach.

The first step towards a decision in this area is to schedule a public Commission meeting with panels of experts and representatives of affected interests (assembled by SECY in consultation with the staff) on the issues associated with development of expanded criteria. This meeting will help the Commission obtain information necessary to reach an informed decision on a path forward and to ascertain the level of interest on the part of insurance companies and the industry in using insurance for decommissioning funding. The Commission will consider discontinuing efforts to develop a guidance document if, among other things, the Commission receives significant input from stakeholders indicating a lack of interest in use of insurance for decommissioning funding.

In preparation for the development of a regulatory guide, the staff should prepare a resource estimate of the costs associated with preparation of a document addressing the following:

(1) the NRC's global view of how insurance can provide financial assurance for decommissioning;
(2) how 10 C.F.R. § 50.75(e)(1)(iii) should be implemented;
(3) how the staff will evaluate proposals to use insurance;
(4) data required by the staff for review of such proposals; and
(5) the preferred standard format, in the form of a sample policy, for any insurance proposals.

The Commission has given careful attention to matters raised by the staff's draft SRP and has reached some initial observations regarding aspects of insurance proposals that must be addressed in the regulatory guide. The Commission is sharing these observations as a baseline position, knowing full well that the public meeting may yield information and additional perspectives that will merit reconsideration of some issues.

First, the Commission is inclined to disapprove the participation of captive insurers. While risk retention groups (RRGs) may be permitted, the Commission thinks they should receive close scrutiny that may require a financial strength rating, and should be approved under limited circumstances. Any guidance in this area should call for a staff review of scenarios that focus on common economic failures of RRG members prior to approval of any proposal involving a RRG in order to ensure an adequate level of risk independence among the members of the group. It may be that risk retention groups, as a class, would not provide the necessary level of financial soundness needed for decommissioning funding assurance purposes.

Second, the Commission believes it is appropriate to allow a limited claims management process that would permit an insurer to have input into how decommissioning activities are completed, e.g., what vendor/contractor will perform work or what an acceptable cost range for activities would be. Claims management provisions should in no way allow an insurer to disapprove any activity or cleanup of any level of residual contamination specifically approved by the NRC as part of a license termination plan. The public meeting described above should explore the position forwarded by some commenters that insurers cannot offer insurance without some claims management and the ability to determine whether a claim is legitimate (covered and incurred cost).

Third, the Commission is considering the use of sublimits to govern the amount of funds available for NRC-directed radiological cleanup activities in policies with multiple purposes. The Commission might also consider a limited priority clause to permit unused or unneeded funds to pass from non-radiological cost coverage to radiological or vice versa.

Fourth, the Commission thinks litigation expenses should be covered under insurance policies. The Commission will consider what percentage of the policy value should be paid out for litigation costs.

Fifth, the Commission believes an insurer should be required to possess an appropriate license to transact the business of insurance that includes the proposed type of license. This can be in lieu of the proposed requirement that the insurer either be licensed in the States where the plants are located or obtain approval or a statement of non-objection from such state authorities.

Sixth, the draft guidance document developed by the staff should be published for public comment, and submitted to the Commission for information prior to publication. The final regulatory guide should be submitted to the Commission for approval prior to publication.

Finally, the Commission considers that any criteria developed in guidance regarding the use of insurance to provide financial assurance for decommissioning should assure that the risks associated with the particular insurance proposals are not substantially greater than prepayment or external trust funds.

cc: Chairman Diaz
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
DOC
OGC
CFO
OCA
OPA

January 18, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-04-0182 - STATUS OF RISK-INFORMED REGULATION IN THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

The Commission has approved the staff's plan to continue applying risk-informed methods on materials and waste repository activities and request to discontinue the periodic report on risk-informed regulation for the materials and waste arenas. The staff should consider applying the risk-informed decision making guidance to planned and emergent regulatory activities as described in the paper and guidance document, and identifying appropriate management controls to track the progress of these efforts (e.g., Operating Plans). The staff should keep the Commission informed on significant activities and results and should provide an overview of this topic as part of the annual Commission briefings on the Materials and Waste programs.

The staff should implement management controls to ensure that negligible values used as screening levels do not become default ALARA levels or used in any way as regulatory limits. The staff should ensure that valuable resources are never applied to lower a risk that is already considered to be negligible.

In addition, the staff should remove Appendix F "Risk-Informing the NMSS Inspection Process" from the guidance document. Although the staff should consider ways to apply a risk-informed approach to the front end of the inspections program (i.e., focusing inspections on the areas of highest risk), the guidance currently contained in the Appendix is not sufficiently developed for even trial use at this time. There is no objection to staff developing a revised variation of Appendix F as a stand alone document to generate discussions on how to develop a risk informed inspection approach in the materials area. But there should be no indication that this document should be for trial use. Before initiating such discussions, the revised Appendix F should be submitted to the Commission for information. The appendix as written is too closely related to equivalent guidance for the commercial reactor program and would not be applicable for many materials licensees. Also, unless presented with strong evidence suggesting otherwise, the Commission does not intend to extend the requirement to conduct Integrated Safety Assessments to additional materials facilities.

(EDO) (SECY Suspense: TBD)

The staff should exercise extreme caution in any attempt to risk-inform security-related matters. The security arena is distinctly different from the safety arena.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
DOC
OGC
CFO
OCA
OPA

September 21, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-05-0156 - UNITED STATES NUCLEAR REGULATORY COMMISSION PARTICIPATION IN THE ORGANIZATION FOR ECONOMIC COOPERATION AND DEVELOPMENT HALDEN REACTOR PROJECT DURING 2006-2008

The Commission has approved the staff's recommendation to continue participation in the Halden Reactor Project.

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

September 21, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-05-0138 - RISK-INFORMED AND PERFORMANCE-BASED ALTERNATIVES TO THE SINGLE-FAILURE CRITERION

The Commission has approved the staff's recommendation to seek additional stakeholder involvement by making the draft technical report on single failure criterion (SFC) available to the public.

Consistent with the direction provided in the Staff Requirements - SECY-05-0130, the staff should develop expeditiously an Advanced Notice of Proposed Rulemaking (ANPR) to consider the spectrum of issues relating to risk-informing the reactor requirements including the effort to develop risk-informed and performance-based alternatives to the single failure criterion. This will assure that efforts to risk-inform the reactor regulations are undertaken in an open, transparent, and integrated manner. Safety, security and preparedness should be integrated throughout this effort.

(EDO) (SECY Suspense: TBD)

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

September 9, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

Karen D. Cyr
General Counsel

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

SUBJECT: STAFF REQUIREMENTS - SECY-05-0120 - SECURITY DESIGN EXPECTATIONS FOR NEW REACTOR LICENSING ACTIVITIES

The Commission has approved the staff's recommendations on security design expectations for new reactor licensing activities, subject to the following comments.

1. The staff should revise the 1994 Commission Policy Statement on the Regulation of Advanced Nuclear Power Plants to integrate the expectations for security and preparedness with the current expectations for safety, and develop an implementation plan for the activities listed below. Concurrently, the staff should develop the security-related characteristics and attributes that should be included in new reactor designs, and involve stakeholders in developing guidance for applicants or prospective applicants on the security-related assessments that should be included in design certification applications.

2. The staff should conduct a rulemaking to require applicants to submit a safety and security assessment addressing the relevant security requirements which were established for currently operating plants by order[1], including the requirements for protection against the supplemented design basis threat and the requirements for enhanced mitigative measures.

   Applicants whose reactor designs are in the design certification review process before the final rule is issued should be encouraged, but not required, to submit a design-specific safety and security assessment as part of the application. If an applicant voluntarily submits this assessment, the staff should review it to assure that the design features identified and described are consistent with the relevant security requirements imposed on currently operating plants by order[1], and that reasonable and practicable safety and security features have been appropriately integrated into the design.

[1] February 25, 2002, All Operating Reactor Licensees, Order Modifying License (Effective Immediately), EA-02-26, 67 FR 9792 (March 4, 2002); April 29, 2003, All Operating Reactor Licensees, Order Modifying License (Effective Immediately), EA-03-086, 68 FR 24,517 (May 7, 2003).

Resolution of security-related design issues at the early stage of the regulatory review process should result in a more robust security posture requiring less reliance on operational security programs if a plant is constructed based on the approved design. However, resolution of the security-related design issues would not constitute final NRC approval of an applicant's overall security program. NRC review and approval of an applicant's security program would still be required before issuing a combined license, or a construction permit and operating license, for a specific site.

The staff's approach to establish security design requirements for new reactor licensing should employ clearly defined regulatory and legal processes. OGC should be an active participant in the staff's development of this approach and ensure that it does not create unnecessary adjudicatory issues.

cc: Chairman Diaz
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
DOC
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OPA

September 14, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

John T. Larkins
Executive Director, ACRS

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

SUBJECT: STAFF REQUIREMENTS - SECY-05-0130 - POLICY ISSUES RELATED TO NEW PLANT LICENSING AND STATUS OF THE TECHNOLOGY-NEUTRAL FRAMEWORK FOR NEW PLANT LICENSING

The Commission has disapproved the staff's recommendations on specifying the minimum level of safety for new reactor designs, and the proposal on the integrated risk from modular or multiple reactors at a site.

The Advisory Committee on Reactor Safeguards (ACRS) should provide its views on these two policy issues, including the feasibility of alternatives to the Quantitative Health Objectives (QHOs) as technology-neutral risk objectives. The staff should then consider ACRS comments in developing a subsequent notation vote paper addressing these policy issues.

The staff should develop expeditiously an Advanced Notice of Proposed Rulemaking (ANPR) to consider the spectrum of issues relating to risk-informing the reactor requirements. The formal program to risk-inform Part 50, as well as other related risk-informed efforts, should be incorporated into this ANPR. Safety, security and preparedness should be integrated throughout this effort.

(EDO) (SECY Suspense: TBD)

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

September 8, 2005

MEMORANDUM TO: Chairman Diaz

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

SUBJECT: COMNJD-05-0006 - MULTINATIONAL DESIGN APPROVAL PROGRAM (MDAP), STAGE 1

This memorandum is to inform you that all Commissioners have concurred in your proposal to move forward with a Multinational Design Approval Process, Stage 1, to further enhance international cooperation in reviewing new power reactor designs. The attached SRM provides staff direction on this issue.

This completes action on COMNJD-05-0006.

Attachment:
As stated

cc: Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
EDO
OGC

September 8, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

Janice Dunn Lee, Director
Office of International Programs

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

SUBJECT: STAFF REQUIREMENTS - COMNJD-05-0006 - MULTINATIONAL DESIGN APPROVAL PROGRAM (MDAP), STAGE 1

The Commission has approved moving forward with Stage 1 of the Multinational Design Approval Program to further enhance international cooperation in reviewing new power reactor designs, and the allocation of 2 FTE in FY 2006 to implement this activity. The detailed working arrangements should be formalized with regulators interested in participating in this program by developing administrative letters or other similar correspondence, as appropriate, under existing bilateral agreements. The NRC Design Certification process will remain the regulatory framework for these efforts, with participating regulatory authorities acting as expert consultants. The NRC staff will remain responsible for regulatory decisions and recommendations concerning reactor design certifications, incorporating technical input from their foreign counterparts, as appropriate. A detailed evaluation of the input provided by the foreign regulators should be performed prior to the staff using the information in its design review.

(OIP/EDO) (SECY Suspense: TBD)

The staff should develop a plan to facilitate communications with external stakeholders, including the public, the industry, reactor vendors, and regulatory authorities in other countries. The staff should also seek comments from the Department of State and the Department of Energy, as appropriate. The staff should inform the Commission of the results of these activities.

(OIP/EDO) (SECY Suspense: TBD)

The staff should provide the Commission with a detailed scope and schedule for implementing Stage 1, once the staff has formalized the detailed working arrangements with the vendors and foreign regulators that have expressed an interest in the program.

(EDO) (SECY Suspense: TBD)

The staff should use the insights gained from implementing Stage 1 to aid in the development of the necessary processes, tools, and terms for Stage 2 of the Multinational Design Approval Program before presenting the Stage 2 proposal to the Commission.

cc: Chairman Diaz

Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
DOC
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OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

August 5, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-05-0113 - DENIAL OF A PETITION FOR RULEMAKING TO REVISE APPENDIX K TO 10 CFR PART 50 AND ASSOCIATED GUIDANCE DOCUMENTS (PRM-50-76)

The Commission has approved the staff's recommendation to deny the petition for rulemaking, PRM-50-76, that requested the agency to revise the metal-water oxidation criteria in Appendix K to 10 CFR Part 50 and Regulatory Guide 1.157. The Commission has approved publication of the *Federal Register* notice and dispatch of the letter to the petitioner subject to incorporation of the comments and changes noted below.

(EDO) (SECY Suspense: 9/9/05)

1. The staff should confirm that the various data sets, tests, and experiments it discussed in support of denial of the petitioner's request are publicly available and that they are appropriately referenced in the *Federal Register* Notice. If the documentation of the referenced data sets, tests, and experiments are in ADAMS, the appropriate accession numbers should also be referenced in the *Federal Register* Notice.

2. The staff should ensure that the *Federal Register* Notice adequately defines all the acronyms used.

3. The following sentence contained on page 2, lines 4 and 5 of the letter to the petitioner and page 21, lines 8 and 9 of the Federal Register Notice needs to be modified to clarify how these experiments relate to the denial of the petition. "The NRC funded more than 50 Zircaloy clad bundle reflood experiments at the National Research Universal (NRU) reactor."

4. The following sentence contained on page 2, lines 5 through 10 of the letter to the petitioner and page 21, lines 10 through 13 of the Federal Register Notice needs to be modified to clarify how these programs relate to the denial of the petition. "The NRC is currently conducting and evaluating experimental and analytical programs on fuel cladding behavior.....to evaluate the adequacy of current 50.46 oxidation-related criteria and models."

5. The following paragraph on page 2 of the letter to the petitioner and page 22 of the

SECY NOTE: TO BE MADE PUBLICLY AVAILABLE 5 WORKING DAYS AFTER DISPATCH OF THE LETTER TO THE PETITIONER.

Federal Register Notice needs to be modified to clarify how this information relates to the denial of the petition.

> "The NRC applied the Cathcart-Pawel oxygen uptake and ZRO2 thickness equations to the four FLECT Zircaloy experiments .... [start new paragraph] The NRC applied the Cathcart-Pawel oxide thickness equation to 15 of their transient temperature experiments ..... This result is consistent with the application of the Cathcart-Pawel equations, which are intended for use in best-estimate LOCA calculations in accordance with RG 1.157."

<u>Additional specific changes to the *Federal Register* notice</u>

6. On page 1, paragraph 1, revise line 5 to read ' ... deficiencies in the correlations used for calculation of ....' Revise line 6 to read ' ... states that the correlations do ~~calculation does~~ not ....'

7. On page 4, paragraph 1, revise line 5 to read ' ... does not include any allowance ~~allow~~ for the ....'

8. On page 4, paragraph 2, revise line 2 to read ' ... ~~tests that, in~~ the petitioner's opinion ~~,~~ that the tests discussed in ANL-6548 do not ....'

9. On page 5, 1st full paragraph (after the bullets), revise line 1 to read ' ... conclusions include a statement that "overlooks the very ....'

10. On page 10, paragraph 1, revise line 5 to read ' ... that ~~the calculated~~ ECCS cooling ....'

11. On page 12, 1st full paragraph, revise line 3 to read ' ... correlation in the temperature range important for clad oxidation calculations for LOCAs. ~~above 1900 °F~~. In the ....' Delete the last sentence (Only directly-heated ... WCAP-7665).)

12. On page 13, 1st full paragraph, revise line 2 to read ' ... discussed under issue 3 ~~2~~.'

13. On page 16, paragraph 1, revise lines 1 and 2 to read ' ... and the industry have continued to ~~are currently~~ conduct~~ing~~ and evaluate~~ing~~ experimental ....' Revise lines 4 through 6 to read ' ... system pressure. As is the case with many other research activities and their link to the agency's regulatory framework, an ~~An~~ important objective of this work is the confirmation ~~to evaluate the adequacy~~ of current § 50.46 ~~oxidation~~ criteria and models and the development of more realistic, performance-based, and contemporary criteria and models.'

14. On page 16, paragraph 2, revise line 1 to read 'The NRC disagrees with the petitioner's assertion ~~petitioner is mistaken in asserting~~ that the ....'

15. On page 18, 2nd full paragraph, revise line 1 to read '~~The high~~ High-temperature tests similar ....' Revise line 2 to read ' ... reactors (PWRs) and would produce very little useful heat transfer information.' Delete the sentences in lines 3 through 7 (Also, no realistic ... transfer information.) Insert the following new sentence at the end:

SECY NOTE: TO BE MADE PUBLICLY AVAILABLE 5 WORKING DAYS AFTER DISPATCH OF THE LETTER TO THE PETITIONER.

'Therefore, the NRC does not believe that such tests are necessary.'

16. On page 21, revise line 2 from the top to read ' ... the correlation for the temperature range important to clad oxidation calculations for LOCAs ~~above 1900 °F~~.'

17. On page 21, 2nd full paragraph, revise line 3 to read ' ... NRU reactor, and is continuing to conduct and evaluate experimental and analytical programs on fuel cladding behavior.'

18. On page 21, delete the 3rd full paragraph (The NRC is ... criteria and models.)

19. On page 22, 2nd full paragraph, revise line 5 to read ' ... of PRM-50-78 and addressed by the staff's evaluation of that petition for rulemaking.'

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

SECY NOTE: TO BE MADE PUBLICLY AVAILABLE 5 WORKING DAYS AFTER DISPATCH OF THE LETTER TO THE PETITIONER.

EA-05-103 - LaSalle 1 and 2 (Exelon Nuclear)

September 7, 2005

EA-05-103

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT 05000373/2005010; 05000374/2005010), LASALLE COUNTY STATION, UNITS 1 AND 2

Dear Mr. Crane:

The purpose of this letter is to provide you with the final results of our significance determination of the preliminary White finding identified in the subject inspection report issued June 22, 2005. This finding was assessed using the Significance Determination Process (SDP) and was preliminarily characterized as White (i.e., a finding with low to moderate increased importance to safety, which may require additional NRC inspection). The White finding involved a single point vulnerability that could result in a loss of all onsite and offsite power sources to both 4160 Vac Division 1 and Division 2 safety-related buses at either of your LaSalle County Station units.

In our letter dated June 22, 2005, the Nuclear Regulatory Commission (NRC) provided Exelon Nuclear with an opportunity to address the White finding documented in the inspection report by either requesting a Regulatory Conference or by providing a written response before we made our final risk significance determination. On July 7, 2005, in a telephone conversation between the LaSalle County Station Plant Manager, Mr. Daniel Enright, and Mr. Bruce Burgess of the NRC Region III Division of Reactor Projects, you informed us that you did not intend to request a Regulatory Conference, and did not intend to provide a written response.

After considering the information developed during the inspection, and in the absence of any additional information provided by you, the NRC has concluded that the inspection finding was appropriately characterized as White (i.e., an issue with low to moderate safety significance which may require additional NRC inspection).

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter (IMC) 0609, Attachment 2. Appeals to reduce the significance of an inspection finding will be considered as having sufficient merit for review by this appeal process only if the contention falls into one of the following categories: (1) actual (verifiable) plant hardware, procedures, or equipment configurations were not considered by the staff; or (2) the staff's significance determination process was inconsistent with the applicable SDP guidance or lacked justification.

The NRC has also determined that the single point vulnerability within your offsite power transformer circuits and the associated failure to assure that applicable regulatory requirements and the design basis for safety-related systems were correctly maintained and controlled commensurate with the standards applied to the original design is a violation of 10 CFR Part 50, Appendix B, Criterion III, as cited in the enclosed Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in the subject inspection report.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response.

Because the finding involved an issue with low to moderate safety significance (White), the NRC would normally use the Reactor Assessment Program Action Matrix to determine the appropriate NRC response to the finding. However, the NRC may also exercise discretion and refrain from considering a safety significant finding in the assessment program if the finding involves design-related engineering calculations or analysis, associated operating procedures, or the installation of plant equipment. In addition, the finding must have been: (1) licensee- identified as a result of a voluntary initiative such as a design basis reconstitution; (2) corrected, or will be corrected, to include immediate corrective action and long term comprehensive corrective action to prevent recurrence, within a reasonable time following identification; (3) unlikely to have been previously identified by recent ongoing licensee efforts such as normal surveillance, quality assurance activities, or evaluation of industry information; and (4) not reflective of a current performance deficiency associated with existing licensee programs, policies, or procedures. In these cases, the NRC may characterize the finding as an "old design issue." A finding determined to be appropriately characterized as an "old design issue" will not be aggregated in the NRC Action Matrix with other performance indicators and inspection findings, nor will the finding individually result in a change from one column to another in the Reactor Assessment Program Action Matrix.

As documented in the subject inspection report, the NRC previously determined that: 1) the inspection finding should be considered licensee-identified as a result of the licensee's immediate review of an operating events experience; 2) the licensee took both immediate and long-term corrective actions to address the inspection finding; 3) the licensee's normal quality assurance and surveillance activities were not likely to have identified the vulnerability associated with the inspection finding; and 4) the performance errors that caused the inspection finding were not reflective of the licensee's existing programs, policies, or procedures. Therefore, based upon a consideration of the facts described above and in the subject inspection report, the NRC has determined that the inspection finding should be characterized as an "old design issue" and the NRC is exercising discretion to not consider the finding as a part of the Reactor Assessment Program. However, consistent with the guidance in IMC 0305, the NRC is considering an inspection, such as a supplemental inspection in accordance with NRC Inspection Procedure 95001, to review your root cause evaluation and corrective actions for the finding. We will notify you, by separate correspondence, of that determination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARs) component of the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (The Public Electronic Reading Room). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select **What We Do, Enforcement,** then **Significant Enforcement Actions.**

Sincerely,

/RA by Geoffrey Grant Acting for/

James L. Caldwell
Regional Administrator

Docket Nos. 50-373; 50-374
License Nos. NPF-11; NPF-18

Enclosure: Notice of Violation

cc w/encl:
Site Vice President - LaSalle County Station
LaSalle County Station Plant Manager
Regulatory Assurance Manager - LaSalle County Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional
Operating Group

Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional
Operating Group
Manager Licensing - Clinton and LaSalle
Senior Counsel, Nuclear, Mid-West Regional
Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission

NOTICE OF VIOLATION

Exelon Nuclear
Exelon Generation Company, LLC
LaSalle County Station, Units 1 and 2

Docket Nos. 50-373; 50-374
License No. NPF-11; NPF-18
EA-05-103

During an NRC inspection conducted from February 1 through May 31, 2005, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

Title 10 CFR Part 50, Appendix B, Criterion III, (Design Control) requires, in part, that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design.

10 CFR 50, Appendix A, General Design Criterion 17, (Electric Power Systems) requires, in part, that onsite electric power supplies, including the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Contrary to the above, the licensee made modifications to the emergency diesel generator (EDG) output circuit breakers that were completed on December 21, 1988, for Unit 2, Division 1; September 26, 1989, for Unit 1, Division 1; March 8, 1991, for Unit 1, Division 2; and February 1, 1992, for Unit 2, Division 2 that were not subject to design control measures commensurate with those applied to the original design. Specifically, the modifications introduced a single failure vulnerability such that a failure (i.e., open circuit) of the common current transformer circuit would have resulted in a loss of all alternating current, including the EDG supplied feeds, for the Division 1 and Division 2 safety buses on both units.

This violation is associated with a White Significance Determination Process finding.

Pursuant to the provisions of 10 CFR 2.201, Exelon Nuclear is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the LaSalle County Station, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-05-103" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

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

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

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

Dated this 7th day of September 2005

EA-05-157 - Kewaunee (Dominion Energy Kewaunee Inc.)

September 16, 2005

EA-05-157

Mr. David A. Christian
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION - FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT 05000305/2005014)

Dear Mr. Christian:

The purpose of this letter is to provide you with the final results of our significance determination of a finding which was described in Inspection Report 05000305/20050010, issued August 16, 2005, that involved the auxiliary feedwater (AFW) system design. Specifically, the AFW system design relied upon pump discharge pressure trip switches that would not have protected the pumps from air ingestion during natural events such as tornado and seismic events. In addition, the AFW system design would not have protected the pumps from "runout" conditions that may be encountered during other design and license basis scenarios. This finding was assessed using the Significance Determination Process (SDP) and was preliminarily characterized as White (i.e., a finding with low to moderate increased importance to safety, which may require additional NRC inspections).

In our letter dated August 16, 2005, the Nuclear Regulatory Commission (NRC) provided Dominion Energy Kewaunee Incorporated an opportunity to either request a Regulatory Conference to discuss this finding, or to explain your position in a written response. In a telephone conversation with Mr. T. Kozak of NRC, Region III, on August 26, 2005, Mr. M. Gaffney of your staff indicated that Dominion Energy Kewaunee Incorporated did not contest the characterization of the risk significance of this finding and declined the opportunity to discuss the issue in a Regulatory Conference or to provide a written response.

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

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC has also determined that the failure to provide adequate design control to ensure that the AFW pumps would be protected from failure due to air ingestion during tornado or seismic events, as well as from failure during potential "runout" conditions, is a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," as cited in the attached Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in Inspection Report 05000305/2005010. In accordance with the NRC Enforcement Policy, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response.

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARs) component of the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (The Public Electronic Reading Room).

Sincerely,

/RA by Geoffrey E. Grant for/

James L. Caldwell
Regional Administrator

Docket No. 50-305
License No. DPR-43

Enclosure: Notice of Violation

cc w/encl:

M. Gaffney, Site Vice President
C. Funderburk, Director, Nuclear Licensing
and Operations Support
T. Breene, Manager, Nuclear Licensing
L. Cuoco, Esq., Senior Counsel
D. Zellner, Chairman, Town of Carlton
J. Kitsembel, Public Service Commission of Wisconsin

NOTICE OF VIOLATION

Dominion Energy Kewaunee Inc.
Kewaunee Power Station

Docket No. 50-305
License No. DPR-43
EA-05-157

During an NRC inspection conducted from April 15 through July 29, 2005, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure that the design basis for safety-related functions of structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. Further, Criterion III requires that the design control measures shall provide for verifying or checking the adequacy of designs.
>
> Contrary to the above, prior to February 11, 2005, the licensee failed to implement design control measures to verify and check the adequacy of the auxiliary feedwater (AFW) system design to mitigate all postulated accidents. Specifically, the AFW system design relied upon pump discharge pressure trip switches that would not have protected the pumps from air ingestion during natural events such as tornado and seismic events. In addition, the AFW system design would not have protected the pumps from "runout" conditions that may be encountered during other design and license basis scenarios, including steam line breaks and station blackouts.

This violation is associated with a White Significance Determination Process finding.

Pursuant to the provisions of 10 CFR 2.201, Dominion Energy Kewaunee Inc. is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 2[illegible] with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the Kewaunee Power Station, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-05-157" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

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

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

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

Dated this 16th day of September 2005

EA-05-134 - Hatch 1 & 2 (Southern Nuclear Operating Company)

September 19, 2005

EA-05-134

Southern Nuclear Operating Company, Inc.
ATTN: Mr. H. L. Sumner
Vice President - Hatch Project
P. O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING (HATCH NUCLEAR PLANT INSPECTION REPORT NO. 05000321/200500009, 05000366/200500009)

Dear Mr. Sumner:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC) final significance determination for a finding involving the removal of the Technical Support Center (TSC) from service on April 25, 2005, to perform ventilation system modifications. The finding was documented in NRC Integrated Inspection Report No. 05000321/2005003 and 05000366/2005003, issued on July 8, 2005, and was assessed under the significance determination process as a preliminary White issue (i.e., an issue of low to moderate safety significance which may require additional NRC inspection). The cover letter to the inspection report informed Southern Nuclear Operating Company, Inc., (SNC) of the NRC's preliminary conclusion and provided SNC an opportunity to request a regulatory conference on this matter.

At SNC's request, an open regulatory conference was conducted on August 16, 2005, to discuss SNC's position on this issue. The enclosures to this letter include the list of attendees at the regulatory conference and material presented by SNC and NRC.

During the conference, SNC provided details related to its pre-modification activities, and its assessment of the significance of the issue. SNC stated that comprehensive preparations were planned and taken before commencement of modification activities such that key emergency response organization members would have been able to perform their tasks without compensatory measures from the main control room (MCR). SNC advised that the MCR would be used as the alternate location for TSC functions as this location was approved for use in the Emergency Plan, was evaluated by SNC as capable of being used successfully to execute TSC functions to support emergency response, and was reaffirmed in SNC's planning process prior to beginning the modification. Prior to taking the TSC out of service, SNC also reviewed procedures that governed the execution of TSC responsibilities and made procedural changes as necessary. Based on the foregoing, SNC concluded that the planning standard function was maintained and, correspondingly, that the finding was of very low safety significance (Green). SNC did not contest the NRC determination that the finding represented a violation of 10 CFR 50.54(q) and 10 CFR 50.47(b)(8). In addition, SNC provided details of its corrective actions in response to the finding.

After considering the information developed during the inspection and the information SNC provided at the conference, the NRC has concluded that the final inspection finding is appropriately characterized as White in the emergency preparedness cornerstone. In summary, the NRC concluded that the removal of the TSC from service for more than 7 days represented the loss of a planning standard function as described in NRC Inspection Manual Chapter 0609, Appendix B, Emergency Preparedness Significance Determination Process. Although the use of the MCR as an alternate TSC location during planned TSC outages is permitted by the Emergency Plan, the NRC considers this to be a temporary measure while repair activities proceed with high priority. Further, the Emergency Plan specifies that using the MCR as an alternate TSC is permitted only if the TSC becomes "uninhabitable during an emergency." In this case, the TSC did not become uninhabitable during an emergency, and SNC's original TSC outage schedule of approximately 5 weeks was not commensurate with the intent to proceed with high priority.

You have 10 calendar days from the date of this letter to appeal the staff's determination of significance for the identified finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC also determined that a violation of 10 CFR 50.54(q) and 10 CFR 50.47(b)(8) occurred because facilities and equipment to support the emergency response were not provided and maintained. The violation is set forth in the enclosed Notice of Violation.

You are not required to respond to this letter unless the description herein does not accurately reflect your position or if you choose to provide additional information. For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 05000321/200500009, 05000366/200500009, and the above violation is identified as VIO 0500321,366/200500009-01, Failure to Maintain Facilities and Equipment to Support Emergency Response. Accordingly, Apparent Violation 05000321,366/2005003-01, is closed.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (should you choose to provide one) will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS) which is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, any response should not include any personal privacy, proprietary, classified, or safeguards information so that it can be made available to the Public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select **What We Do, Enforcement,** then **Significant Enforcement Actions.**

Should you have any questions regarding this letter, please contact Mr. Malcolm Widmann, Chief, Branch 2, Division of Reactor Projects, at (404)562-4550.

Sincerely,

/RA/

William D. Travers
Regional Administrator

Docket Nos. 50-321 and 50-366
License Nos. DPR-57 and NPF-5

Enclosures:
1. Notice of Violation
2. List of Attendees
3. Material presented by SNC
4. Material presented by NRC

cc w/encls:

J. T. Gasser
Executive Vice President
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

George R. Frederick
General Manager, Plant Hatch

Reece McAlister
Executive Secretary
Georgia Public Service Commission
244 Washington Street, SW
Atlanta, GA 30334

Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

Raymond D. Baker
Manager Licensing - Hatch
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

Arthur H. Domby, Esq.
Troutman Sanders
Electronic Mail Distribution

Laurence Bergen
Oglethorpe Power Corporation
Electronic Mail Distribution

Director
Department of Natural Resources
205 Butler Street, SE, Suite 1252
Atlanta, GA 30334

Manager, Radioactive Materials Program
Department of Natural Resources
Electronic Mail Distribution

Chairman
Appling County Board of Commissioners
69 Tippins Street., Suite 201
Baxley, GA 31513

Resident Manager
Oglethorpe Power Corporation
Edwin I. Hatch Nuclear Plant
Electronic Mail Distribution

Senior Engineer - Power Supply
Municipal Electric Authority
of Georgia
Electronic Mail Distribution

NOTICE OF VIOLATION

Southern Nuclear Operating Company
Edwin I. Hatch Nuclear Plant
Units 1 and 2

Docket Nos. 50-321 and 50-366
License Nos. DPR-57 and NPF-5
EA-05-134

During an NRC inspection completed on June 30, 2005, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

10 CFR 50.54(q) requires, in part, that a licensee authorized to operate a nuclear power reactor shall follow and maintain in effect emergency plans which meet the standards in Section 50.47(b). 10 CFR 50.54(q) also states that a licensee may make changes to these plans without Commission approval only if the changes do not decrease the effectiveness of the plans and the plans, if changed, continue to meet the standards of Section 50.47(b).

10 CFR 50.47(b)(8) requires that adequate emergency facilities and equipment to support the emergency

response be provided and maintained. Section H of Revision 18 of the Edwin I. Hatch Nuclear Plant Emergency Plan, which implements the requirements of 10 CFR 50.47(b)(8), states that in the event that the Technical Support Center (TSC) becomes "uninhabitable during an emergency," the control room will serve as an alternate TSC location.

Contrary to the above, between April 25 and May 4, 2005, the licensee failed to maintain in effect a provision of its emergency plan in that adequate emergency facilities and equipment to support the emergency response were not provided. In this case, the licensee failed to follow and maintain in effect its emergency plan when the TSC was removed from service during this period to allow for modification activities. The removal of the TSC for the modification did not represent a condition in which the TSC was uninhabitable during an emergency.

This violation is associated with a White Significance Determination Process finding for Units 1 and 2 in the emergency preparedness cornerstone.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in the information provided by SNC at the conference (Enclosure 3). However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation - EA-05-134," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region II, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

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

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

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

Dated this 19th day of September 2005

EA-05-114 - Crystal River 3 (Florida Power Corp.)

September 21, 2005

EA 05-114

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing &
Regulatory Programs
15760 West Power Line Street
Crystal River, FL 34428-6708

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (CRYSTAL RIVER UNIT 3, NRC INSPECTION REPORT NO. 05000302/2005011)

Dear Mr. Young:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC) final significance determinati[on] for a finding involving unprotected post-fire safe shutdown cables and related non-feasible local manual operator actions. The finding was documented in NRC Inspection Report No. 05000302/2005007, issued on June 16, 2005, and was assessed under the significance determination process as a preliminary "greater than Green" issue (i.e., an issue of at least low to moderate safety significance which may require additional NRC inspection). The cover letter to the inspection report informed Florida Power Corporation (FPC) of the NRC's preliminary conclusion, provided FPC an opportunity to request a regulatory conference on this matter, and forwarded the details of the NRC's preliminary estimate of the change in core damage frequency (CDF) for this finding.

At FPC's request, an open regulatory conference was conducted on July 22, 2005, to discuss FPC's position on this issue. The enclosures to this letter include the list of attendees at the regulatory conference and material presented by FPC and NRC.

During the conference, FPC presented the results of its estimate of the increase in CDF due to the performance deficiency including influential assumptions and risk analysis methodology. FPC concluded that the finding was of very low safety significance. The critical aspects of FPC's analysis and inputs that differed from the NRC's preliminary estimate included the following: (1) fully developed fires would produce enough smoke to require extensive removal efforts with a gas-powered ejector (NOTE: FPC estimated that a sufficient amount of smoke would be removed within 20 minutes to allow an operator to reset the emergency diesel generator (EDG) lockout breaker in the 3B 4160-VAC switchgear compartment and recover the 4160-VAC electrical bus.); (2) FWP-7, the non-safety-related feedwater pump, and its associated power and control circuits would remain free from fire damage and could be started from the main control room to provide and maintain secondary side heat removal; (3) the EDGs could operate unloaded without incurring damage for at least 1 hour given the potential lack of room ventilation; (4) the emergency feedwater initiation control system (EFIC) would be available for at least 2 hours instead of 30 minutes as assumed in the NRC's preliminary estimate; and (5) FPC would use the Technical Support Center (TSC) to provide guidance to the operating and response staff for diverse emergency and auxiliary feedwater lineups and for electrical distribution alignment. FPC did not contest that the finding represented a violation of 10 CFR Part 50, Appendix R, Section III.G.2.

After considering the information developed during the inspection and the information FPC provided at the conference, the NRC has concluded that the final inspection finding is appropriately characterized as White in the mitigating systems cornerstone. In summary, the most critical differences between the NRC's assessment of the change in CDF and that of FPC's involved the likelihood of success of an operator action to reset the EDG lockout breaker to recover the 4160-VAC electrical bus and credit for use of FWP-7. The NRC ultimately concluded that the probability of failure to reset the EDG lockout was much greater than that assumed by FPC due to the extreme environmental conditions produced by the fire combined with the very poor ergonomics associated with accomplishing a task in this situation. Therefore, possible accomplishment of this task could not be considered until smoke removal efforts were successfully employed. In considering the use of FWP-7, the NRC agreed with FPC that some credit was warranted which would result in a reduction in the NRC's preliminary estimate.

Regarding other aspects of FPC's analysis, the NRC agrees with FPC that the EDG could operate unloaded for at least 1 hour without incurring damage and that EFIC would be available for at least 2 hours. Regarding the use of the TSC, the NRC concluded that the combination of time constraints, the complexity of the emergency situation, power/communications availability, and the variability in the actual TSC response precluded TSC credit.

You have 10 calendar days from the date of this letter to appeal the staff's determination of significance for the identified finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC also concluded that a violation of 10 CFR Part 50, Appendix R, Section III.G.2, occurred in that the protection and metering circuits were not physically separated or protected from fire damage as required. The violation is set forth in the enclosed Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in NRC Inspection Report No. 05000302/2005007 dated June 16, 2005. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in NRC Inspection Report No. 05000302/2004009 dated March 14, 2005; NRC Inspection Report No. 05000302/2005007 dated June 16, 2005; and the information provided by FPC at the July 22, 2005, regulatory conference (Enclosure 3). Therefore, you are not required to respond to this letter unless the description therein does not adequately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 05000302/200500011, and the above violation is identified as VIO 0500302/200500011-01, Unprotected Post-Fire Safe Shutdown Cables and Related Non-feasible Local Manual Operator Action. Accordingly, Apparent Violation 05000302/2005007-01 is closed.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (should you choose to provide one) will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), which is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, any response should not include any personal privacy, proprietary, classified, or safeguards information so that it can be made available to the Public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select **What We Do, Enforcement,** then **Significant Enforcement Actions.**

Should you have any questions regarding this letter, please contact Mr. D. Charles Payne, Chief, Engineering Branch 2, Division of Reactor Safety, at (404)562-4669.

Sincerely,

/RA/

William D. Travers
Regional Administrator

Docket No.: 50-302
License No.: DPR-72

Enclosures:

1. Notice of Violation
2. List of Attendees
3. Material presented by FPC
4. Material presented by NRC

cc w/encls:

Daniel L. Roderick
Director Site Operations
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

Jon A. Franke
Plant General Manager
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

Richard L. Warden
Manager Nuclear Assessment
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

Michael J. Annacone
Engineering Manager
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

R. Alexander Glenn
Associate General Counsel (MAC - BT15A)
Florida Power Corporation
Electronic Mail Distribution

Steven R. Carr
Associate General Counsel - Legal Dept.
Progress Energy Service Company, LLC
Electronic Mail Distribution

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, FL 32304

William A. Passetti
Bureau of Radiation Control
Department of Health
Electronic Mail Distribution

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Department of Community Affairs
Electronic Mail Distribution

Chairman
Board of County Commissioners
Citrus County
110 N. Apopka Avenue
Inverness, FL 36250

Jim Mallay
Framatome Technologies
Electronic Mail Distribution
FPC

NOTICE OF VIOLATION

Florida Power Corporation
Crystal River Nuclear Plant
Unit 3

Docket No. 50-302
License No. DRP-72
EA-05-114

During an NRC inspection completed on June 8, 2005, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR 50.48(b)(1) requires, in part, that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G, Fire Protection of Safe Shutdown Capability.
>
> Section III.G.2 states that, except as provided for in Section III.G.3, where cables or equipment (including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground) of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:
>
> a. separation of cables and equipment and associated non-safety circuits of redundant trains by a fire barrier having a 3-hour rating (Structural steel forming a part of or supporting such fire barriers shall be protected to provide fire resistance equivalent to that required of the barrier.);
>
> b. separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards (In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area.); or
>
> c. enclosure of cable, equipment, and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating. (In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area.)
>
> Contrary to the above, on January 26, 2005, the licensee failed to ensure that one of the redundant trains of systems necessary to achieve and maintain hot shutdown conditions would be free of fire damage via one of the three means specified in 10 CFR Part 50, Appendix R, Section III.G.2. Specifically, cables for the electrical protection and metering circuit located in the 3A 4160-V engineered safeguards (ES) switchgear room were vulnerable to fire damage that could disable both the 3A 4160-V ES switchgear and the redundant train 3B 4160-V ES switchgear resulting in a loss of all safety-related alternating current power.
>
> This violation is associated with a White Significance Determination Process finding for Unit 3 in the mitigating systems cornerstone.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in NRC Inspection Report No. 05000302/2004009 dated March 14, 2005; NRC Inspection Report No. 05000302/2005007 dated June 16, 2005; and the information provided by FPC at the July 22, 2005, regulatory conference (Enclosure 3). However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation - EA-05-114," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region II, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

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

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

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

Dated this 21st day of September 2005

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

September 16, 2005

NRC INFORMATION NOTICE 2005-26: RESULTS OF CHEMICAL EFFECTS HEAD LOSS TESTS IN A SIMULATED PWR SUMP POOL ENVIRONMENT

ADDRESSEES

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

PURPOSE

The U.S. Nuclear Regulatory Commission is issuing this information notice to inform addressees about recent NRC-sponsored research results related to head loss from chemical effects in a simulated PWR sump pool environment. The NRC anticipates that recipients will review the information for applicability to their facilities and consider taking actions, as appropriate, to avoid similar issues. However, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Generic Safety Issue (GSI) 191 addresses the potential for debris accumulation on PWR sump screens to affect emergency core cooling system (ECCS) pump net positive suction head margin. The NRC has issued Bulletin 2003-01, "Potential Impact of Debris Blockage On Emergency Sump Recirculation At Pressurized Water Reactors," and Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents At Pressurized Water Reactors," related to the GSI-191 resolution.
GL 2004-02 requests, in part, that licensees evaluate the maximum head loss postulated from debris accumulation (including chemical effects) on the submerged sump screen. Chemical effects are corrosion products, gelatinous material, or other chemical reaction products that form as a result of interaction between the PWR containment environment and containment materials after a loss-of-coolant accident (LOCA). NRC and the nuclear industry jointly developed an integrated chemical effects test (ICET) program to determine if chemical reaction products can form in representative PWR post-LOCA containment sump environments. These tests were conducted by Los Alamos National Laboratory at the University of New Mexico. The ICET series involved five tests, each representing a different subset of expected post-LOCA environments within existing PWR plants. Although chemical products were observed in all of

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the ICET environments, the head loss associated with these products was not evaluated as it was outside the scope of the ICET program. NRC initiated additional testing to obtain some insights on the head loss associated with chemical products that may form in PWR sump pools.

Head loss testing is being performed at the Argonne National Laboratory. Initial testing has been done in a piping loop containing a simulated sump pool environment intended to represent the ICET Test 3 conditions. ICET Test 3 was performed in a borated water environment containing trisodium phosphate (TSP), various metallic and non-metallic sample coupons representative of containment materials, and a mixture of insulation (80% calcium silicate, 20% fiberglass) samples. This environment was selected for initial head loss testing based on the early formation of chemical product during ICET Test 3 and the characteristics of this product observed during and after this test (NRC ADAMS Package Accession Number ML052140490). During initial testing to simulate these observed products, significant head loss was measured across a test screen containing a preexisting fiber bed. The Argonne tests and initial test results are described in detail in the attachment, "Chemical Effects/Head Loss Testing Quick Look Report, Tests 1 and 2," dated September 16, 2005.

DISCUSSION

As part of the GL 2004-02 response, licensees are required to evaluate the sump screen head loss consequences of any chemical effects in an integrated manner with other postulated post-LOCA conditions. These recent research results indicate that a simulated sump pool environment containing phosphate and dissolved calcium can rapidly produce a calcium phosphate precipitate that, if transported to a fiber bed covered screen, produces significant head loss. The attachment report contains several interesting observations:

- Significant head loss was observed in tests combining TSP with a higher concentration of dissolved calcium (simulating the ICET Test 3 environment) and in tests with TSP and lower dissolved calcium concentrations (i.e., less than the ICET 3 environment).

- Small-scale leaching tests were done with calcium silicate insulation. The amount of calcium that will dissolve appears to depend more on the initial pH of the solution than on the amount of calcium-silicate insulation placed into solution. Lower initial pH solutions produced greater amounts of dissolved calcium.

- The amount of calcium phosphate precipitant in an ICET Test 3 type environment may be limited by the amount of phosphate available from the TSP.

This information is relevant to plants containing phosphate (e.g., plants using TSP as a sump pool buffering agent) and calcium sources (e.g., insulation, concrete) that may dissolve within the post-LOCA containment pool with sufficient concentrations to form calcium phosphate precipitate. These test results indicate that substantial head loss can occur if sufficient calcium phosphate is produced in a sump pool and transported to a preexisting fiber bed on the sump screen.

Although significant increases in head loss were observed due to chemical effects in these tests, it is important to note that these head loss results were obtained in a recirculating test loop not intended to be prototypical of a PWR plant containment. For example, the calcium phosphate precipitant was formed by introducing calcium chloride into a TSP buffered solution immediately upstream and at a higher elevation than a screen with a preestablished fiber bed. The test loop orientation and method of calcium introduction result in transport of virtually all chemical products to the fiber bed covered screen. Parameters that may influence head loss in these tests include screen approach velocity, fiber bed thickness, relative arrival times for debris and chemical precipitates, and loop fluid recirculation time. Applicability of these results to plant specific environments may also be affected by these and other variables (e.g., insulation materials, break location, and sump design).

The NRC is continuing head loss testing in simulated PWR sump pool environments that use other chemical species to buffer pH.

CONTACTS

This information notice does not require any 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/
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Attachment: Chemical Effects/Head-Loss Testing Quick Look Report, Tests 1 and 2

Chemical Effects/Head–Loss Testing
Quick Look Report, Tests 1 & 2
September 16, 2005

J. Oras, J.H. Park, K. Kasza, K. Natesan, and W. J. Shack
Argonne National Laboratory

Abstract

This report describes the results of preliminary tests to determine the potential for chemical products observed in the third Integrated Chemical Effects Test (ICET–3) to increase the head–loss associated with sump screen debris beds. The first test was intended to simulate the conditions in ICET–3. The second test was parametric and intended to determine the effect of a range of chemical product loadings on head–loss. With a pre-existing physical debris bed approximately 16 mm (5/8 in) thick consisting of equal weights of NUKON fiber and CalSil insulation, a large increase in head loss was observed for the chemical product loading intended to simulate ICET–3 conditions. In the parametric test with a similar physical debris bed, increases in head–loss were observed for a chemical product loading one-twentieth of the simulated ICET–3 conditions.

Background - Integrated Chemical Effects Tests (ICET)

The ICET project is a joint effort by the U.S. Nuclear Regulatory Commission (NRC) and the nuclear utility industry, undertaken through a memorandum of understanding between the NRC and the Electric Power Research Institute (EPRI).[1] The ICET tests simulate the chemical environment present inside a containment water pool after a loss-of-coolant accident. The chemical systems were monitored for an extended time to identify the presence, composition, and physical characteristics of any chemical products that form during the test. The ICET test series was conducted by Los Alamos National Laboratory (LANL) at the University of New Mexico (UNM).

The containment pool environments selected for study were based on input from the Westinghouse Electric Company, the NRC and EPRI. The specific conditions, material types, and parameters in the ICET test series are intended to be broadly representative of all domestic PWRs. The Westinghouse Owners Group and the Babcock & Wilcox Owners Group aided in soliciting information. To obtain the necessary details of plant-specific conditions within containment (materials present, containment pool conditions, etc.), Westinghouse reviewed plant-specific documents, (such as Post-LOCA Hydrogen Generation Evaluations), other available plant documents (e.g., updated final safety analysis reports), and submitted survey questions to plant personnel. The plant survey responses formed the primary source of data for determining the parameters used to define the ICET test conditions.[2]

The third ICET test, ICET–3, investigated the chemical behavior of boric acid/LiOH solutions containing 80% calcium silicate/20%fiberglass insulation with a trisodium phosphate (TSP) buffering agent to obtain a target pH of 7. The steam generator is the largest plant component that may have fiberglass or calcium silicate insulation and that might be affected by a postulated large break LOCA. Based on the dimensions of a representative steam generator and accounting for a conservatively–large zone–of–influence (ZOI) volume, Westinghouse estimated that the volume of fiberglass or calcium silicate insulation debris that could be generated is 141.6 m^3 (5,000 cubic feet). The smallest containment pool volume, based on the survey information, is about 36,500 ft^3 so a conservative estimate of the insulation debris per volume of containment pool fluid is 0.137 ft^3/ft^3. For insulation with 80% CalSil, this gives about 25 g/l of CalSil if all the CalSil debris is assumed to be immersed in the sump fluid.

The ICET–3 test was performed by first adding boric acid (2800 ppm) and LiOH (3 ppm) to water in the ICET tank. HCl was also added to simulate degradation of electrical insulation. The resultant pH of this solution was 4.2. CalSil corresponding to about 20 g/l was placed in the submerged portion of the ICET tank; the remainder of the CalSil was only wetted by the sprays during the initial 4 hours of testing. The circulation pump was then turned on and the solution was allowed to circulate for about 5 hours. At this time, a solution containing dissolved TSP was metered into the test chamber solution over the next four hours. Within 20 minutes after the beginning of metering TSP into solution, a white flocculent precipitate was observed in the tank. The precipitate appeared to be neutrally buoyant. The precipitate was presumed to be calcium phosphate. Subsequent analysis showed that substantial amounts of calcium phosphate are present in the precipitates, although other products could also be present. It has also not yet been determined which specific varieties of calcium phosphates are present in the ICET-3 products. Hereafter in this report, this ICET-3 product is generically referred to as $Ca_3(PO_4)_2$, tricalcium phosphate, for convenience.

Small Scale Dissolution Test Results

No measurements were made of the dissolved calcium levels in ICET–3 at times prior to the addition of the TSP through the sprays. Therefore, small scale dissolution experiments were performed using additions of CalSil to solutions containing 2800 ppm boric acid and 3 ppm LiOH to estimate the dissolved Ca level initially present in ICET-3. In some cases, small amounts of HCl (to simulate breakdown of electrical cables) were added since these were included in the ICET–3.

Small scale dissolution tests were performed for a range of CalSil concentrations that encompassed the ICET–3 conditions for durations ranging from 35 minutes to 24 hours. The results of these tests are summarized in Table 1. The Ca concentration values in this table are the dissolved Ca levels at the end of the leaching period. The "initial pH" values given in the table are the initial starting pH of the solution before the CalSil addition while the "final pH" values represent the pH at test termination. The CalSil dissolution raises the pH of initially acidic solutions to near pH 7 due to the hydrolysis of potassium and sodium released from the CalSil. However, the pH of the solutions already buffered with either NaOH or LiOH did not vary much upon CalSil dissolution. The amount of dissolved Ca most strongly depends on the initial pH of the solution as seen in Fig. 1 while the initial CalSil loading has very little effect for the loadings examined in this study.

Dissolution of the CalSil is more rapid in initially acidic solutions, but will occur even in near neutral and buffered solutions as seen in Table 1. In ICET–3, in which the initial solution was acidic, the phosphate was exhausted by the end of the second day. Thus in ICET-3, the amount of calcium phosphate formed is ultimately limited by the amount of TSP available. Assuming the product is $Ca_3(PO_4)_2$, 1 mole of TSP can consume 1.5 moles of $CaSiO_3$. Because CalSil is mostly $CaSiO_3$, this implies that the formation of precipitate will be phosphate limited for CalSil loadings down to about 2 g/l (for TSP additions of 4 g/l), but this could vary somewhat depending on the actual calcium phosphate species that form.

Based on the small scale test results, the dissolved Ca level in the ICET–3 tests before the start of the TSP injection is estimated to be about 200 ppm. Because not all the phosphate is consumed by this amount of dissolved Ca, only the initial "burst" of $Ca_3(PO_4)_2$ formation (within the first four hours of the ICET simulated post-LOCA environment) is associated with this inventory of dissolved Ca. As noted previously, additional $Ca_3(PO_4)_2$ likely continued to form in the ICET-3 test until all the phosphate was depleted.

Table 1. Small Scale Dissolution Tests on CalSil

No.	Test Conditions					Dissolved Ca (ppm)	Notes
	Initial pH (RT)	T(C)	Time	CalSil g/l	Final pH (RT)		
1	4.0	60	35 min	6	7.5	176	Solution pH = 4.0 made from B(OH)3 + Li(OH) + HCl
2	4.0	60	35 min	15	6.9	256	
3	4.0	60	35-min	25	6.7	244	
4	4.0	60	35-min	166	6.5	228	
5	4.0	60	4-h	6	6.7	196	
6	4.0	60	4-h	15	6.9	195	
7	4.0	60	4-h	25	7.1	195	
8	4.0	60	4-h	166	7.7	168	
9	4.5	60	4-h	6	6.7	156	Solution pH = 4.5 made from B(OH)3 + Li(OH) + HCl
10	4.5	60	4-h	15	6.9	169	
11	4.5	60	4-h	25	7.1	184	
12	4.5	60	4-h	166	8.0	127	
13	7.0	62	4-h	2	7.1	45	Solution pH =7 made by B(OH)3 + Li(OH) + HCl + NaOH addition (No TSP added)
14	7.0	62	4-h	6	7.4	88	
15	7.0	62	4-h	25	7.2	69	
16	7.0	62	24-h	2	7.2	73	
17	7.0	62	24-h	6	7.3	108	
18	7.0	62	24-h	25	7.4	102	
19	10.1	60	3.5-h	6	10.0	17	Solution pH = 10.0 made by B(OH)3 + Li(OH) + HCl + LiOH excess addition (No TSP added)
20	10.1	60	3.5-h	15	10.0	18	
21	10.1	60	3.5-h	25	10.0	20	
22	10.1	60	3.5-h	166	9.7	23	

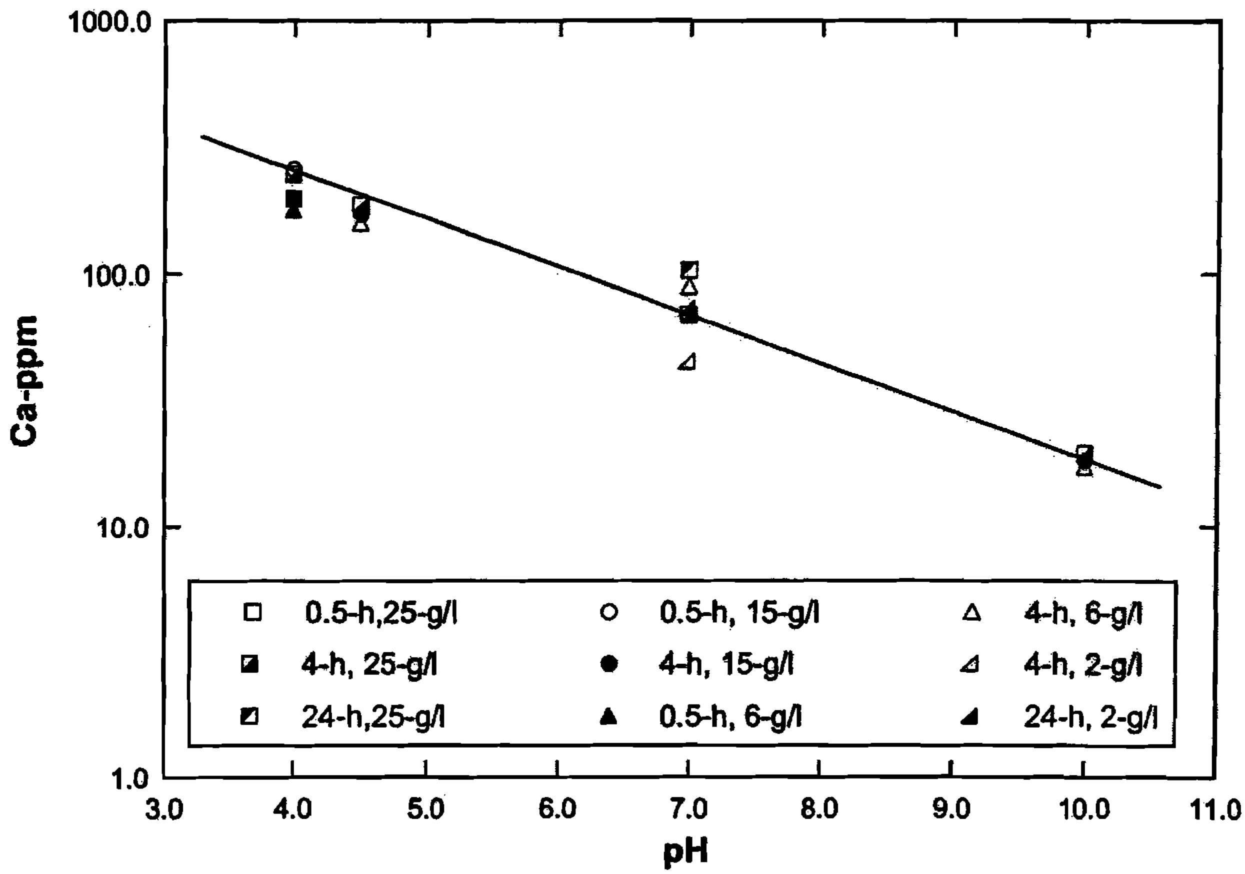

Figure 1. Dependence of dissolved Ca level on initial solution pH

ANL Test Facility

A schematic of the ANL test loop is shown in Fig. 2. The test screen has an effective diameter of 6 in. The fluid volume in the loop is 4.2 ft^3. At 0.1 ft/s, the transit time around the loop is about 4 minutes. For these tests, a perforated plate with a 51% flow area and staggered 3/16 in. holes was installed in the test–section. The test screen is shown in Fig. 3. In scaling results from the ANL test facility, the mass of chemical product per unit area of screen must be considered. The amount of chemical product produced scales with fluid volume while the screen area per fluid volume determines the product mass per unit screen area.

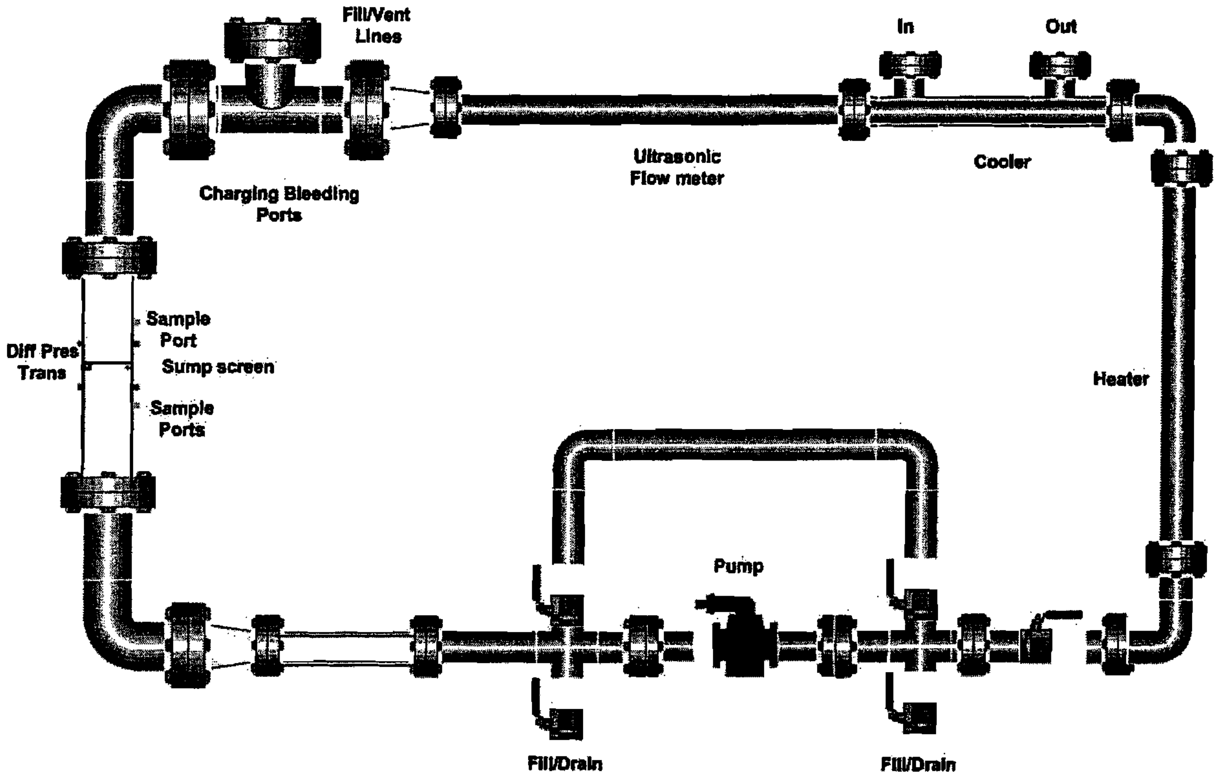

Figure 2. Schematic of the test loop

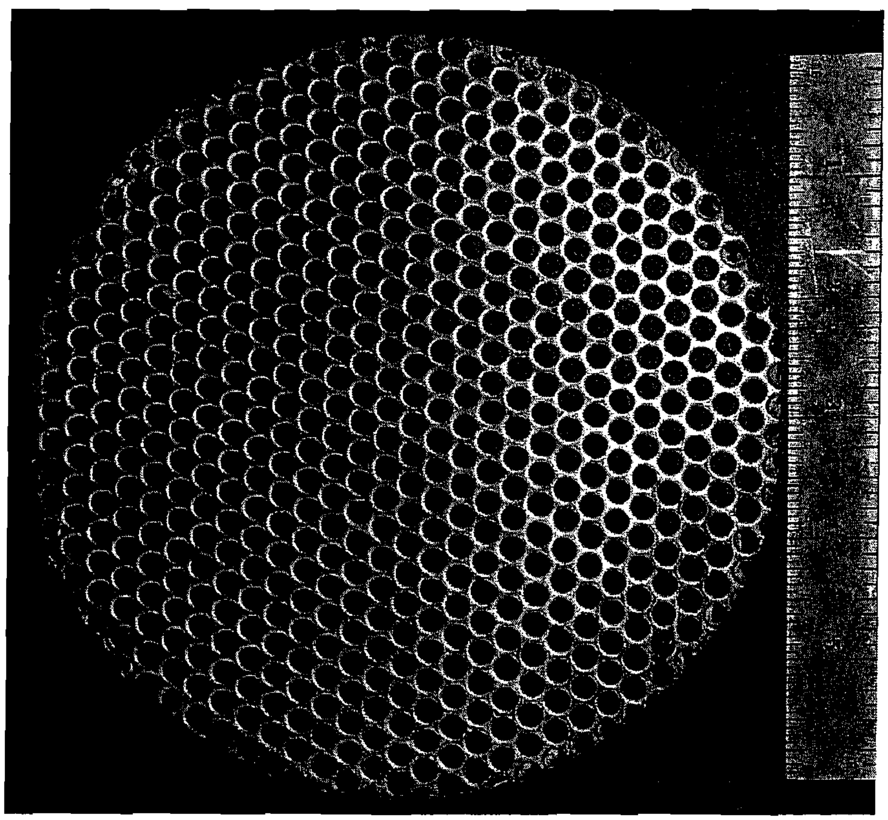

Figure 3. Perforated plate test screen

Head Loss Test #1 Results

The initial tests in the Argonne National Laboratory (ANL) chemical effects/head–loss testing program were intended to investigate the potential head loss associated with the chemical products observed in the third Integrated Chemical Effects Test (ICET–3).

In the ICET–3 tests, the TSP was added to the CalSil solution through the sprays. In the ANL tests, the loop is filled with a solution containing boric acid, LiOH, and TSP. The concentration of TSP corresponds to that metered into the test solution over 4 hours in ICET–3 (about 4 g/l). Calcium chloride ($CaCl_2$) solution is then added to supply the desired inventory of dissolved Ca. In the first head loss test, the Ca inventory was taken to be that corresponding to the estimated Ca concentration in the ICET solution at the start of the TSP spray, which, as discussed previously, has been estimated to be about 200 ppm. As noted previously, this will result in the formation of an amount of $Ca_3(PO_4)_2$ per volume of solution comparable to that observed in the initial stages of ICET–3.

The loop was filled with deionized water and heated to 130°F. Boric acid in powder form was slowly added to the loop and circulated until it was dissolved. The LiOH and TSP were added as solutions. The concentrations of these chemicals in the loop were also chosen to match those in ICET–3. The test temperature was lower than that in ICET–3 (140°F), because the test loop was not fully insulated. Because of the retrograde solubility of $Ca_3(PO_4)_2$, the lower temperature results in the formation of slightly less precipitate.

After the chemical solution was prepared, the physical debris bed was built by adding a slurry containing 15 g NUKON/15 g CalSil to the loop with the loop flow at 0.1 ft/s. The bed was about 3/4 in thick. The NUKON bed formed essentially in the first pass of the debris past the test screen. The pressure drop across the bed slowly increased as the test loop solution recirculated, presumably due to increasingly effective filtration of fine CalSil particles. After recirculating for about 45 minutes, the flow rate was then increased to 0.2 ft/s. At this flow rate, the bed compressed to about 5/8 in thick. The flow rate was then reduced back to 0.1 ft/s. The pressure drop and flow velocity at each stage of the debris bed formation is shown in Fig. 4. The physical debris bed at this point in the test is shown in Fig. 5.

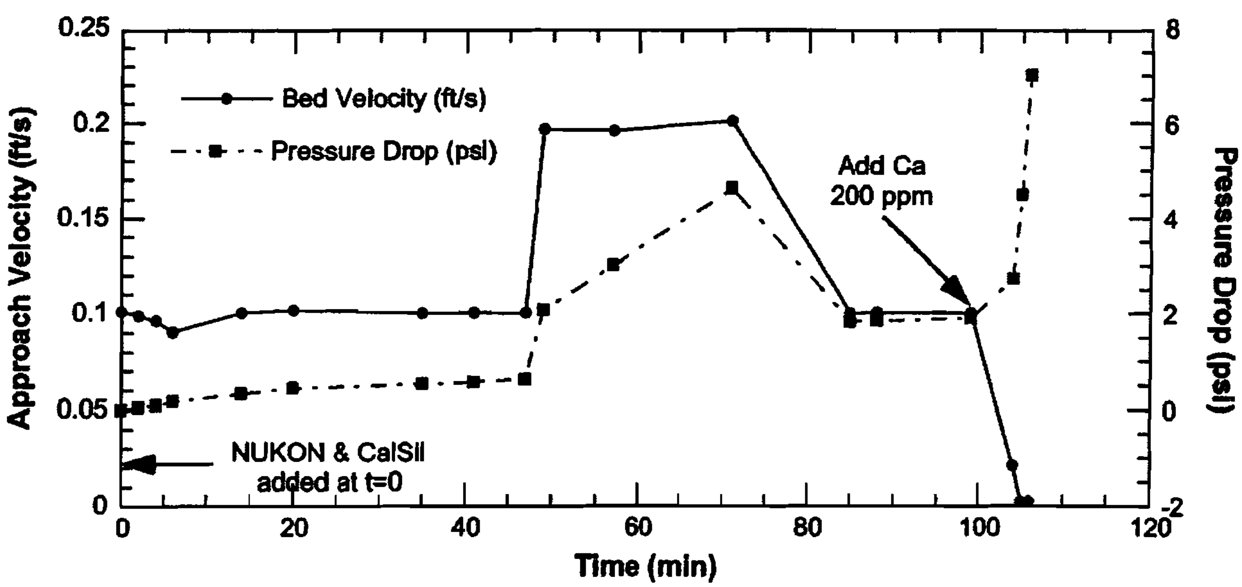

Figure 4. Flow rate and pressure drop as a function of time in Test 1.

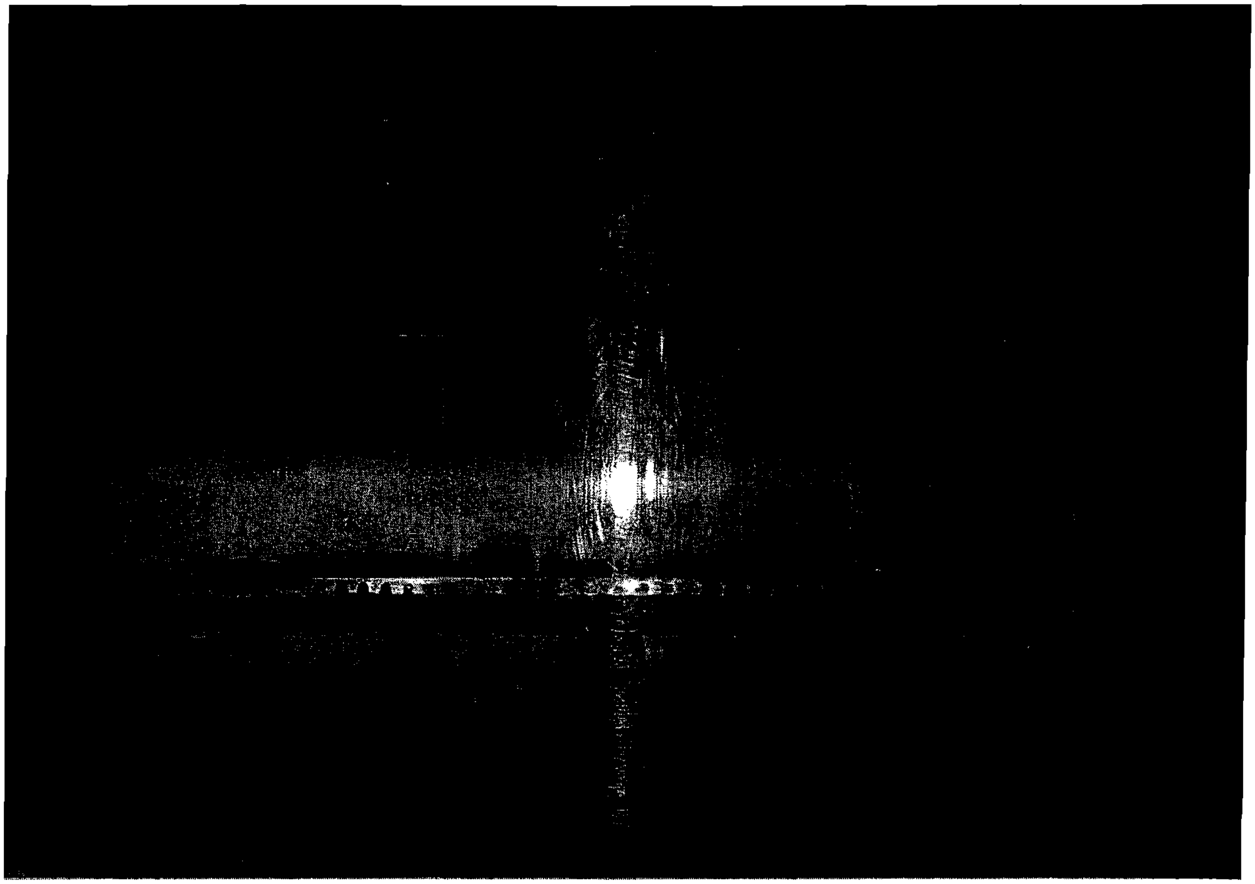

Figure 5. NUKON/CalSil bed before formation of the $Ca_3(PO_4)_3$ precipitate

The $CaCl_2$ was then added to the vertical part of the test loop just above the clear test section. A total of 400 ml of $CaCl_2$ solution was added over a 4 minute period (the transit time around the loop at 0.1 ft/s) to obtain the 200 ppm dissolved Ca inventory. A fine, milky precipitate was observed as shown in Figure 6 just after the introduction of the $CaCl_2$. The pressure drop across the bed increased from 1.7 psi to greater than 7.0 psi within 10 minutes of introducing the $CaCl_2$. An accurate pressure drop measurement could not be obtained beyond this point, because the loop was running unpressurized, and the pump started to cavitate as the precipitate continued to accumulate on the bed. The flow rate and pressure drop as a function of time after $CaCl_2$ addition are also shown in Fig. 4. As discussed previously, the 200 ppm Ca inventory is likely not sufficient to produce the full amount of $Ca_3(PO_4)_2$ formed during ICET–3. However, no additional Ca was added to simulate the depletion of all the available phosphate as in ICET–3, since the pressure drop across the bed had already caused the pump to cavitate. Figure 7 shows the accumulation of the precipitate on the debris bed just before the pump was shut off.

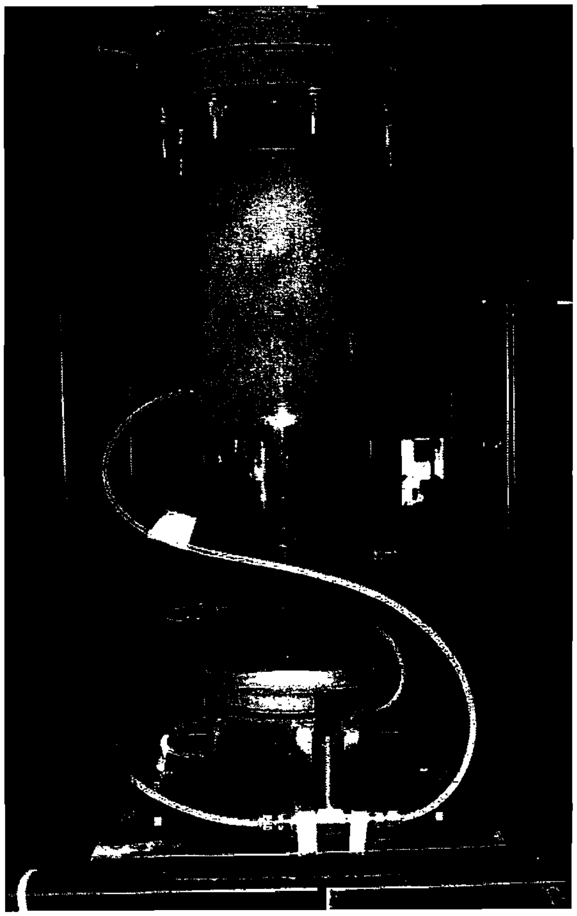

Figure 6.
$Ca_3(PO_4)_2$ forming after addition of $CaCl_2$ and approaching the debris bed.

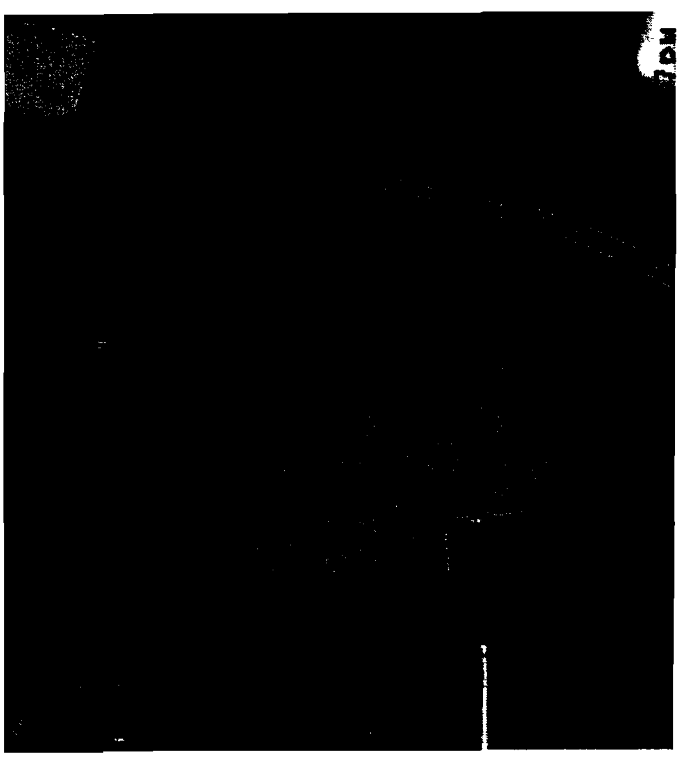

Figure 7.
Precipitate buildup on the fiber debris bed just after the pump was turned off.

Head Loss Test #2 Results

The initial procedure for the second test was similar to the first test. The loop was filled with deionized water and heated to 130°F. Boric acid in powder form was slowly added to the loop and circulated until it was dissolved. The LiOH and TSP were added as solutions.

The physical debris bed was again built from 15 g of NUKON and 15 g of CalSil. The bed was built at 0.1 ft/s and the flow rate was not increased above this value in contrast to the previous test. The debris bed was somewhat thinner than the initial debris bed for Test #1 at 0.1 ft/s (5/8 in for Test #2 and 3/4 in for Test #1). The pressure drop across the bed was also slightly smaller at this flow rate (0.4 psi in Test #2 and about 0.6 psi for Test #1).

For this test, the $CaCl_2$ additions were made in stepwise fashion starting with an initial addition equivalent to 10 ppm (one-twentieth of the simulated ICET-3 inventory) of dissolved Ca. Then amounts were added incrementally corresponding to total dissolved Ca inventories of 25 ppm, and 50 ppm. Each addition was metered in over a 4 minute period as in the first test.

When $CaCl_2$ equivalent to an inventory of 10 ppm dissolved Ca in the loop volume was added, the pressure drop at a flow rate of 0.1 ft/s increased from 0.4 psi to 1.4 psi. The $Ca_3(PO_4)_2$ precipitate was again visible, but the cloud was much fainter than the previous test which had a 200 ppm Ca inventory. Additional $CaCl_2$ was then added to simulate a 25 ppm inventory. The pressure drop increased from 1.4 psi to 6.4 psi and the pump again started to cavitate, since the test loop was unpressurized. The velocity was then decreased to 0.01 ft/s at which point the pressure drop decreased to 0.5 psi. A final increment of $CaCl_2$ was added to simulate a 50 ppm inventory of total dissolved Ca. At a flow rate of 0.01 ft/s, the pressure drop increased from 0.5 psi to 1.0 psi within 4 minutes. Under continuing operation for another 12 minutes, the pressure drop increased to 5.2 psi, but the velocity could not be maintained as the suction pressure on the pump dropped. The flow rate and pressure drop as a function of time in Test 2 are shown in Fig. 8.

An interesting qualitative difference was noted between the $CaCl_2$ additions at flow rates of 0.1 ft/s and those at 0.01 ft/s. At 0.1 ft/s, the precipitate was a finely dispersed milky cloud. At 0.01 ft/s, these particles seemed to agglomerate into light, flocculent assemblies up to perhaps 0.25 in. in diameter as shown in Fig. 9. These larger assemblies appear similar to the material observed in the ICET–3 tank where velocities are likely lower than 0.1 ft/s.

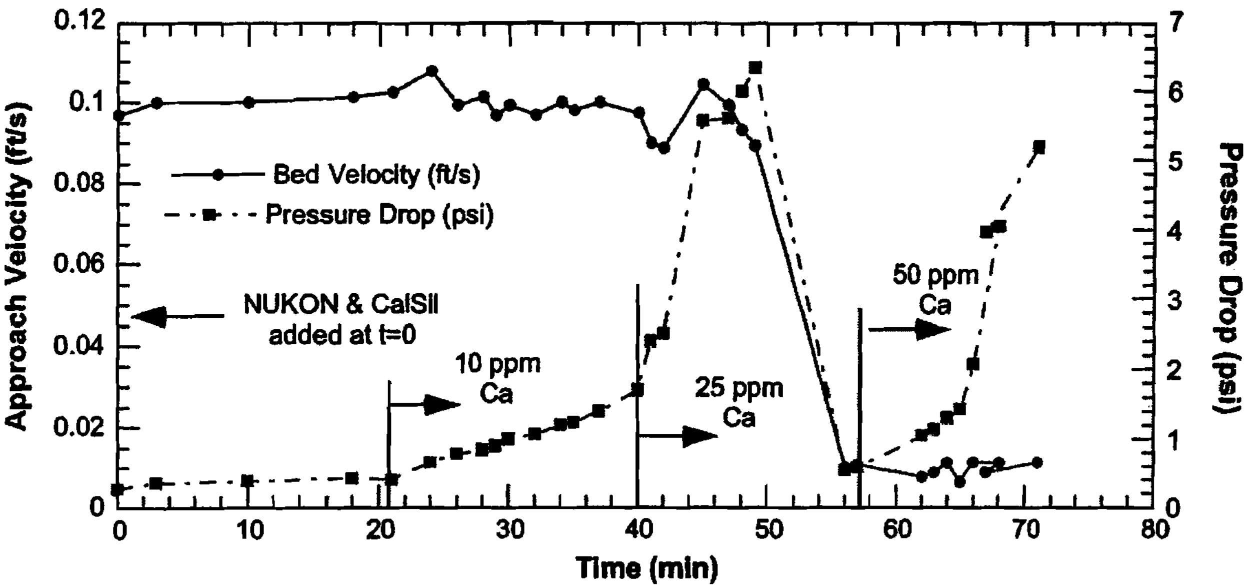

Figure 8. Flow rate and pressure drop as a function of time in Test 2.

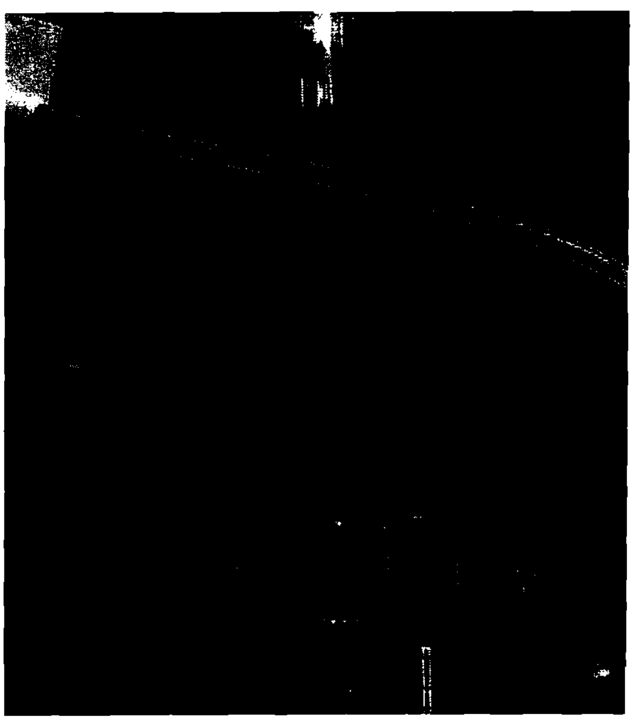

Figure 9.
Flocculent precipitates observed at 0.01 ft/s in Test 2

References

1. For more information on the ICET program see: http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance/tech-references.html

2. *Test Plan: Characterization of Chemical and Corrosion Effects Potentially Occurring Inside a PWR Containment Following a LOCA*, Prepared by Timothy S. Andreychek, Westinghouse Electric Company (ADAMS ML052100426).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

September 26, 2005

NRC REGULATORY ISSUE SUMMARY 2005-20: REVISION TO GUIDANCE FORMERLY CONTAINED IN NRC GENERIC LETTER 91-18, "INFORMATION TO LICENSEES REGARDING TWO NRC INSPECTION MANUAL SECTIONS ON RESOLUTION OF DEGRADED AND NONCONFORMING CONDITIONS AND ON OPERABILITY"

ADDRESSEES

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

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this Regulatory Issue Summary (RIS) to inform licensees that it has revised the guidance contained in two sections of NRC Inspection Manual Part 9900, Technical Guidance, "Operable/Operability: Ensuring the Functional Capability of a System or Component" and "Resolution of Degraded and Nonconforming Conditions," and has combined these two documents into a single document. The revised inspection guidance reflects relevant changes that have been made to NRC regulations, policies, and practices, and clarifies selected issues based on operating experience. This RIS requires no action or written response on the part of an addressee.

BACKGROUND INFORMATION

The NRC staff inspection guidance contained in the two NRC Inspection Manual sections described above were initially provided to licensees in Generic Letter (GL) 91-18, issued on November 7, 1991. The NRC staff revised the guidance in NRC Inspection Manual Part 9900, Technical Guidance, "Resolution of Degraded and Nonconforming Conditions," and issued it in Revision 1 of GL 91-18 on October 8, 1997. The purpose of Revision 1 of GL 91-18 was to more explicitly discuss the role of the 10 CFR 50.59 evaluation process in the resolution of degraded and nonconforming conditions.

In the summer of 2003, the NRC staff sought public comment on the technical guidance, which included holding a public workshop in August 2003. The staff revised the guidance based on the inputs received, and held a second public workshop to discuss it in August 2004. Subsequently, the NRC staff met several times in 2005 with an industry task force formed by the Nuclear Energy Institute (NEI), and resolved the comments received from various stakeholders.

ML052020424

SUMMARY OF ISSUE

Attached is a revised NRC Inspection Manual, Part 9900, Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety." This guidance supercedes the guidance previously provided in GL 91-18 and Revision 1 to GL 91-18.

The attached inspection manual section provides guidance to NRC inspectors for reviewing the actions of licensees pertaining to the operability of structures, systems, and components (SSCs) following the discovery of degraded and nonconforming conditions in SSCs. However, many licensees have found NRC's guidance to be very useful in developing their plant-specific processes, and therefore the NRC staff is communicating it to licensees as a RIS.

The NRC revised its inspection guidance to reflect ongoing regulatory changes, including implementation of the revised reactor oversight process, the requirement that licensees appropriately assess and manage risk related to proposed maintenance activities (10 CFR 50.65(a)(4)), and implementation of the revised change control process in 10 CFR 50.59, "Changes, Tests and Experiments." The revision also clarifies selected issues in the guidance based on operating experience and industry feedback.

In addition, the NRC concluded that the two inspection manual documents were closely related. The NRC staff therefore combined the documents, and at the same time re-wrote them to make them clearer and more process-oriented. However, the NRC understands that licensees may collectively refer to the processes described in the revised Part 9900 as the "GL 91-18 process" or the "operability determination process (ODP)."

BACKFIT DISCUSSION

This RIS requires no action or written response and, therefore, is not a backfit under 10 CFR 50.109. Consequently, the staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment was published in the *Federal Register* on August 3, 2004 (69 FR 46599), to give interested parties an opportunity to suggest ways for improving the guidance. The staff concludes that this RIS and the attached NRC inspection guidance are informational and pertain to a staff position that does not represent a departure from current regulatory requirements and practices.

SMALL BUSINESS REGULATORY ENFORCEMENT FAIRNESS ACT OF 1996

This RIS is not a "rule" as defined in 5 U.S.C. 804 and therefore is not subject to the Congressional review provisions of the Small Business Regulatory Enforcement Fairness Action of 1996.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not contain any information collections and, therefore, is not subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). The information collection requirements referenced in Manual Chapter 9900 are approved by the Office of Management and Budget approval number 3150-0011 which expire February 28, 2007. The NRC may not conduct or sponsor, and a person is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

CONTACT

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

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Attachment: NRC Inspection Manual Part 9900: Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse To Quality or Safety"

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

NRC Yellow Announcement

UNITED STATES NUCLEAR REGULATORY COMMISSION

Announcement No. 065

Date: September 28, 2005

To: **All NRC Employees**

SUBJECT: NRR MANAGEMENT CHANGES

The Office of Nuclear Reactor Regulation is reorganizing in order to prepare for the anticipated increase in the new reactor licensing workload and to better align the organization for risk-informed regulation. (See attachment for reorganization chart).

Bill Borchardt remains the Office Deputy Director. Cynthia Carpenter will remain Director of Program Management, Policy Development and Planning Staff.

As part of the reorganization, three Associate Directors will report to the Office Director as follows: the Associate Director for Engineering and Safety Systems, the Associate Director for Operating Reactor Oversight and Licensing, and the Associate Director for Risk Assessment and New Projects.

Brian Sheron will serve as the Associate Director for Engineering and Safety Systems. Reporting to Sheron will be Thomas Martin, who will become Director of the Division of Safety Systems; John Grobe, who will become Director of the Division of Component Integrity; and Michael Mayfield, who will be Director of the Division of Engineering.

Reporting to Martin will be Deputy Directors of the Division of Safety Systems, Jared Wermiel and John Hannon. Reporting to Grobe will be William Bateman, who will be Deputy Director of the Division of Component Integrity. Gene Imbro will serve as Deputy Director of the Division of Engineering, reporting to Michael Mayfield.

Bruce Boger will serve as Associate Director for Operating Reactor Oversight and Licensing. Reporting to Boger will be Frank Gillespie, who will become Director of the Division of License Renewal; Catherine Haney, who will be Director of the Division of Operating Reactor Licensing; and Michael Case, who will become Director of the Division of Inspection and Regional Support.

Reporting to Gillespie will be Pao-Tsin Kuo, who will be Deputy Director in the Division of License Renewal. Cornelius Holden and Edwin Hackett will serve as Deputy Directors in the Division of Operating Reactor Licensing, reporting to Haney. Stuart Richards and Patrick Hiland will become Deputy Directors in the Division of Inspection and Regional Support, reporting to Case.

Gary Holahan will serve as Associate Director for Risk Assessment and New Projects. Reporting to Holahan will be David Matthews, who will become Director of the Division of New Reactor Licensing; James Lyons, who will serve as Director of the Division of Risk Assessment; and Christopher Grimes, who will become Director of the Division of Policy and Rulemaking.

Reporting to Matthews will be Deputy Directors of the Division of New Reactor Licensing, William Beckner and Jose Calvo. Michael Tschiltz and Theodore Quay will be Deputy Directors in the Division of Risk Assessment, reporting to Lyons. Finally, Ho Nieh and Herbert Berkow will serve as Deputy Directors in the Division of Policy and Rulemaking, reporting to Grimes.

Details regarding the reorganization are described in SECY-05-0146, "Proposed Reorganization of the Office of Nuclear Reactor Regulation," dated August 12, 2005. The Commission approved the proposed reorganization in a Staff Requirements Memorandum dated August 25, 2005.

...ew organization goes into effect on October 30, 2005.

The SECY paper can be accessed at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2005/secy2005-0146/2005-0146scy.html

/RA/
J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachment: Reorganization Chart

NRC Yellow Announcements Index

OFFICE OF NUCLEAR REACTOR REGULATION

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DEPUTY DIRECTOR
R. William Borchardt

Program Management, Policy Development & Planning Staff
Cynthia A. Carpenter, Director
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- ORGANIZATIONAL EFFECTIVENESS BRANCH
- INFORMATION MANAGEMENT BRANCH

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  Jared S. Wermiel, Dep. Dir.
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  - PWR SYSTEMS BRANCH
  - NUCLEAR PERF & CODE REVIEW BRANCH
  - BALANCE-OF-PLANT BRANCH
  - SAFETY ISSUE RESOLUTION BRANCH
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  William H. Bateman, Dep. Dir.
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  - PIPING & NDE BRANCH
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  - ELECTRICAL ENGINEERING BRANCH
  - QUALITY AND VENDOR BRANCH A
  - QUALITY AND VENDOR BRANCH B

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  - LICENSE RENEWAL BRANCH C
  - ENVIRONMENTAL BRANCH A
  - ENVIRONMENTAL BRANCH B
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  - PLANT LICENSING BRANCH B
  - PLANT LICENSING BRANCH C
  - PLANT LICENSING BRANCH D
  - PLANT LICENSING BRANCH E
  - PLANT LICENSING BRANCH F
  - PLANT LICENSING BRANCH G
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  - HEALTH PHYSICS BRANCH

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  - PRA LICENSING BRANCH B
  - PRA OPERATIONAL SUPPORT & MAINT. BRANCH
  - FIRE PROTECTION BRANCH
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  - GENERIC COMMUN. & POWER UPRATE BRANCH
  - SPECIAL PROJECTS BRANCH
  - RES & TEST REACTORS BRANCH A
  - RES & TEST REACTORS BRANCH B

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Inside NRC

Volume 27 / Number 19 / September 19, 2005

NRC keeps open cross-cutting issues at 24 units, latest assessments show

In its mid-cycle assessment of plant performance, NRC identified 24 units where inspectors found substantive crosscutting issues in the areas of human performance (HP) and/or problem identification and resolution (PI&R).

Eighteen of the units also appeared on this list in March when NRC issued the results of its annual assessments (INRC, 21 March, 3): Salem-1 and -2 (PI&R), Cooper (PI&R), Callaway (HP), LaSalle-1 and -2 (HP), Fermi-2 (HP), Columbia (HP), Oyster Creek (PI&R), Kewaunee (PI&R), Indian Point-2 (PI&R), Hope Creek (PI&R), Palo Verde-1, -2, and -3 (HP and PI&R), Point Beach-1 and -2 (HP and PI&R), and Perry (PI&R and HP).

Six units were added to the mid-cycle cross-cutting issue list: Arkansas Nuclear One (ANO)-1 and -2 (PI&R), Byron-1 and -2 (HP), Duane Arnold (HP), and Watts Bar-1 (HP). Cross-cutting issues involving a safety-conscious work environment (SCWE) also remain open at Hope Creek and Salem.

Over the first six months of 2005, NRC closed cross-cutting issues at Dresden-2 and -3 (HP) and Diablo Canyon-1 and -2 (PI&R) after finding sufficient improvements at those plants.

In NRC's five-column action matrix, Salem-1 and -2, ANO-1 and -2, Cooper, Callaway, Byron-1 and -2, LaSalle-1 and -2, Fermi-2, and Duane Arnold are in Column 1 (the licensee response column). Watts Bar-1, Columbia, Oyster Creek, Kewaunee, Indian Point-2, and Hope Creek are in Column 2 (the regulatory response column). Palo Verde-1, -2, and -3 are in Column 3 (the degraded cornerstone column), and Point Beach-1 and -2 and Perry are in Column 4 (the multiple/degraded cornerstone column).

Both Hope Creek and Salem will continue to get more regulatory attention than is typical for Column 2 and Column 1 plants under an action matrix deviation memorandum approved by Executive Director for Operations Luis Reyes (INRC, 22 Aug., 13). The enhanced oversight will be used to monitor PSEG Nuclear's progress in addressing the PI&R and SCWE cross-cutting issues at both plants.

Getting on the cross-cutting issues list can be a precursor to declining performance. The Palo Verde units, which were added to the cross-cutting issues list in March, moved from NRC's Column 1 in March, following issuance of the agency's annual assessment, to Column 3 after this midcycle assessment.

In an Aug. 30 letter to plant operator Arizona Public Service, NRC Region IV Administrator Bruce Mallett said that during the past 12 months there were 18 green findings (of very low safety significance) connected with human performance. Mallett said that recent examples included the failure of personnel to follow procedures during the operation of fuel handling equipment and an inadequate surveillance procedure that resulted in an inadvertent safety injection actuation. Mallett also said that over the past 12 months there were 20 green findings with PI&R attributes, including such recent examples as the failure to implement corrective actions to preclude the failure of gasket retaining bolts on emergency core cooling system valves and the failure of personnel to suspend spent fuel movements upon discovery of a degraded condition.

Four units where cross-cutting issues were identified in NRC's annual assessment letters in March moved from Column 1 to Column 2 in this mid-cycle assessment: Columbia, Indian Point-2, Oyster Creek, and Kewaunee. Watts Bar-1 also moved from Column 1 in March to Column 2 in this assessment period, with NRC identifying a new cross-cutting issue in the HP area. NRC's Charles Casto, director of the division of reactor projects in Region II, told the Tennessee Valley Authority that the HP issue was of concern "because failures to follow plant procedures resulted in increased challenges to plant equipment from preventable transients and events."

Additional inspections at 22 units

In NRC's mid-cycle assessments of plant performance, 22 units are scheduled for additional NRC inspections, two fewer than the number of units NRC identified for additional inspections this spring. Ten units were added in this midcycle assessment to NRC's list of plants needing additional inspections, but 12 units that were on the increased inspection list in March were dropped from the current listing.
NRC's reactor assessment program collects information

from inspections and performance indicators (PIs) and uses this information to determine the agency's inspection effort at nuclear plants. Under the agency's action matrix, NRC will provide a baseline inspection for plants listed in the Column 1. For plants in Columns 2, 3, and 4, the regulatory oversight increases with an expanded number of inspections. The listing of plants in NRC's most recent action matrix is on NRC's Web site (http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/actionmatrix_summary.html).

Under NRC's system of color-coding inspection findings and PIs, a green inspection finding or PI indicates very low risk significance. White, yellow, and red inspection findings or PIs represent increasing degrees of safety significance.
New on the latest list of Column 2 plants were:
Columbia, Indian Point-2, Kewaunee, Peach Bottom-2, River Bend, Three Mile Island-1, and Watts Bar-1.
Remaining on the Column 2 list from earlier this year:
Cook-2, Hope Creek, Oconee-1, -2, and -3, Oyster Creek, Sequoyah-1, and Vermont Yankee.
Moving to Column 1 from their previous listing in March in Column 2 were: Cooper, Fermi-2, Brunswick-2, ANO-1, Calvert Cliffs-2, Cook-1, Robinson-2, Salem-1, San Onofre-2, Surry-1 and -2, and Waterford-3.
Perry and Point Beach-1 and -2 remained in NRC's Column 4.

In a letter to Nuclear Management Co. (NMC), which operates Point Beach, NRC Region III Administrator James Caldwell said the agency has noted that the number and significance of HP-related events have declined in recent quarters, "mainly attributed to the concerted effort NMC has applied toward the HP area." But Caldwell said NRC would keep the issue open in order to determine "the sustainability" of NMC's program. Caldwell went on to note that in the PI&R area, NRC was still finding instances where NMC's corrective action program "was not fully implemented in an effective manner."

In his letter to FirstEnergy Nuclear Operating Co. (Fenoc) about Perry's performance, Caldwell said the agency was "concerned with your progress" in addressing cross-cutting deficiencies in HP and PI&R. Caldwell said that "numerous" examples of those deficiencies continued after NRC sent Fenoc assessment letters in August 2004 and March 2005, indicating that efforts to fix those problems "have not been fully effective."

Fenoc has developed a Performance Improvement Initiative (PII) for Perry, but NRC found that the initiative had not been very effective. In response to the agency's request, Fenoc submitted a revised version of the PII Aug. 8. But in the Aug. 30 mid-cycle assessment letter, Caldwell asked Fenoc for information, within 30 days, on any actions Fenoc plans to take, in addition to ones described in the PII revision, to address the two cross-cutting issues.

In letters to PSEG Nuclear about Salem and Hope Creek, NRC Region I Administrator Samuel Collins used identical language as to why the SCWE cross-cutting issue would remain open at both plants. Collins said that "both PSEG and NRC reviews have shown that there exists a range of worker perceptions regarding the advisability of raising issues or challenging decisions in the current environment." On July 1, Fenoc's Davis-Besse returned to the reactor oversight process as a Column 2 plant, based on a white finding in the emergency preparedness cornerstone regarding siren testing.

Since 2002, when reactor vessel head degradation was found at Davis-Besse, NRC oversight of the unit had followed NRC Inspection Manual Chapter 0350, "Oversight of Operating Reactor Facilities in an Extended Shutdown as a Result of Significant Performance Problems."

—Michael Knapik, Washington

Inside NRC

Volume 27 / Number 20 / October 3, 2005

NRC stresses early detection for industry's RCS leakage guide

The NRC wants to ensure that reactor coolant system (RCS) leakage guidelines being developed by owners groups will detect and decrease leakage in a timely fashion, agency staff told industry representatives at a Sept. 29 meeting.

Limits for unidentified leakage specified in plant-specific technical specifications are typically 1 gallon per minute (gpm) for PWRs and 5 gpm for BWRs. Unidentified leakage at rates above those levels require plant shutdown, but the NRC staff also is concerned about leakage below those limits.

"Under certain circumstances, low levels of unidentified leakage can pose a safety concern because (1) it could imply that a component no longer has adequate structural integrity even when the amount of leakage is less than the [tech spec] limits and (2) the leakage may affect the integrity of the leaking or other component, e.g. through boric acid corrosion," Michael Mayfield, director of the division of engineering at NRC's Office of Nuclear Reactor Regulation (NRR), said in a July 7 letter to Alexander Marion of the Nuclear Energy Institute (NEI) requesting a meeting on the issue. Mayfield acknowledged in the letter that the industry is usually vigilant about investigating unidentified leaks that are less than their tech spec limits. However, he said the tech specs do not spell out what actions should be taken and how quickly they should be done.

"As a result, licensee actions in response to detecting unidentified leakage tend to be ad-hoc and vary depending on the magnitude of the leakage, the rate of increase in the leakage rate, and other factors (past inspections, need for other maintenance, etc.)," Mayfield said (INRC, 11 July, 18). At last week's meeting, representatives of the Westinghouse Owners Group (WOG) and BWR Owners Group (BWROG) discussed guidelines on leakage detection being developed for their members. WOG decided to last year to "standardize RCS leak rate calculation, standardize action levels and response to elevated leak rates, and improve other leak detection and monitoring techniques," Cal Walrath of WOG said in his presentation.

The PWR standard RCS leak rate guidelines now being developed will "establish a standard inventory balance calculation and technical bases; provide a consistent industry position on relevant issues; provide industry best practices for leak mitigation, detection, and action plans; and provide guidelines that can be used as a tool to help plants evaluate and improve their current program," Walrath said. The PWR guidelines will provide "standard action levels" for "unidentified and identified leakage as monitored by the RCS inventory balance" and as detected by monitoring of the containment atmosphere, sump, and air cooler condensate flow rate, Walrath said. The action levels will be evaluated against "actual RCS leakage events," and "standard guidance for conducting a RCS leakage investigation will be developed," he said.

A draft of the PWR leak rate guidelines will be circulated within WOG in November and a draft of the action levels and response guidelines in April 2006, Walrath said. Pilot plant implementation of the guidelines will be assessed in June, with final guidelines issued in September, he said. Seabrook and Ginna have been identified as likely pilot plants, and WOG hopes to have a few more, Joe Congdon of Westinghouse said at the meeting.

BWROG formed a generic RCS committee in 2004 to "develop best practices for monitoring and managing BWR RCS leakage...assess current BWR RCS leakage monitoring capability, [and] monitor industry RCS activities," Tom Veitch of BWROG said in his presentation. BWROG is developing a "best practices document" which is projected to be completed by the end of 2006, Veitch said.

Neither WOG nor BWROG has authority to make the RCS leakage guidelines mandatory for its members, Walrath and Veitch said in response to NRC staff questions. However, both said they anticipated the guidelines would be widely if not universally adopted.

The Institute of Nuclear Power Operations (INPO) conducts a "primary system integrity review visit program," and the possibility of integrating the RCS leakage guidelines into those visits will be discussed when the guidelines are finalized, NEI's Marion said.

Leakage detection

William Bateman, chief of the materials and chemical engineering branch at NRR, expressed concern that, under current tech specs, some PWRs "go a full operating cycle with 0.3 to 0.4 gpm unidentified leakage." This can result in "a lot of boron in containment. How do you know that boron isn't causing problems" such as corrosion of flange bolts or the reactor vessel, Bateman asked.

Periodic sampling for boric acid can determine whether leakage is primary or secondary coolant, as well as whether it has increased over a "baseline" leakage rate established at the previous outage, and walkdowns during outages can provide "good confidence" that corrosion is not occurring, Walrath said. There is "much more likelihood with [the new] guidelines of detecting small leaks over a period of time," Congdon added.

WOG Chairman Ted Schiffley noted that, in a rare mandatory action, WOG last year issued boric acid inspection guidelines that members must follow to address leakage once it has been detected. The guidelines include maintaining a database of identified leaks and tracking their status, as recommended by NRC's Davis-Besse Lessons Learned Task Force in 2002.

Stephen Monarque of NRR asked whether licensees have "localized detection" capability, and suggested that implementation of on-line leakage monitoring systems should be seriously considered. The state of the technical art for systems to detect RCS boundary leakage was reviewed in an Argonne National Laboratory report, Nureg/CR-6861, issued last December (INRC, 10 Jan., 8).

Operators' ability to detect leaks locally is currently "very, very limited," Veitch replied.

Such systems are "being tried out at a few plants like Davis-Besse," but "there's no assurance that's going to be the answer," he said. "Our best tool is visual inspection when the opportunity presents itself," Walrath said.

The NRC staff "is not considering at this point any regulatory action associated with" the RCS leakage guidelines, Bateman said. Staff is looking at revising and updating Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," the current version of which is dated May 1973, but there is no schedule yet for that project, Ken Karwoski of NRR said.—***Steven Dolley, Washington***

Inside NRC

Volume 27 / Number 20 / October 3, 2005

Diaz's multinational design initiative sparks interest at IAEA meeting

Regulators from around the world last week expressed support that ranged from enthusiastic to guarded for NRC Chairman Nils Diaz's proposal for a Multinational Design Approval Program (MDAP) to approve non-U.S. reactor designs under NRC rules and a multinational review process developed by 2009.

If the program goes forward, the world's nuclear safety authorities could ultimately converge on safety issues and criteria, practices, and implementation. It would permit more standardized designs and streamlined licensing, though each regulator would retain sovereign authority over siting, design certification, environmental assessment, and licensing of facilities within its borders.

Diaz had invited all regulators at the IAEA general conference to attend a special briefing on the subject Sept. 27 in Vienna. The NRC commissioners recently approved moving ahead to explore the MDAP concept (INRC, 19 Sept., 2). Diaz explained to officials in a packed briefing room that the NRC wants to take advantage of all the expertise available overseas on non-U.S. reactor designs so as not to delay design approval "unnecessarily." Time is short to review new designs if new reactors are to be built in the U.S. and elsewhere

beginning in 2010, he said.
At the same time, he said, design reviews and the designs themselves could be improved through multinational cooperation and the MDAP would eventually have a multitude of other applications, such as for converging engineering codes, quality assurance requirements, and confirmatory research.

As envisaged by Diaz, the MDAP would have three stages. In Stage I, foreign regulators would participate in NRC's review of foreign-designed reactors that are proposed for certification in the U.S. That could involve Framatome ANP's U.S. EPR and later encompass Atomic Energy of Canada Ltd.'s ACR-700/1000 or South Africa's Pebble Bed Modular Reactor.

The regulators' contributions would be remunerated as normal consultant work, and the product would be an NRC document that would be just shy of a formal final design approval (FDA).

The FDA and design certification would be solely NRC's responsibility. That is the only way NRC can legally include foreign regulators' experience in its reviews, NRC officials said. A legal review confirmed that an MDAP can be carried out under existing NRC regulations, Diaz said.

He said letters were on their way to Andre-Claude Lacoste, director general of French nuclear regulatory authority DGSNR, and Jukka Laaksonen, director of Finnish agency STUK, inviting them formally to participate in the EPR design review and to bring to NRC their experience in reviewing EPR's safety case in France and Finland. DGSNR issued an EPR design approval in September 2004. Early this year, STUK approved the world's first EPR for construction at Olkiluoto, making Finland a "special case" and eligible to participate from the outset in an NRC EPR review. DGSNR and STUK have been collaborating closely on EPR design review for more than two years.

The MDAP allows for regulators beyond those from the design-holding country to participate in Stage I, the NRC's own design approval process, but only if the vendor and the regulator from the design country agree. Diaz referred to this as "Stage I, Part B," saying it would be beneficial to all parties: "We're going to put everyone to work." It can be done through bilateral agreements and does not require the specific approval of Congress, he said. Diaz indicated that foreign experts participating in the program could be paid. Responding to concerns voiced by Japan about protecting proprietary information, Diaz said there were ways to do that and that it shouldn't be a problem.

China appears to be a prime candidate to participate in an EPR design review, since its nuclear industry is considering a formal bid by Framatome to supply four EPRs. A Chinese official said after last week's meeting said that the country was interested but that Beijing needed more information about the conditions of the program, including how much it might cost and how much manpower it would take, before it could make a decision. However, NRC officials later expressed confidence that China would sign on as a member of the MDAP "core group."

Linda Keen, chairman of the Canadian Nuclear Safety Commission, said Canada was "certainly interested" in being part of the EPR design review, in addition to any certification review of the Canadian ACR design.

Core group

If foreign regulators decide to participate, things could move very quickly. Diaz wants to constitute a "core group" of seven or eight regulators interested in the proposal by the end of this month and to set up a working group to establish "key issues and the program" for Stage I, in which foreign regulators would participate in NRC design reviews of foreign reactor designs.

By January, Diaz hopes to assemble a group to work out the program's Stage II, which would aim for a multinational approval process, based not on NRC regulations but on agreed international safety standards. In this stage, foreign regulatory bodies would take the lead in multinational reviews of their national designs. Canada, for example, would lead the review of the ACR, or South Africa for the PBMR. The OECD Nuclear Energy Agency (NEA) would serve as secretariat for the program in that stage. A year later, in January 2007, the program would begin "receiving lessons learned" from Stage II, Diaz said.

"A significant number of decisions regarding growth of nuclear power over the next 25 years are going to be made" in 2006-2010, Diaz said. "This tool should be available by the 2009 timeframe."

In Stage III, world regulators would review fourth-generation reactor, or so-called Gen IV, designs, under a common methodology with common requirements. Diaz said that tools were needed in time to be "compatible with the Gen IV schedule," which foresees advanced reactor designs being commercially deployed by 2030.

In fact, Diaz said the idea of a multinational review was prompted by pressure from the U.S. DOE-led Generation IV International Forum (GIF) for a clear indication of the regulatory requirements for the next generation of reactors.

Since Diaz floated the idea of a Gen IV regulators' club two years ago, he has come to realize that the concept also should be applied to near-term reactor designs. Diaz launched the MDAP idea formally a year ago at an IAEA meeting in Beijing (INRC, 18 Oct. '04, 7).

Diaz emphasized the proposed program is not a "harmonization" of safety rules and "not a competition among regulators." Rather, he said, it is a way to achieve more efficient reviews. For example, he said NRC had spent "seven years and hundreds of thousands of man-hours to review the first set of design certifications." Three designs have been certified by NRC: Combustion Engineering's System 80+, General Electric's ABWR, and Westinghouse's AP600.

In stages

Diaz said that in "three or four years" NRC "can and should amend our rulemakings to allow design approval" to be done under a multinational program and criteria.

In Stage I of an MDAP, the NRC commission would ask the staff to analyze NRC's licensing regulations (10 CFR Part 52) for "how they comply" with the IAEA nuclear safety standards, Diaz said. In Stage II, the IAEA standards would be used as one basis for more detailed specifications. The output of Stage II would be a "safety case bank" whose commodity would be a design approval of each proposed reactor, Diaz said.

It wasn't immediately clear how far the multinational approval would go, with one official saying it should cover only the nuclear steam supply system and others saying that containment—a potentially contentious issue between Europe and the U.S.—was too important to plant safety to be left out of the process. Work that goes into the certification review of a reactor design represents about 80% of the total regulatory effort aimed at deploying a new nuclear unit, according to Diaz.

Involvement
There was much discussion at last week's meeting about whether the program would be multinational—the term preferred by Diaz—or international. Ken Brockman, head of IAEA's nuclear installations safety division, said it was "hard to see where this would go" after Stage I if it was not an international process.

Luis Echavarri, director general of the NEA, said that "multinational is the right word" because "not all countries are going to be interested." He also emphasized the NEA's experience in assuring secretariats for multinational programs, including GIF, and its "flexibility" in being able to include non-member countries.

NRC officials said after the meeting that the makeup of the "core group" was not yet clear. But alongside the U.S., the group is likely to comprise the Finland, South Africa, Japan, the U.K., South Korea, and Canada.

At last week's meeting, Laaksonen greeted the proposal with enthusiasm, saying the project should get started as soon as NRC receives the application for certification from Framatome. He said STUK "can provide independent analysis" of the design and brings, among other things, "two years' experience in (overseeing) manufacturing of main components" for Olkiluoto-3.

Neither Lacoste nor anyone else from DGSNR was at the Vienna meeting, making France the only major nuclear power country not to attend. DGSNR's absence was interpreted by more than one attendee as little short of a boycott. Lacoste had told Inside NRC on Sept. 27 that he had received the invitation only two weeks earlier and could not cancel a previous engagement, and that his deputy and department head involved with the issue both also had prior commitments.

Lacoste said in a telephone interview Sept. 28 that he had not yet received any formal invitation to participate in the EPR review, though he has expressed interest in it in the past. He said he did not yet know in detail how the process would work but "a priori," DGSNR would bring its expertise to the NRC review.

Diaz, Lacoste, Keen, and other members of the International Nuclear Regulators Association (INRA) were to meet in Munich Sept. 29-30 under the chairmanship of Wolfgang Renneberg, head of the reactor safety department at the German Ministry of Environmental Protection &

Nuclear Safety.

Russia's nuclear regulatory chief, Andrey Malyshev, was also not at the Vienna meeting, but he sent four people in his place. The previous day, he told Inside NRC that he preferred a "multinational methodology" for design review "within the framework of the IAEA." Malyshev said Russian reactor designs had been "certified by the IAEA" in reviews of Russian reactors being built in China and India and that one Russian VVER design was "under certification" by European utilities.

A Russian regulatory agency official who participated in the Vienna meeting said that there was no official position yet on whether Russia would join the MDAP. But, she said, based on last week's discussions, "it seems to me Stage I is no place for us, because the U.S. is not going to build Russian reactors, and we are not going to build EPR in Russia. So it's not necessary for us to participate in Stage I, and maybe not even possible."

But if and when Stage II comes around, she said, "when common approaches should be developed under the banner of the NEA, for sure we will participate."

Regulators from some other large countries said that they would like to join the initial stage to participate in NRC's review of EPR and perhaps other foreign designs. But many others said they were standing ready for Stage II, when the process turned to developing international safety rules and criteria.

In an interview following the meeting, Laaksonen said he had no problem with participating in the NRC review of EPR as a contractor, because "it's the only way we can participate" and it's in the interest of both parties.

A question not raised during the meeting was whether the existing U.S. reactor designs that have either been certified by NRC or are in the pipeline, notably Westinghouse's AP1000 and General Electric's ESBWR, could or should be subjected to multinational reviews.

Laaksonen said that "U.S. NRC certification is not a multinational certification" and that no other regulator besides NRC had been involved in an in-depth safety review of AP1000, with common safety criteria. He said STUK has "never agreed to the AP1000 design basis," nor has it reviewed GE's ABWR or the ESBWR. STUK did do a prereview of an early version of the ABWR after GE proposed it for construction at Olkiluoto. But Laaksonen said that "was not a complete design" and that STUK never reviewed the most recent version of the design because TVO hadn't picked it for construction.

Laaksonen said there were no grounds to expect that an NRC review of EPR would lead to any safety issues that would call into question the design of Olkiluoto-3, affirming, "our plant is so far superior to anything in operation in the U.S."—***Ann MacLachlan, Vienna***

October 04, 2005

Indian Point reactor shut

By Greg Bruno
Times Herald-Record
gbruno@th-record.com

Buchanan – A control rod failure at Indian Point 3 has prompted Entergy Nuclear Northeast to temporarily shut down one of the plant's two nuclear reactors, the latest in a series of mishaps to draw criticism of the plant.

Jim Steets, an Entergy spokesman, said plant operators will spend the week repairing and inspecting 53 control rods, which are used to slow the chain reaction in the reactor core.

Workers began shutting down the reactor Friday after one of the rods malfunctioned. The failure was the result of a loss of power to the mechanical arm that lowers the rods into place.

"The plant was shut down to fix a short between the drive mechanism and the control rods," Steets said.

"You need electric power to hold the rods in place. If you loose power, they are automatically inserted."

The suspension of power production, which removes about 1,000 megawatts from the state's power grid, is expected to last through the week. A spokesman for the New York Independent System Operator said the disruption would not impact supplies or rates statewide.

Friday's malfunction was one in a string of recent problems to besiege the 29-year-old reactor, which sits on the banks of the Hudson River in Westchester County, 35 miles north of Midtown Manhattan.

It was also the latest chance for plant critics to publicly assail Entergy's commitment to safety.

In recent months, Indian Point's emergency warning system has failed repeated tests, prompting lawmakers in Washington to call for immediate Nuclear Regulatory Agency intervention.

And last month, Entergy announced the discovery of a small leak in one of the plant's spent-fuel pools. The probe of the leak is continuing.

"It's no longer a matter of if or when Indian Point will malfunction; it already has, repeatedly," said Alex Matthiessen, director of the environmental group Riverkeeper.

"Enough is enough."

St. Petersburg Times ONLINE TAMPA BAY

Second nuclear plant in the works

Progress Florida's CEO Bill Habermeyer says nuclear power has a big advantage: It emits no carbon.

By LOUIS HAU, Times Staff Writer
Published October 4, 2005

If plans fly, Progress Energy expects to select a design and site, possibly in rural Central Florida, by the end of this year. The plant could be up and running by 2015.

Spurred by customer growth, rising conventional fuel costs and a pronuclear stance by the Bush administration, Progress Energy Florida may build a second nuclear power plant in Florida, with rural counties in Central Florida providing some of the most attractive options for a site.

If the St. Petersburg utility proceeds with such plans, the new plant would become Florida's first new nuclear generation project since 1983, when Florida Power & Light opened a second reactor at its St. Lucie nuclear complex near Fort Pierce. In 1977, Progress Energy, then known as Florida Power Corp., began operating its first and only nuclear power plant at its Crystal River complex in Citrus County.

By the end of this year, Progress' corporate parent, Progress Energy Inc. of Raleigh, N.C., expects to select a potential site and design for a nuclear plant to meet the growing electricity demands of its expanding customer base. Altogether, Progress operates four nuclear facilities in the Carolinas and Florida.

Central Florida is high on Progress Energy's list of potential sites. A location in Polk, Seminole, Osceola or Highlands counties would put a nuclear power plant closer to major transmission lines in the state. And while Pinellas and Pasco counties account for the greatest portion of electricity demand in the company's Florida service territory, Central Florida is experiencing the greatest customer growth, a key consideration when siting a power plant.

Choosing a site and vendor for a nuclear plant is among the first formal steps of a lengthy license application process that could take years. For now, a nuclear plant appears to be the most likely expansion option for Progress, although company officials do not rule out the possibility that the company will eventually decide to build a plant powered by coal instead.

In an interview Monday, Progress Florida president and chief executive Bill Habermeyer said that nuclear power's lack of carbon emissions and its ability to potentially reduce American dependency on foreign energy sources give it significant advantages.

"When you look at the choices ahead, I think nuclear provides a better alternative," he said.

Habermeyer added that the Crystal River site, while attractive, also has some disadvantages. The complex already includes four large coal-fired generating units producing more than 3,000 megawatts of electricity. Adding a second nuclear reactor to the site would mean that "you're putting a lot of generation at one location," he said.

But building a nuclear power plant in a new location - and bringing with it the prospect of storing highly radioactive nuclear waste on site - could pose a formidable political challenge. Another challenge: Nuclear power plants require a nearby water source for cooling, an additional hurdle away from Florida's coastline.

Habermeyer acknowledged that building on a new site will inevitably trigger some opposition. But he added that he hopes local residents and public officials will recognize what he said are the environmental and economic advantages of nuclear power as well.

Soaring gas, oil and coal prices have sent electricity prices soaring, further reducing consumers' longstanding resistance to nuclear power.

"Ultimately, you have to have a generating source that can provide sufficient electricity to power this country," he said.

Construction of a plant could begin in five years, with the plant becoming operational as early as 2015.

Progress' nuclear ambitions come at a pivotal time for the U.S. nuclear power industry, which has long operated under a pall of safety concerns since the 1979 partial meltdown at Three Mile Island nuclear power plant near Harrisburg, Pa. In fact, the U.S. Nuclear Regulatory Commission has not issued a license for a new nuclear reactor since then.

But the prospects for U.S. nuclear power generation have grown rosier under the Bush administration. In a bid to encourage applications to build and operate nuclear plants, the Energy Department offered in some cases to pay up to half the cost of applying for the required license, which can run into hundreds of millions of dollars.

In addition, President Bush signed long-delayed energy legislation in August that provides production tax credits, loan guarantees and risk protections for companies building nuclear reactors.

Progress executives have unusually deep roots in nuclear power. Habermeyer, Progress Energy Inc. chairman and chief executive Bob McGehee and retired chairman and chief executive Bill Cavanaugh are all veterans of the U.S. Navy's nuclear submarine program.

Still, nuclear is not the expansion choice of every power company.

Florida Power & Light of Juno Beach, the state's larger nuclear power plant operator, recently decided against a new nuclear generator. Instead, it has proposed building a coal plant in St. Lucie County.

--Louis Hau can be reached at hau@sptimes.com or 813 226-3404.

Nuke unit shut down a 3rd time

Seal leak forces Palo Verde repair

Kèn Alltucker
The Arizona Republic
Oct. 4, 2005 12:00 AM

A reactor at the Palo Verde nuclear power plant has been shut down for the third time this year due to a leaking oil seal.

Arizona Public Service Co. described this week's repair of the oil seal in Unit 3's coolant pump as a planned move to take care of the persistent problem, which also forced the reactor's shutdown in May and July. APS expects to complete the repairs this week and start up Unit 3 next weekend.

APS officials acknowledge that some oil seals are wearing more quickly than expected, so the utility has launched a "root-cause" investigation to get a better idea about why.

"For some reason, we're getting less life out of these seals than others," said James Levine, APS' executive vice president of generation.

"We have some time here to continue with our root cause (investigation) and determine if we have to do something different."

The utility also will closely inspect the oil seals in Palo Verde's other reactors, Units 1 and 2.

Levine said there was some evidence that at least one Unit 2 oil seal would need to be replaced soon, although no timeline for its replacement has been established.

Crews also likely will replace some seals during Unit 1's refueling outage that will begin this weekend. Unit 1's outage is expected to last 75 to 80 days as crews tackle major jobs, including replacing steam generators, low-pressure turbines and computer systems.

Levine said APS can't compare notes with other nuclear power plant operators because Palo Verde is the only plant that uses the German-made coolant pumps that are the focus of the examination.

There are two oil seals for each of the four coolant pumps in each reactor. Some nuclear plants in South Korea use similar parts, so APS will seek to find out whether similar problems have been found there, Levine said.

Reactor shutdowns at Palo Verde this year have been costly for APS and the plant's other owners. Palo Verde is the nation's largest nuclear power plant - a form of energy that is cheaper to generate than other sources of electricity such as coal, oil or natural gas.

APS told the Arizona Corporation Commission last month that it cost more than $30 million to replace energy lost due to unplanned outages at Palo Verde from April through August. Salt River Project, the second-largest Palo Verde owner, estimates the outages from April through August cost it $19.5 million.

APS revealed the Palo Verde outage costs to the Arizona Corporation Commission as part of its fuel-cost "adjuster" case that seeks to pass along higher fuel costs to ratepayers. If the utility gets its way, Arizona ratepayers could see a temporary 2.1

percent hike in electricity bills.

Also this week, the U.S. Nuclear Regulatory Commission will dispatch a special investigative team to Palo Verde from the agency's Arlington, Texas, regional headquarters.

The team will review the plant's equipment and safety systems. The special investigation stems from the plant's "yellow" safety violation that resulted in a $50,000 fine levied in April after inspectors found air in a pipe that could have disrupted the plant's emergency cooling system.

NRC spokesman Victor Dricks said the special team would gather information at Palo Verde during interviews and inspections this week. The team will return for a follow-up inspection this fall before issuing a final report in December.

Reach the reporter at (602) 444-8285 or ken.alltucker@arizonarepublic.com.

(3)

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
October 6, 2005

1

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**
- **Four (4) Open Items**
  - 1 Scoping and Screening: OI 2.4-3
  - 1 Unit 1 Periodic Inspection: OI 3.0-3 LP
  - 1 Time-limited aging analysis: OI 4.7.7
  - 1 RHRSW Piping: Inspection Finding
- **Two (2) Confirmatory Items**
- **Four (4) License Conditions; Three Standard and one specific to BFN**
- **Fourth Condition requires completion of Unit 1 CLB differences (13 Items) described in LRA Appendix F**

2

Section 2.4: Scoping and Screening of Containments, Structures and Supports

Open Item 2.4-3 Drywell Shell Corrosion

- Potential Corrosion of the inaccessible portion of the Drywell affected by the leakage from the refueling cavity seal
- The staff proposed two options:

(1) Bring the refueling cavity seal in the scope of LR or
(2) Periodically monitor the potential degradation of the inaccessible side of the dry well

3

Section 3.7: AMR of Unit 1 Systems in Lay Up

Open Item 3.0-3 LP

- Unit 1 Periodic Inspection Program
  - BFN submitted Unit 1 Periodic Inspection Program
  - Staff in reviewing the Program element needed additional confirmation.
  - Section 3.0 of the final SER will include the staff evaluation of this program

4

Section 4.7.7: Stress Relaxation Core Plate Hold-Down Bolts

- TLAA evaluated for loss of pre-load in accordance with 10 CFR 54.21 (c)(1)(ii)
- Expected loss of preload of 20% which bounds the original BWRVIP-25 value
- In a plant specific analysis
  - Core plate hold-down bolts will maintain sufficient preload to prevent sliding of core plate
  - Hold-down bolts meet ASME Section III, Class 1, Level D service limits at the end of PEO

5

Section 4.7.7: Stress Relaxation Core Plate Hold-Down Bolts

- **Open Item 4.7.7**
  - The staff reviewed the method of analysis based on GE's generic stress relaxation data on irradiated stainless steel materials
  - The staff requested additional information to address
    - Horizontal and vertical loads for all operating conditions during PEO
    - Sliding of core plate from core plate rim during the PEO
    - Axial and bending stresses of bolts

6

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

Caudle Julian, Senior Project Manger

DRP, Region II

7

Browns Ferry License Renewal Inspection

- AMP inspection conducted November 29 - December 17, 2004
- Inspection concluded that existing programs to be credited as aging management programs for license renewal are generally functioning well.
- Inspectors observed that the applicant had not yet begun the implementation process for new and enhanced AMPs
- AMP procedures have yet to be defined and composed

8

Browns Ferry License Renewal Inspection

- For existing programs, the identification and selection of which particular existing procedures constitute the AMP had yet to be done
- Region II concluded that NRC will perform another inspection when the applicant has progressed further with AMP implementation.
- In walking down plant systems and examining plant equipment the inspectors found no significant adverse conditions and it appears plant equipment was being maintained adequately.

Browns Ferry License Renewal Inspection

Second (optional) AMP Inspection

•Conducted September 19 - 23, 2005

•Reviewed sample of 40 AMP Implementation Packages containing proposed procedures

- Packages contained some errors and were not meticulously reviewed
- Applicant initiated PER for corrective action

Browns Ferry License Renewal Inspection

Second (optional) AMP Inspection (cont.)

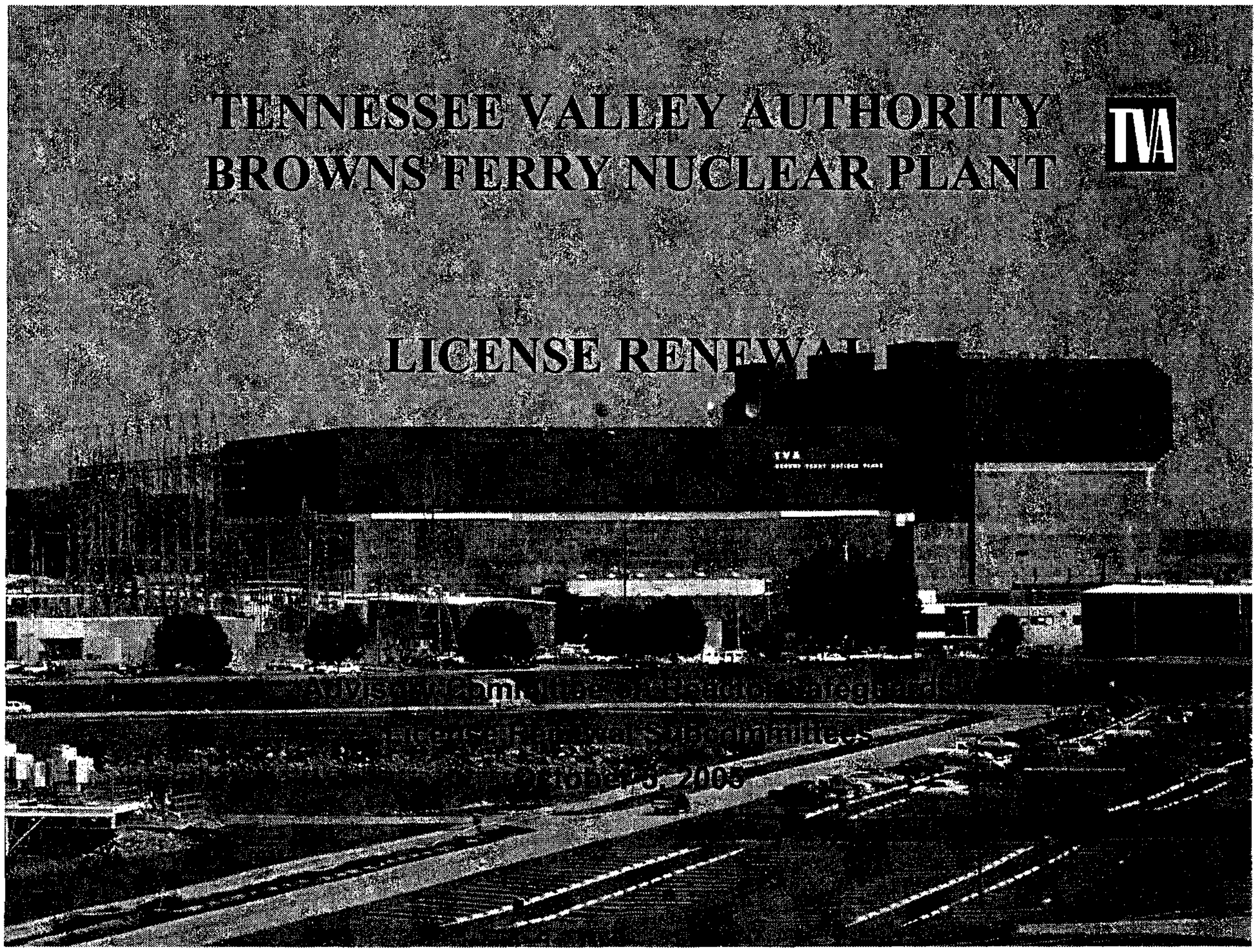
- Reviewed plans for tracking future actions using TROI system
  - Not initially linked to Implementation Packages but quickly corrected
  - Inspection sample commitments were included
  - Much duplication and varying format resulting in confusing document
- Applicant decided to track future actions using PER process
- Region II will follow up on these issues during a future inspection

11

Conclusion

- Region II concluded that NRC will perform another inspection when the applicant has progressed further with AMP implementation.
- In walking down plant systems and examining plant equipment the inspectors found no significant adverse conditions and it appears plant equipment was being maintained adequately.

12



Agenda

- Description of Browns Ferry
- BFN License Renewal Application
  - Scoping
  - Time-Limited Aging Analysis
  - Aging Management Programs
- Unit 1 Layup
- Unit 1 Operating Experience
- License Renewal Commitments
- Open Items

Description of Browns Ferry

- All Three BFN Units are General Electric BWR 4 Reactors with Mark I Containments
- Approximate Years of Operation
  - Unit 1 – 10
  - Unit 2 – 23
  - Unit 3 – 18
- BFN Units 2 and 3 in Operation Since Recovery in 1991 and 1995, Respectively
- Unit 1 in Recovery Outage with Restart Scheduled for May 2007
  - Unit 1 will be operationally identical to Units 2 and 3
- NRC Performance Indicators Green

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 Differences Between Unit 1 and Units 2 and 3
  - These differences will be eliminated prior to Unit 1 restart (May 2007)
- Requests for Additional Information ~230 (13 are Environmental, Remainder are Safety Evaluation)

Scoping

- Scoping Basis
  - Updated Final Safety Analysis Report
  - Safe Shutdown Analysis calculation
  - Maintenance Rule documentation
  - Controlled Plant Component Database
  - Licensing Basis and Design Basis documents
- Specific Scoping for Regulated Events
  - Fire Protection
  - Environmental Qualification
  - Anticipated Transients Without Scram
  - Station Blackout
- 77 Mechanical / Electrical Systems in Scope

Time-Limited Aging Analysis

- Neutron Embrittlement of the Reactor Vessel and Internals
- Metal Fatigue
- Environmental Qualification of Electrical Equipment
- Primary Containment Fatigue
- Plant Specific Time-Limited Aging Analyses
  - Reactor Building Crane Load Cycles
  - Radiation Degradation of Drywell Expansion Gap Foam
  - Irradiation Assisted Stress Corrosion Cracking (IASCC) of Reactor Vessel Internals
  - Stress Relaxation of the Core Plate Hold-Down Bolts
  - Emergency Equipment Cooling Water Weld Flaw Evaluation

Reactor Vessel Time-Limited Aging Analysis

- Neutron Embrittlement of the Reactor Vessel and Internals
  - Unit 1: Conservatively evaluated using 54 Effective Full Power Years at Extended Power Uprate conditions
    - Peak fluence of limiting weld 1.95×10^{18} n/cm^2
  - Units 2 and 3: Conservatively evaluated using 52 Effective Full Power Years at Extended Power Uprate conditions
    - Peak fluence of limiting weld 2.3×10^{18} n/cm^2

License Renewal Aging Management Programs

- 39 Aging Management Programs Total
  - 38 are common to Units 1, 2, and 3
  - 1 is for only Unit 1 (i.e., Unit 1 Periodic Inspection Program)
- 11 Existing Aging Management Programs Requiring No Enhancement
- 11 Existing Aging Management Programs Revised Only to Include Unit 1
- 11 Existing Aging Management Programs Require Enhancement for all Units
- 6 New Aging Management Programs

License Renewal
Aging Management Programs

- Existing Aging Management Programs Requiring No Enhancement
  - 10 CFR 50 Appendix J Program
  - Above ground Carbon Steel Tanks Program
  - ASME Section XI Inservice Inspection Subsections IWB, IWC, and IWD Program
  - ASME Section XI Subsection IWE Program
  - Bolting Integrity Program
  - BWR Control Rod Drive Return Line Nozzle Program
  - Diesel Starting Air Program
  - Fuel Oil Chemistry Program
  - Inspection of Overhead Heavy Load and Light Load Handling Systems Program
  - Reactor Head Closure Studs Program
  - Systems Monitoring Program

License Renewal
Aging Management Programs

- Existing Aging Management Programs Requiring Revision Only to Incorporate Unit 1
  - BWR Feedwater Nozzle Program
  - BWR Penetrations Program
  - BWR Reactor Water Cleanup System Program
  - BWR Stress Corrosion Cracking Program
  - BWR Vessel Inside Diameter Attachment Welds Program
  - Chemistry Control Program
  - Closed-Cycle Cooling Water System Program
  - Environmental Qualification Program
  - Fire Protection Program
  - Flow-Accelerated Corrosion Program
  - Open-Cycle Cooling Water System Program

License Renewal Aging Management Programs

- Existing Aging Management Programs Requiring Enhancement (All Units)
  - ASME Section XI Subsection IWF Program
  - Buried Piping and Tanks Inspection Program
  - BWR Vessel Internals Program (includes steam dryers)
  - Compressed Air Monitoring Program
  - Electrical cables not subject to 10 CFR 50.49 Environmental Qualification requirements used in Instrumentation Circuits Program
  - Fatigue Monitoring Program
  - Fire Water System Program
  - Inspection of Water-Control Structures Program
  - Masonry Wall Program
  - Reactor Vessel Surveillance Program
  - Structures Monitoring Program

License Renewal Aging Management Programs

- New Aging Management Programs (for all Three Units)
  - Accessible Non-Environmental Qualification Cables and Connections Inspection Program
  - Bus Inspection Program
  - Inaccessible medium voltage cables not subject to 10 CFR 50.49 Environmental Qualification Requirements Program
  - One-Time Inspection Program
  - Selective Leaching of Materials Program
- New Aging Management Program (Unit 1 Only)
  - Unit 1 Periodic Inspection Program

One-Time Inspection Program

- Applies to Units 1, 2, and 3
- Verifies the Effectiveness of Aging Management Programs by Confirming that Unacceptable Degradation is not Occurring
- Where No Aging Management Program is Identified, the Inspections Confirm either:
  - Aging Effects are not Occurring, or
  - Aging Effects are Occurring at a Rate that does not Affect the Intended Function
- To be Completed Prior to the Period of Extended Operation
- Examples of inspection items:
  - Reactor coolant pressure boundary piping, valves, fittings less than four inches
  - Bottom thickness of above ground tanks
  - Submerged concrete and component supports
  - Ventilation ducts

Unit 1 Periodic Inspection Program

- Periodic Inspections will be Performed after Unit 1 is Returned to Operation to Verify No Additional Aging Effects are Occurring
- The Periodic Inspection Sample Locations will be a Subset of Non-Replaced Piping Locations (takes credit for restart inspections)
- First Round of Periodic Inspections will be Completed Prior to Period of Extended Operation and after Several Years of Unit 1 Operation
- An Inspection will be Performed during the Period of Extended Operation
- Subsequent Inspection Frequency will be Determined Based on Inspection Results

Unit 1 Layup Program

- Criteria
  - EPRI NP-5106, "Sourcebook for Plant Layup and Equipment Preservation", Revisions 0 (1987) and 1 (1992)
- Types of Layup
  - Dry
  - Wet
- Lessons Learned from Unit 3 Layup and Subsequent Restart Applied to Unit 1

Unit 1 Layup Program

- Examples of Systems in Layup
  - Dry

Core Spray	Condensate
Reactor Core Isolation Cooling	Feedwater
High Pressure Coolant Injection	Off Gas
Residual Heat Removal	Main Steam

  - Wet

Reactor Vessel	Control Rod Drive
Recirculation	

- Results Met or Exceeded EPRI Guidelines
- Performed Visual, Surface, Ultrasonic, and Remote Inspections to Assess Unit 1 Condition
- No credit was taken for the lay-up program in determining the acceptability of structures, systems, or components for Unit 1 restart

Unit 1 Operating Experience

- 10 CFR 54.17(c):

  An application for a renewed license may not be submitted to the Commission earlier than 20 years before the expiration of the operating license currently in effect.

- Unit 1 Met this Requirement

- Unit 2 and Unit 3 Operating Experience is Applicable to Unit 1

Unit 1 Operating Experience

- Unit 1 has 10 Years of Operation
- Unit 3 Shutdown for 10 Years
  - Extensive layup experience with Unit 3 directly applicable to Unit 1
  - No layup induced aging effects during 10 years of ensuing operation
- Layup Experience from Unit 3 Incorporated into Unit 1 Recovery
  - RHR service water piping
  - Small bore piping
- Unit 1's Licensing Basis will be the same as that of Unit 2 and Unit 3 at Restart (Appendix F)
- Unit 1's Design, Configuration, Operating Procedures, Technical Specifications, and Updated Final Safety Analysis Report Identical to Unit 2 and Unit 3
- Internal and External Plant Operating Experience Incorporated into BFN Corrective Action Program

License Renewal Commitments

- Commitments made Through Application and Requests for Additional Information
- Tracked with Onsite Commitment Tracking System and Corrective Action Program
- ~ 114 Commitments made to Date

Open Items

- Core Plate Hold-Down Bolts
- Drywell Shell Corrosion
- Inspection of RHRSW Piping

Summary

- Three Unit Application at Current Licensed Thermal Power
- Prepared using Generic Aging Lessons Learned Report (Rev. 0, 2001)
- Appendix F ensures Unit 1 Differences are resolved prior to Restart of the Unit
- Unit 2 and Unit 3 Operating Experience is Applicable to Unit 1

4

United States
Nuclear Regulatory Commission

GENERIC ISSUE 80
PIPE BREAK EFFECTS ON CRD HYDRAULIC LINES IN BWRs

Presented by
Abdul Sheikh
Harold VanderMolen
Office of Nuclear Regulatory Research

October 6, 2005

Safety Significance

- Initiating event is a large break LOCA
- If pipe break is near CRD hydraulic lines, whipping pipe may crimp or kink some withdraw lines, preventing a cluster of rods from scramming
- ECCS refills reactor vessel with cold water
- Possible reactivity excursion
- Additional post-LOCA heat source

Findings

- Core damage frequency well below thresholds
- Public risk well below thresholds

History of GI-80

- Raised by ACRS in 1983
- Prioritized as low priority in 1984, based on pipe layout geometry
- Closed out in 1995
- Reopened in 1998, based on discovery of new piping configurations
- NUREG/CR-6395 in 1999 identified breaks in RCS and RHR piping that may damage CRD piping in BWR plants

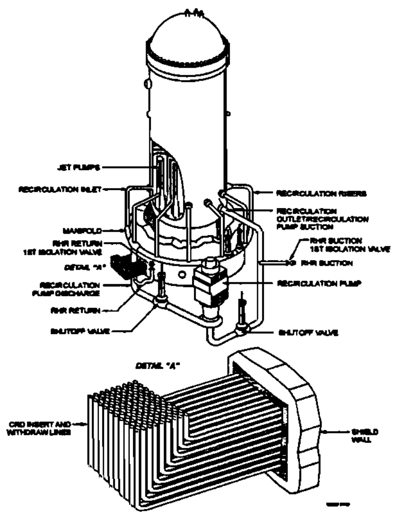
Technical Assessment Objective

- Perform detailed analysis of RHR and RCS break interactions with CRD piping
- Determine if complete closure of CRD piping on impact with RHR and RCS piping is possible
- Determine revised CDF based on a conservative estimate of the probability of crimping of CRD piping
- Use revised CDF to determine appropriate rating for GI-80 in accordance with MD 6.4

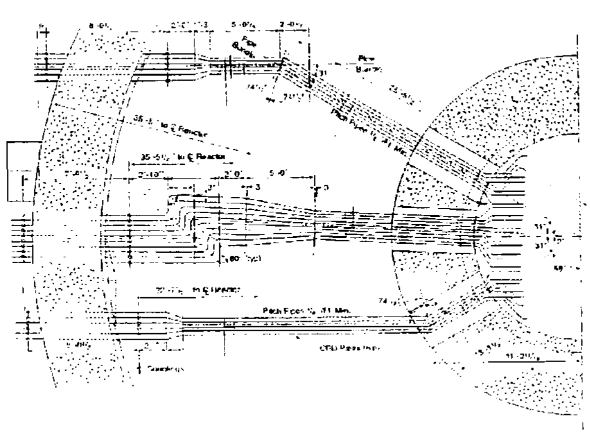
BWR Mark I and Mark II RCS Piping and CRD Arrangements

- The RHR and RCS piping layout and configuration for GE Mark I and Mark II is essentially the same
- The CRD piping layout for Mark I GE2 is different from Mark I/II GE3/GE4/GE5
- Mark I GE 2 has three sets of CRD bundles
- Mark I/II GE3/GE4/GE5 has four sets of CRD bundles

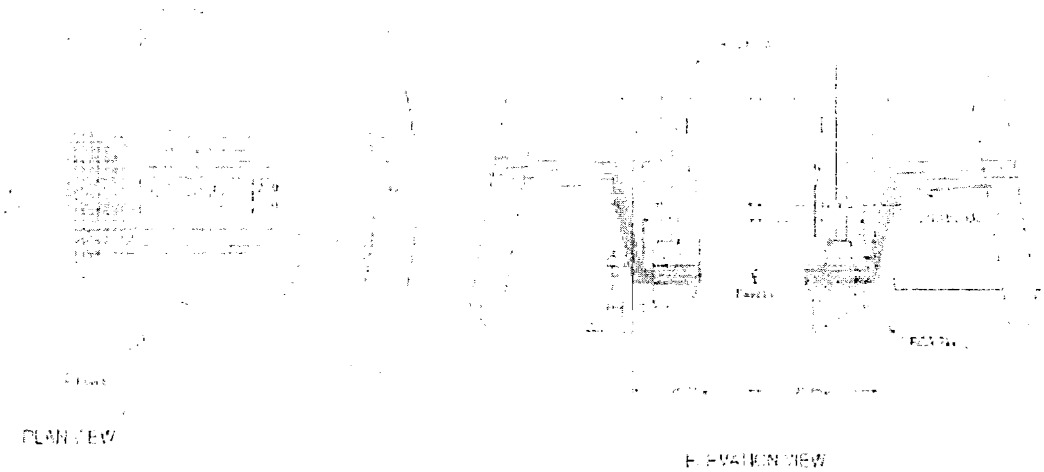
GE Mark I and II RCS Piping Layout



Mark I GE2 CRD Piping Layout



Mark I/II GE3/GE4/GE5 CRD Piping Layout



Event Quantification Process

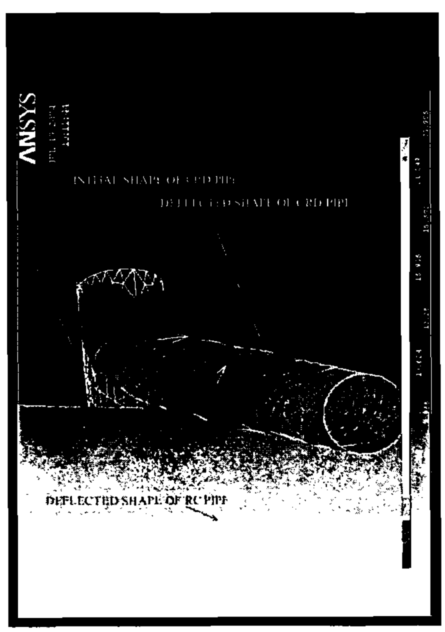
CDF for RHR/RCS piping impact on CRD piping is a function of the following factors

- Pipe rupture initiating event (IE)
- Fraction of piping considered in IE that is from RHR or RCS system (PIPETYPE)
- Fraction of RHR or RCS system piping that can impact CRD piping (TYPEFRAC)
- Probability of pipe whip or jet impingement that can cause CRD system failure (RUPTPROB)

Probability of CRD System Failure (RUPTPROB) for Mark I–GE2 Containment

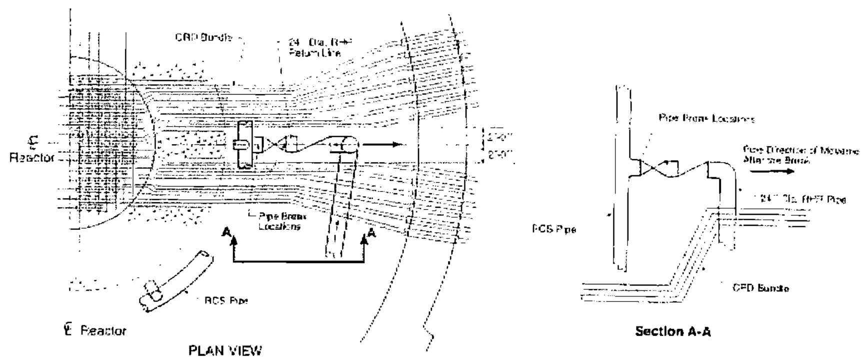
- RHR piping cannot impact CRD piping
- CRD piping bundle located 18 feet above the postulated break location of RCS piping
- 25 inch clearance between RCS and CRD piping
- RCS piping will fail (plastic hinge will form) before it can impact the CRD bundle
- Finite element analysis performed for a hypothetical impact to CRD piping
- CRD piping flexible and will bend without significant crushing or crimping before rupture
- Analysis results consistent with the behavior in the test results documented in NUREG/CR-3231
- RUPTPROB can be conservatively taken as 0.1 for CDF calculations

Deflected Shape of CRD Pipe



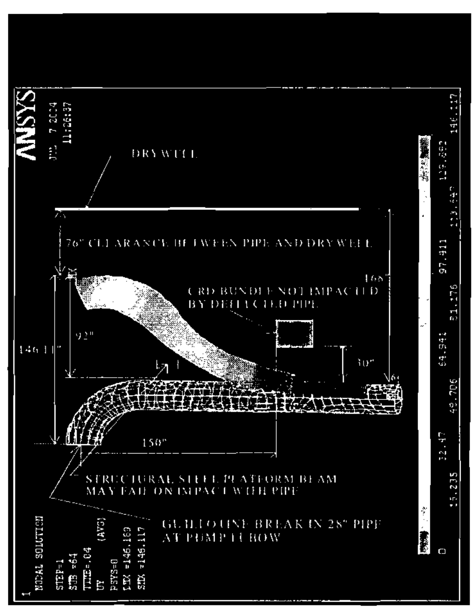
RUPTPROB for Mark I and II–GE3/GE4 Containment

Impact of RHR pipe on the CRD piping is improbable
RUPTPROB can be conservatively taken as 0.1 for CDF calculations



RUPTPROB for Mark I–GE3/GE4 Containment (Contd)

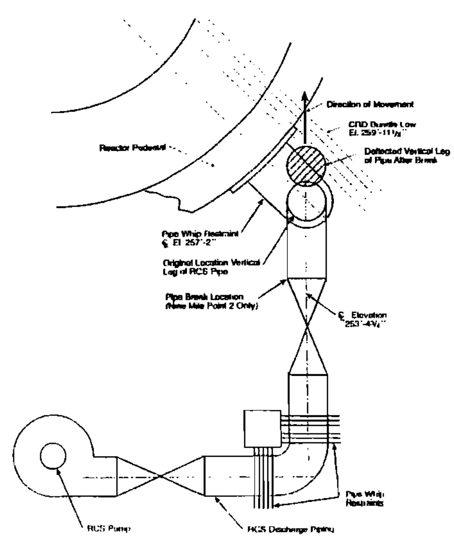
- Deflected Shape of the RCS Pipe Relative to CRD Piping
- Impact of RCS pipe on the CRD piping is improbable
- RUPTPROB can be conservatively taken as 0.1 for CDF calculations



RUPTPROB for Mark II–GE5 Containment

- Layout of RHR and CRD piping similar to Mark II GE3/GE4 plants
- Pipe whip restraint will prevent RHR pipe to impact CRD piping
- There is a 18 inch gap between RHR and CRD piping
- RCS pump discharge pipe break downstream of isolation wall postulated only for the Nine Mile Point 2
- There is a possibility of RCS piping impacting the CRD bundle if the pipe whip restraint fails
- After impact with RCS piping, the CRD piping will bend without significant crushing or crimping before rupture
- Pipe whip restraints on the vertical leg of RCS piping and on circular header will prevent RCS pump vertical discharge pipe break impact on the CRD piping
- RUPTPROB can be conservatively taken as 0.1 for all CDF calculations

RCS Pump Discharge Pipe Break Downstream of Isolation Valve in Mark II-GE5 Containment



Probabilistic Analysis Approach

- Initiating event frequency – used "classic" large LOCA frequency, lognormal distribution
- PIPETYPE – Normal distribution, based on four existing licensee submittals
- TYPEFRAC – Normal distribution, based on review of plant drawings
- RUPTPROB – Exponential distribution, based on ANSYS calculations

17

Probabilistic Analysis Results - Sequence frequency

Product Line	Sequence	Point Estimate	Mean	Median	5th Percentile	95th Percentile
GE2 Mark I	RCS	2.7E-7	2.8E-7	5.9E-8	1.8E-9	1.1E-6
GE3&4 Mark I	RCS	3.9E-7	4.0E-7	8.6E-8	2.6E-9	1.6E-6
	RHR	1.3E-7	1.3E-7	2.3E-8	8.6E-10	5.4E-7
	total	5.2E-7	5.3E-7	1.2E-7	4.0E-9	2.1E-6
GE4 Mark II	RCS	3.0E-8	5.1E-8	8.3E-9	1.7E-10	2.0E-7
	RHR	2.7E-7	2.8E-7	5.9E-8	1.8E-9	1.1E-6
	total	3.0E-7	3.3E-7	7.2E-8	2.3E-9	1.3E-6
GE5 Mark II	RCS	2.7E-7	2.8E-7	5.9E-8	1.1E-9	1.1E-6
	RHR	3.0E-7	3.1E-7	6.6E-8	2.1E-9	1.3E-6
	total	5.8E-7	5.9E-7	1.3E-7	4.6E-9	2.4E-6

Public Risk

- Estimated for GE4 in Mark I containment (most common)
- Based on NUREG-1150 plant damage state for ATWS initiated by stuck-open safety/relief valve
- Result was less than one person-rem per reactor-year

Conclusions

- Core damage frequency and public risk are well below thresholds
- Generic Issue GI-80 will be closed out with no additional requirements

(5)

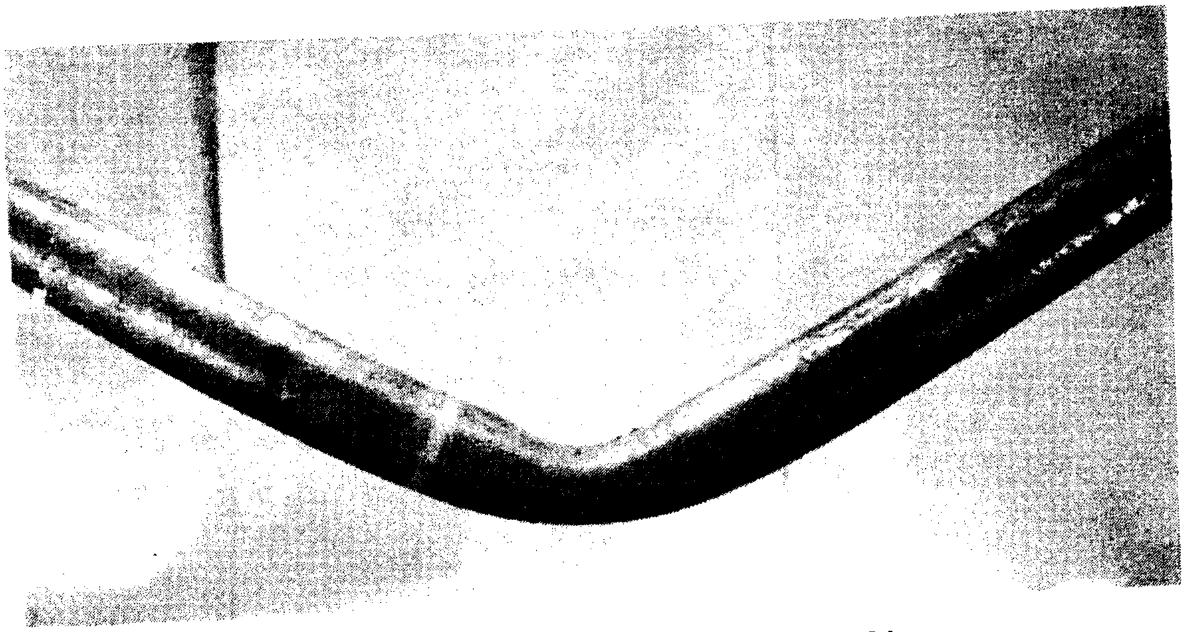


FIGURE 2.20. Impacted 3" SCH 160 Pipe

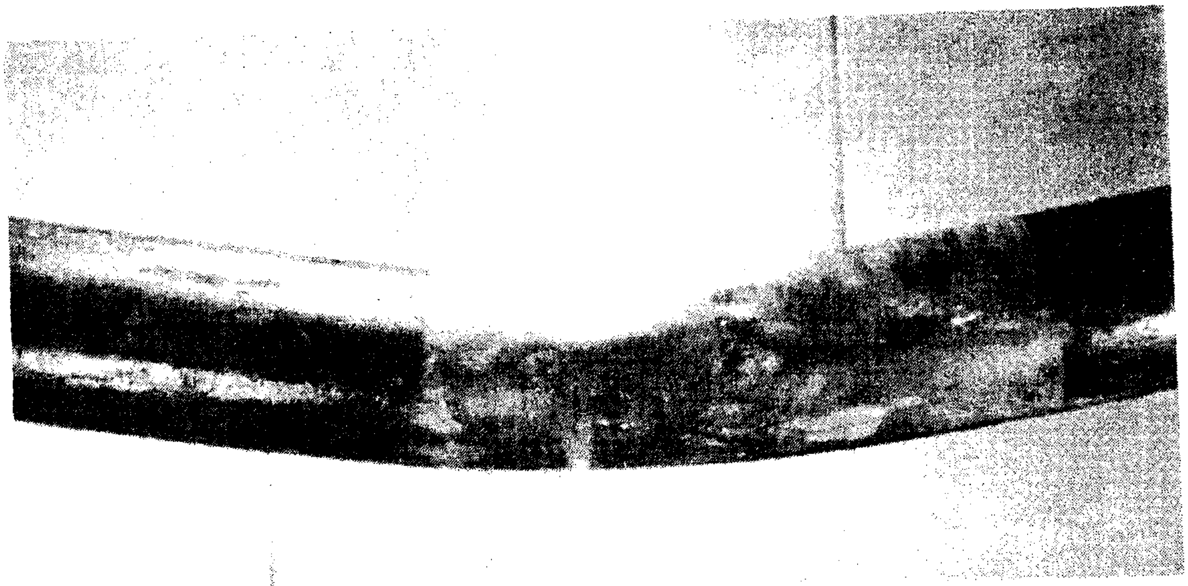


FIGURE 2.21. Impacted 6" SCH 40 Pipe

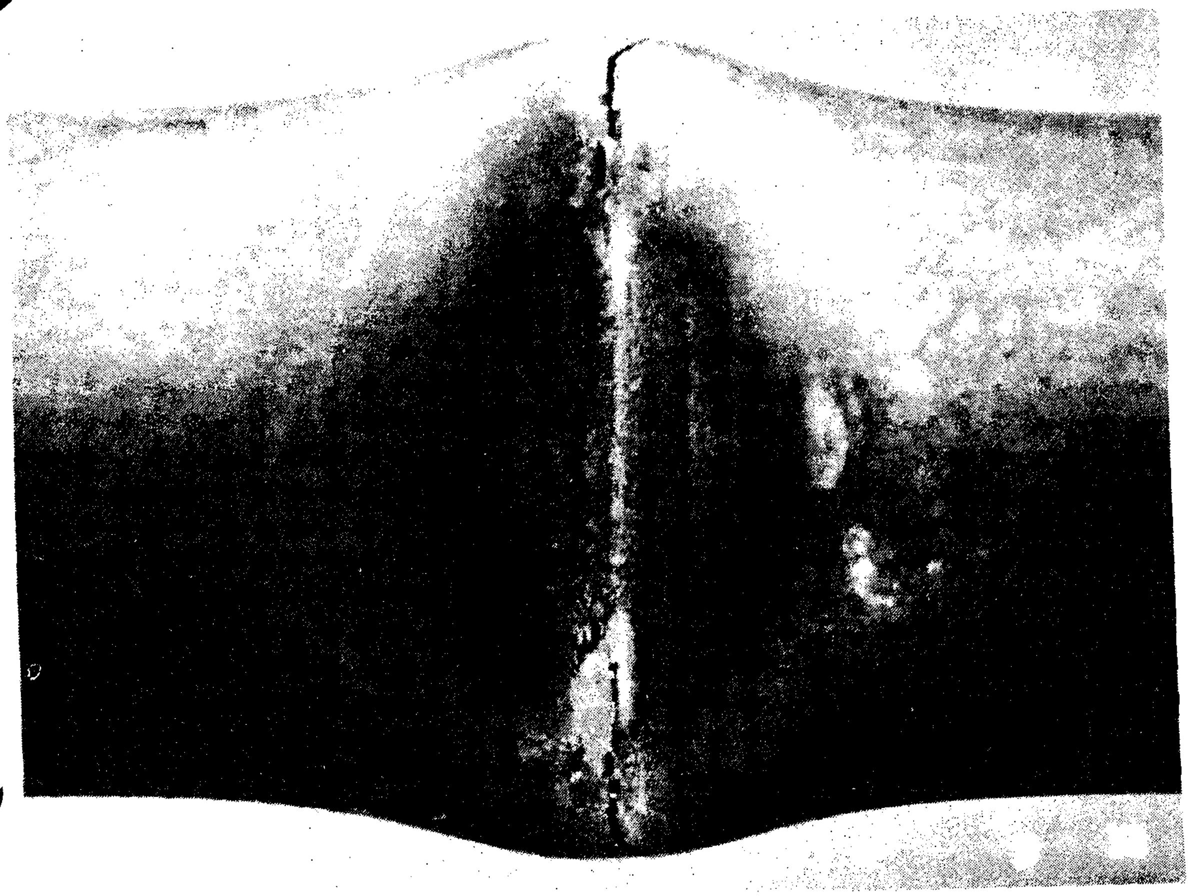


FIGURE 2.24. Failed 6" SCH 40 Pipe - Top View

FIGURE 2.25. Failed 6" SCH 40 Pipe - Side View

(6)

REGULATORY GUIDE FOR NFPA 805 RULE
ADVISORY COMMITTEE FOR REACTOR SAFEGUARDS
October 6, 2005

Paul Lain. Project Manager for NFPA 805 Implementation
Fire Protection Section
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Status of NFPA 805 Implementation

- Duke Power Sends Letter of Intent – 2/28/05
  (3 sites, 7 plants)

- Progress Energy Sends Energy Letter of Intent – 6/10/05
  (4 sites, 5 plants)

- NRC/RGN II/Progress Energy/Duke Power Kicks Off of Pilot Implementation – 8/11/05

- First Observation Visit Scheduled for November 2005

High Level Issues on Transition Plans - Use of Fire PRAs

- Each plant that transitions to NFPA 805 plans to trace cables and develop or enhance Fire PRAs.
- Progress Energy requested an extension to the discretion period to develop Fire PRAs.
- DSSA and Office of Enforcement is considering changes to accommodate additional changes necessary to enforcement discretion policy to enable the development of fire PRAs
- Staff has informed licensees that transition to NFPA 805 without a fire PRA is impractical
- Staff plans to use completed and emerging regulatory guides, RES products, and industry standards to ensure that the pilots rely on acceptable methods for PRA and fire Modeling

REGULATORY GUIDE FOR NFPA 805 RULE
ADVISORY COMMITTEE FOR REACTOR SAFEGUARDS
October 6, 2005

Bob Radlinski
Fire Protection Engineer
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

OBJECTIVE

- Review ACRS comments and Staff responses

- Describe changes made to the NFPA 805 Regulatory Guide and NEI 04-02 to address ACRS comments

ACRS Comments

- ACRS Comment: The Regulatory Guide should not be issued in its present form.
- How Addressed: The Regulatory Guide has been revised to incorporate the ACRS comments. We plan to issue the Reg Guide next year after submitting draft final versions of the Reg Guide and NEI 04-02 to the ACRS in December 2005

ACRS Comments (cont)

- ACRS Comment: The "initial fire modeling" approach should not be used as an alternative to estimates of changes in CDF and LERF.
- How addressed: NEI 04-02 has been revised to eliminate this approach (See revised Figure 5-1)

ACRS Comments (cont)

- ACRS Comment: The staff should not endorse methods for evaluating ΔCDF and ΔLERF that are not based on a fire PRA

- How Addressed: 10 CFR 50.48(c) and NFPA 805 allow risk assessments to be performed without a full fire PRA. However, to the extent possible, licensees are encouraged to develop a full fire PRA and the Regulatory Guide does not specifically endorse non-PRA methods.

ACRS Comments (cont)

- ACRS Comment: NEI 04-02 contains many statements that are inconsistent with the Commission's policy of promoting the use of PRA methods. In the Regulatory Guide, the staff should make it clear that it does not endorse such statements.

- How Addressed: Statements in Appendix J, "Plant Change Evaluations", and Section 5.3, "Plant Change Process", were revised to be consistent with revised Figure 5-1.

ACRS Comments (cont)

- ACRS Comment: The staff should ensure that the parts of NEI 04-02 that it endorses use correct methodology and language.
- How Addressed:
  - Held public meeting to share ACRS comments with NEI and discuss resolution
  - Held several follow-up phone calls with NEI
  - NRR and RES staff members reviewed all draft revisions to NEI 04-02
- Based on our reviews and discussions with NEI, we believe that the methodology and language used in the final version are correct.

NFPA 805 Regulatory Guide Changes

- Agreed with ACRS Comments and incorporated in final documents
- State that risk evaluations (for non-screened changes) should use PRA methods and tools
- Added PRA quality references including RG 1.174, RG 1.200 and the ANS fire PRA standard
- Noted that future additional guidance for fire PRAs will follow these referenced documents

NEI 04-02 Changes

- Agreed with ACRS comments and incorporated in final documents
- Eliminated all statements indicating that a change could be evaluated using the fire modeling approach without a risk assessment
- Encourages licensees to use a detailed, quantitative approach to plant change risk assessments

NEI 04-02 Changes (cont)

- Clarifies safety factors used to address uncertainties associated with fire models (the Reg Guide includes statement that margin must be large enough to bound uncertainties)
- Clarifies simplified assessment of ΔLERF

REGULATORY GUIDE FOR NFPA 805 RULE
ADVISORY COMMITTEE FOR REACTOR SAFEGUARDS
October 6, 2005

Sunil Weerakkody, Chief
Fire Protection Section
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
October 6, 2005

OBJECTIVE

- Inform ACRS about how the NRC Staff and NEI addressed ACRS Comments
- Inform ACRS about other issues remaining to be addressed
- Seek ACRS agreement with respect to changes made to the Regulatory Guide and NEI-04-02 to address ACRS comments.

STAFF INTRODUCTION

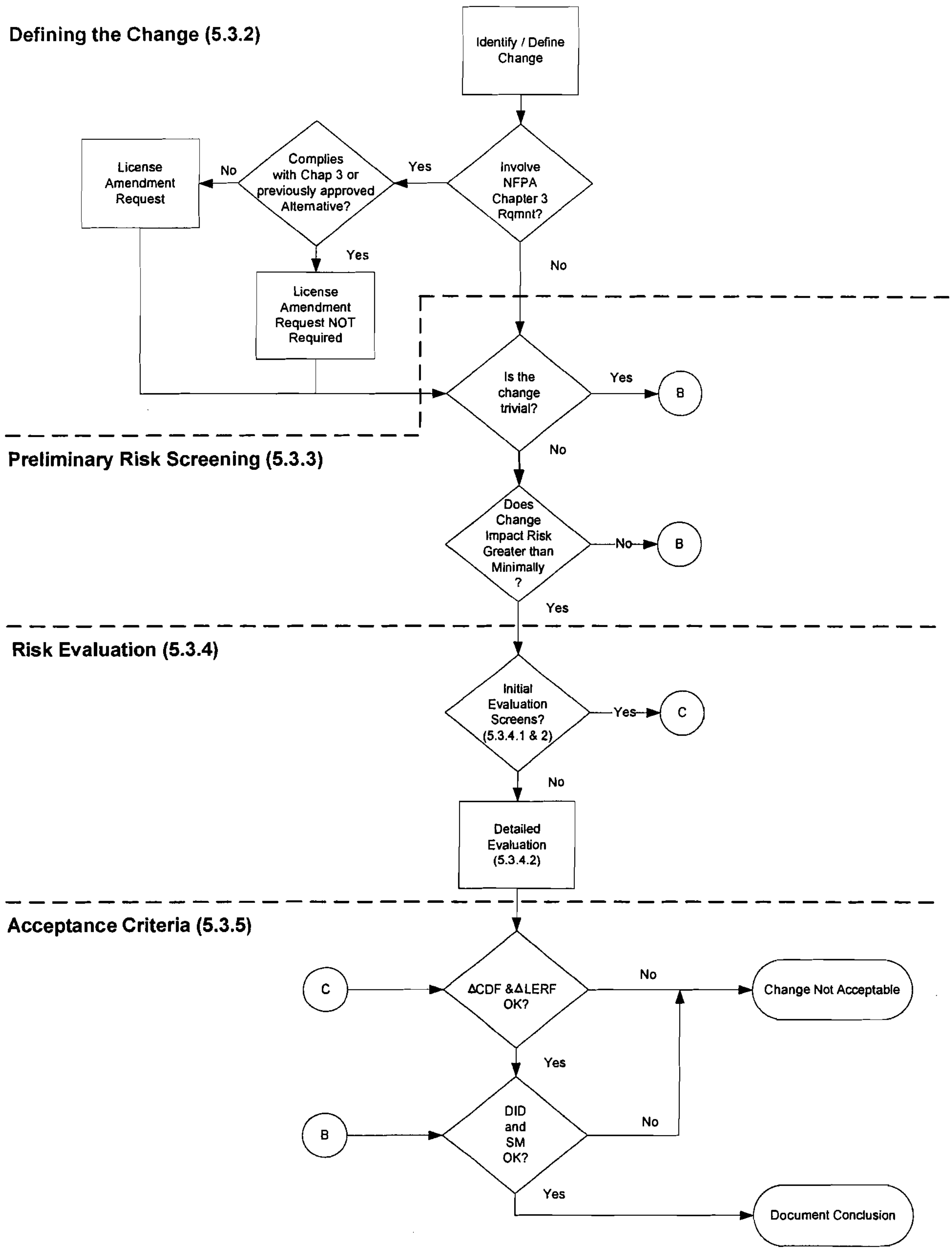
- Outline:
  - Status of NFPA 805 Implementation – Paul Lain
  - Changes to the Regulatory Guide and NEI-04-02 – Robert Radlinski
  - Additional changes to the Regulatory Guide – Sunil Weerakkody

NEXT STEP

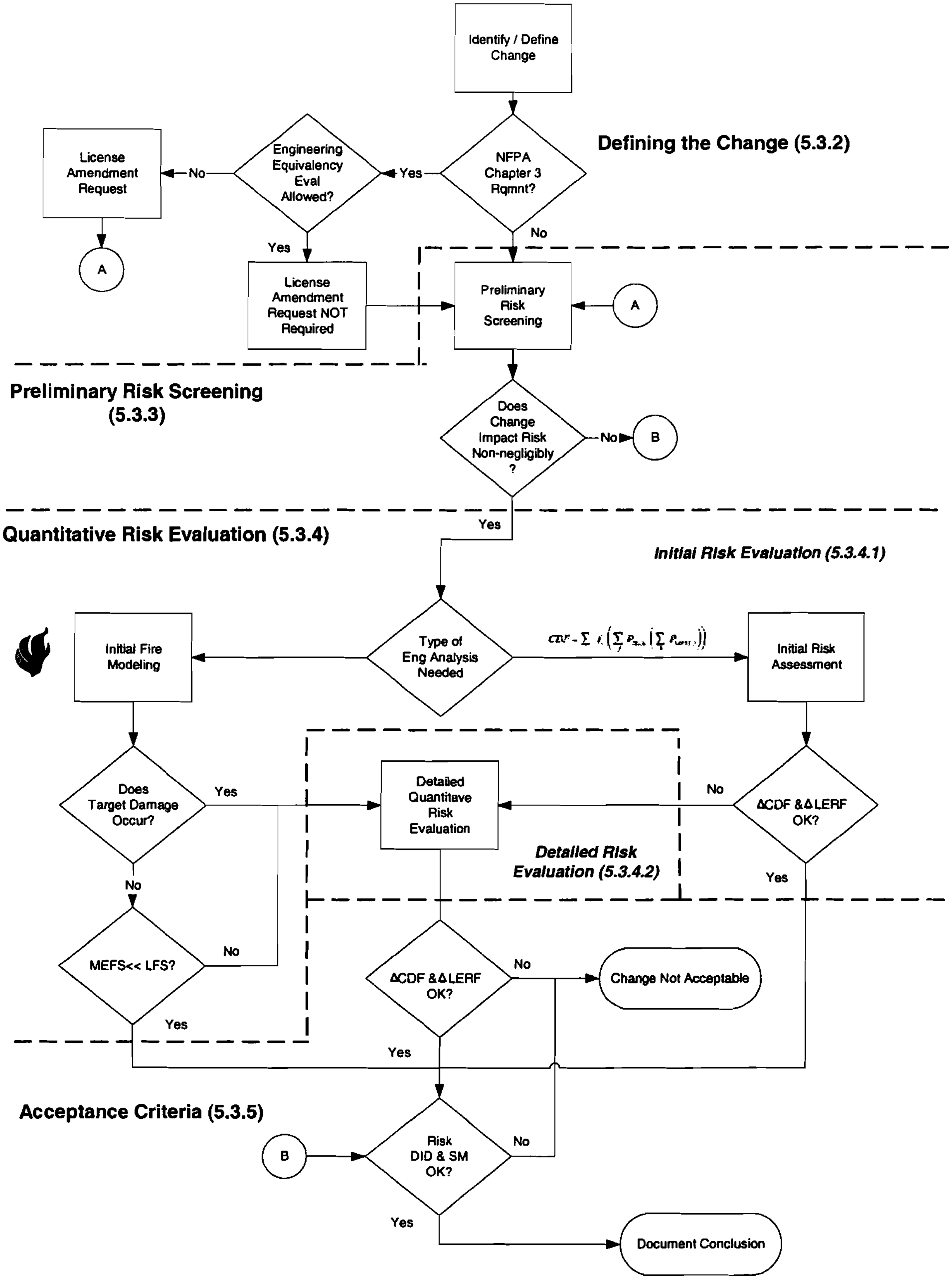
- Will provide the finalized Regulatory Guide and NEI-04-02 to ACRS, and seek endorsement to issue the Regulatory Guide after all changes are made

OTHER ISSUES

- 10 CFR 50.69, 10 CFR 50.48(a), 10 CFR 50.48(c)
- "Self-Approval"
  - Prior approval of methods
  - Threshold values
- Cumulative Risk
  - Baseline CDF
  - Tracking



NEI 04-02 Figure 5-1 Revision 1



NEI 04-02 Figure 5-1 Revision 0

10

An Assessment of the Structural Integrity Challenge Posed by Boric Acid Wastage in the Davis Besse RPV Head

Mark EricksonKirk
Office of Nuclear Regulatory Research
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B. Richard Bass, Paul Williams, Wally McAfee, and Sean Yin
Oak Ridge National Laboratory

ACRS Briefing
USNRC Headquarters • Rockville, MD • 6th October 2005

VG 1

Objectives of Our Analyses

- **As Found (only possible reality benchmark)**
  - **Assess the structural integrity of the primary reactor coolant pressure boundary for the conditions that existed at Davis Besse on February 16, 2002**
- **Looking forward (SDP support)**
  - **Assess the structural integrity of the primary reactor coolant pressure boundary for conditions postulated to exist at Davis Besse had it not been taken off-line for a scheduled maintenance outage on February 16, 2002**
- **Looking backward (ASP support)**
  - **Assess the structural integrity of the primary reactor coolant pressure boundary for conditions postulated to exist at Davis Besse for February 16, 2002 minus one year (ASP analysis)**

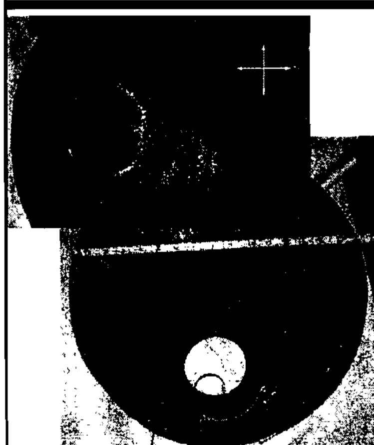
VG 2

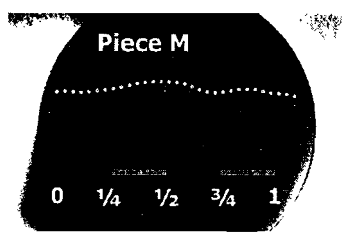
Presentation Outline

- **Description of the as found state**
- **As Found analysis**
  - **Methodology**
  - **Results**
- **Forward & backward looking analyses**
  - **Methodology**
  - **Results**

VG 3

16th Feb 02 Conditions at Davis Besse
("as found")

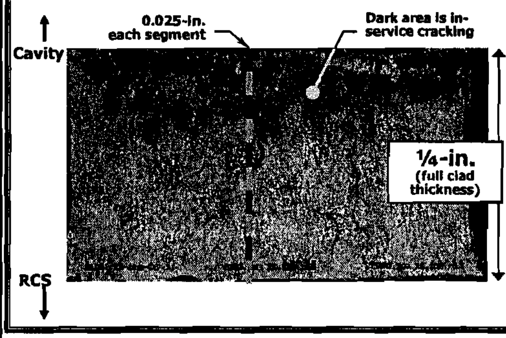




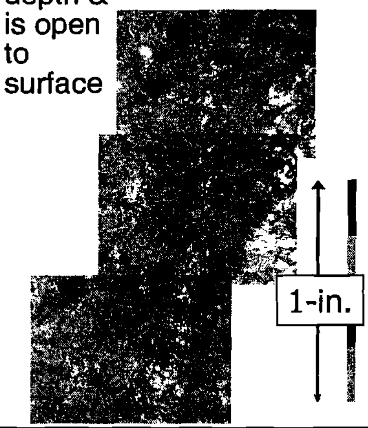
VG 4

Davis Besse Crack Characterization

Crack Depth
- 0.1-in max (over 20% of length)
- 0.065-in. average

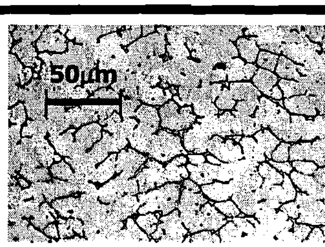


Crack Length
- 2-in max
- Central 0.66-in. has significant depth & is open to surface



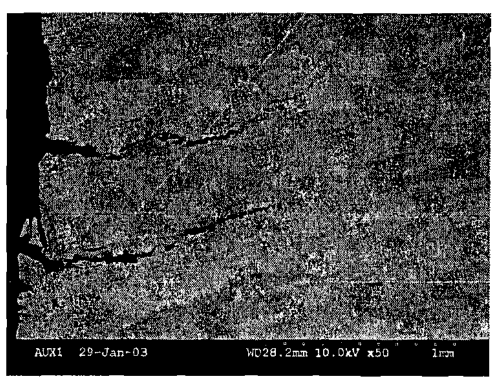
VG 5 [All metallography by J. Hyres, BWXT]

Crack Extension Mechanism



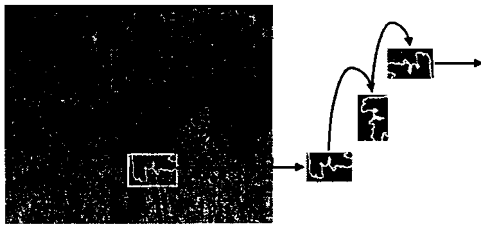
Dendritic solidification structure in 308SS
- Dark = ferrite
- Light = austenite

Cracks in the 308 stainless steel cladding formed when the concentrated boric acid solution in the cavity preferentially dissolved the ferrite phase. Thus, the cracking is intergranular (i.e., between the austenite grains).



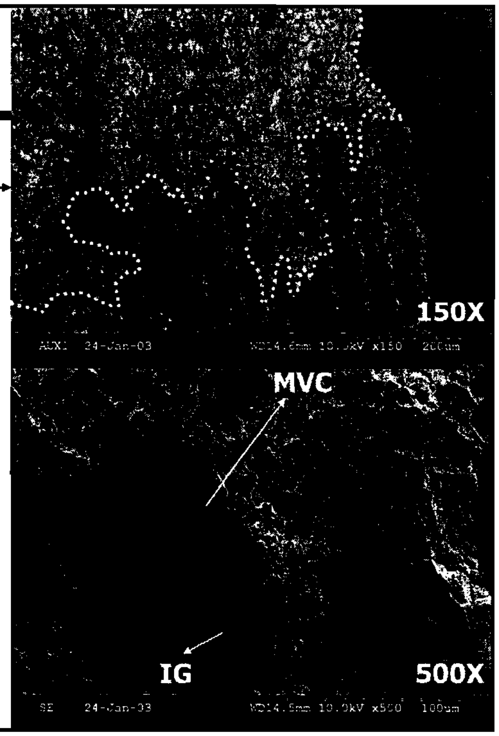
VG 6

Crack Morphology

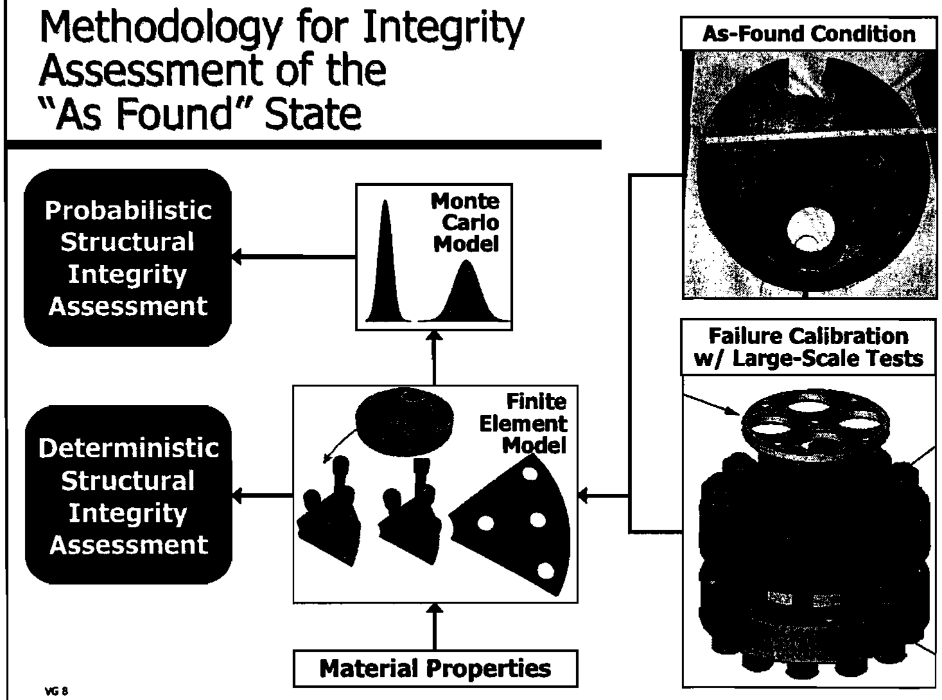


- In-service cracks are inter-granular & result from preferential attack of ferrite phase by boric acid.
- Lack of ductile tearing on service darkened side of fracture suggests that operating pressure did not load cladding above the ductile crack initiation threshold (i.e., cladding rupture was not imminent on 16 Feb 02).

VG 7



Methodology for Integrity Assessment of the "As Found" State



VG 8

Input Information

- **As-found configuration**
  - **Cavity geometry**
  - **Crack size and morphology**

  While described on the preceding slides, the totality of this information was not available at the outset

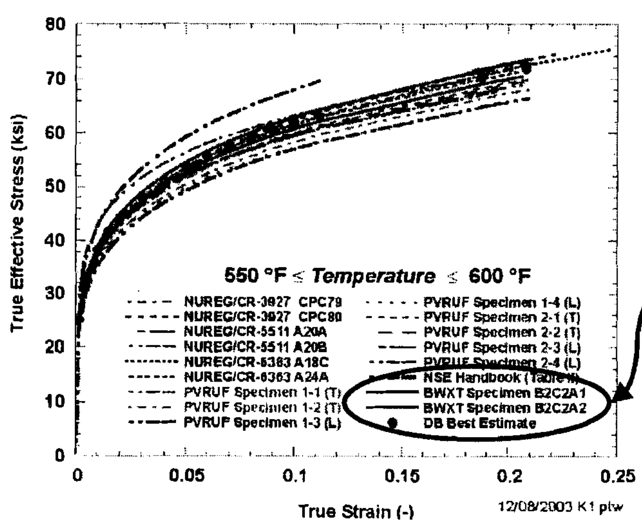
  - ✓ **June 2003: BWXT failure analysis report (for Framatome ANP)**
  - ✓ **April 2004: BWXT report on detailed crack examination (for ORNL)**

- **Cladding strength & fracture toughness properties**

- **Cladding failure mode & predictive benchmark**
  - **Discerned based on burst testing**

VG 9

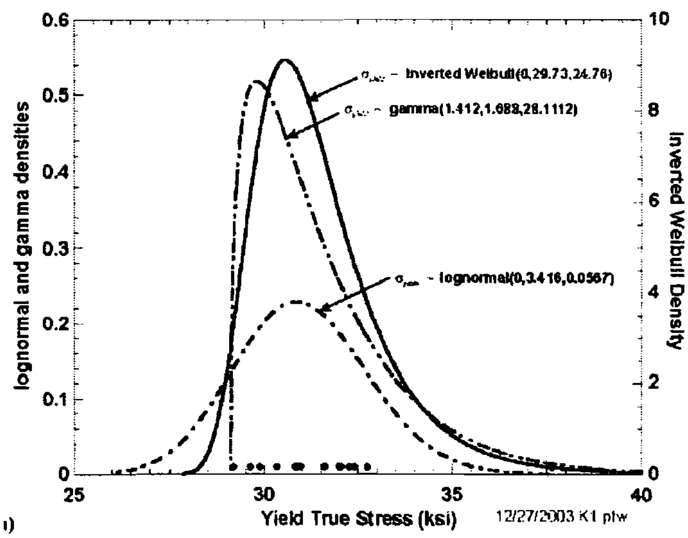
Cladding Strength



Tests on DB cladding agree well with literature data

VG 10

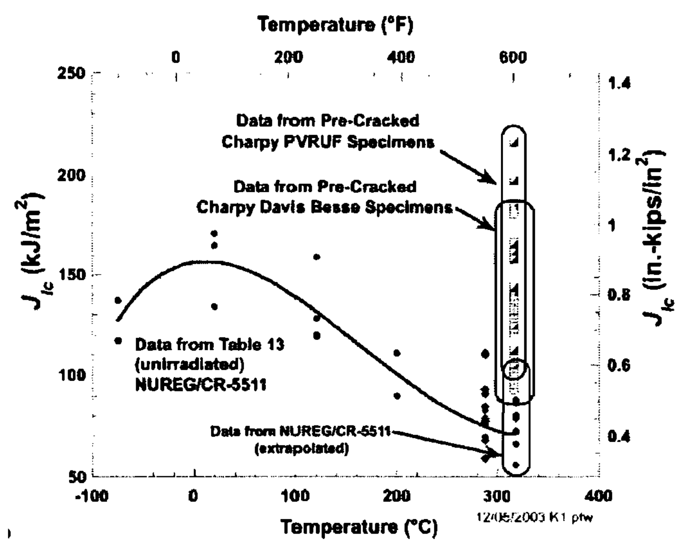
Cladding Strength



These data provide the basis for a statistical description used in our Monte Carlo analysis.

VG 11

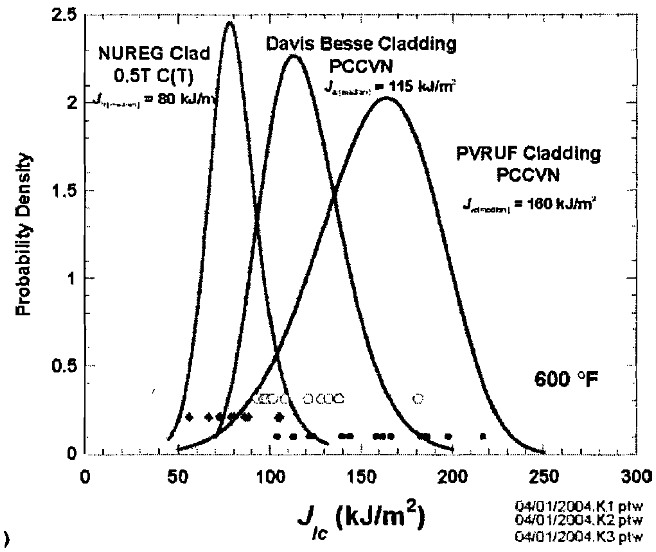
Cladding Fracture Toughness



Tests on DB cladding agree with literature data

VG 12

Cladding Fracture Toughness



These data provide the basis for a statistical description used in our Monte Carlo analysis.

VG 13

Burst Testing

- **Captures the essential structural characteristics of state of DB on 2-16-02**
  - **Un-backed cladding area**
  - **Flaw depth**
- **Not intended as a 1:1 model or representation of DB**
- **Objectives of tests**
  - **Validate opinion that cladding will fail by ductile tearing**
  - **Asses accuracy / conservatism in model predictions of failure loads**

Pre-Test

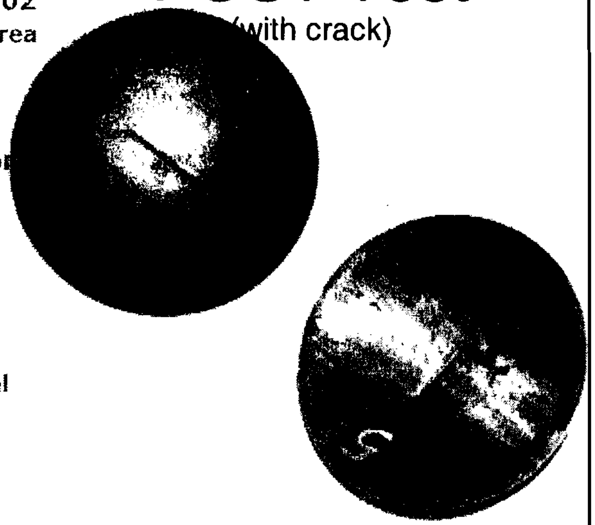
VG 14

Burst Testing

- Captures the essential structural characteristics of state of DB on 2-16-02
  - Un-backed cladding area
  - Flaw depth
- Not intended as a 1:1 model or representation of DB
- Objectives of tests
  - Validate opinion that cladding will fail by ductile tearing
  - Asses accuracy / conservatism in model predictions of failure loads

POST Test

(with crack)



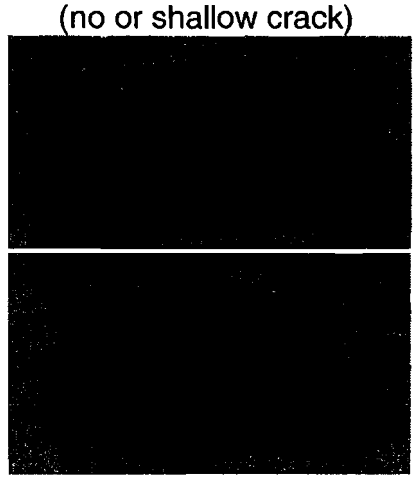
VG 15

Burst Testing

- Captures the essential structural characteristics of state of DB on 2-16-02
  - Un-backed cladding area
  - Flaw depth
- Not intended as a 1:1 model or representation of DB
- Objectives of tests
  - Validate opinion that cladding will fail by ductile tearing
  - Asses accuracy / conservatism in model predictions of failure loads

POST Test

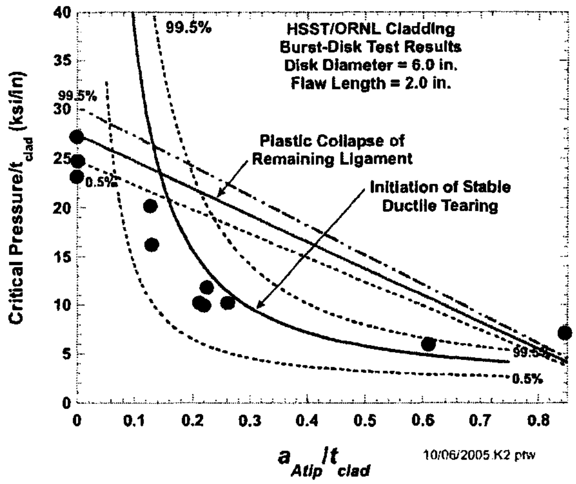
(no or shallow crack)



VG 16

Comparison of Burst Test Data with Model Predictions

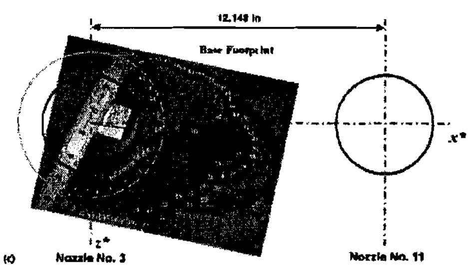
- Good test-to-test repeatability
- For deeper cracks (> ≈15% of the cladding thickness
  - Stable failures (leaks)
  - Predicted well by ductile cracking model
- For shallower cracks or no cracks
  - Catastrophic failures (blowout of un-backed area)
  - Predicted well by plastic collapse model



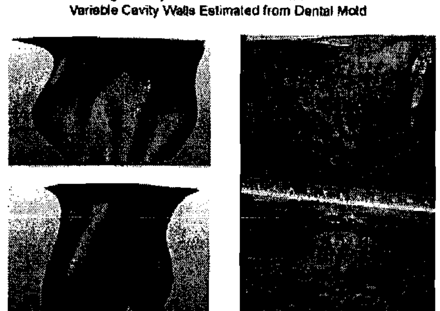
✓ Failure mode characterized
✓ Model validated

VG 17

Geometric Inputs to Finite Element Model of As-Found State



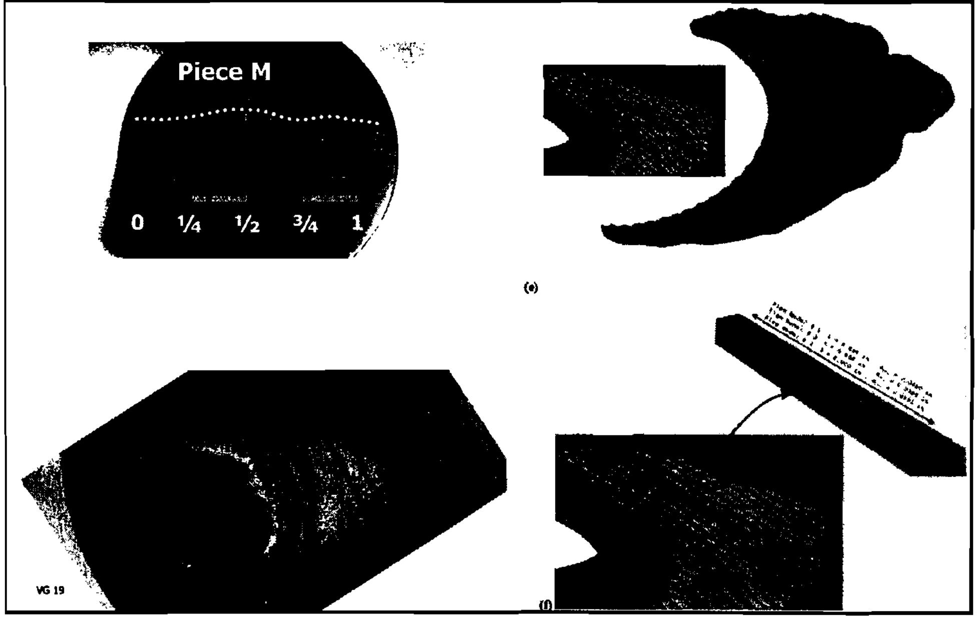
(c) Wastage Cavity Submodel Includes Definition of Variable Cavity Walls Estimated from Dental Mold



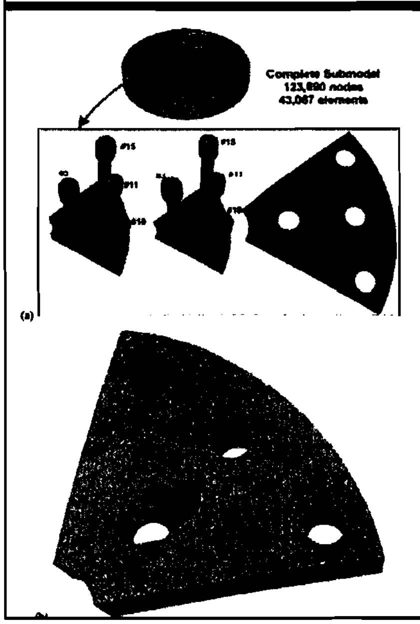
(d)

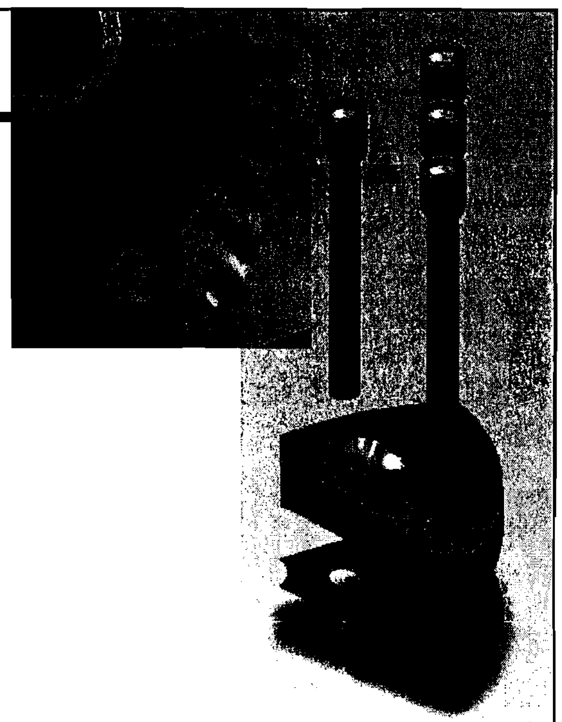
VG 18

Geometric Inputs to Finite Element Model of the As-Found State



Finite Element Model of As-Found State





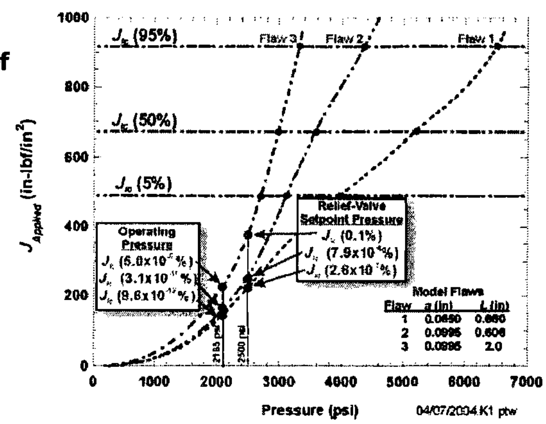
As-Found Analysis

- is based on the following
  - Geometrically accurate finite element model to estimate stresses
  - Actual DB properties for cladding and ferritic steel strength
  - Actual DB properties for cladding toughness
  - 3 Different idealizations of as-found crack network
    - ✓ Flaw 1: Best-estimate depth (0.065-in.) & length (0.66-in.)
    - ✓ Flaw 2: Bounding depth (0.1-in.) & best estimate length (0.66-in.)
    - ✓ Flaw 3: Bounding depth (0.1-in.) & length (2-in.)

VG 21

As-Found Analysis Results

- Prediction
  - Pressure in excess of relief valve setpoint pressure needed to rupture cladding
  - Factor of at least 1¼ safety margin exists against ductile crack initiation occurring at the operating pressure even assuming
    - ✓ A bounding flaw characterization
    - ✓ A lower bound fracture toughness characterization

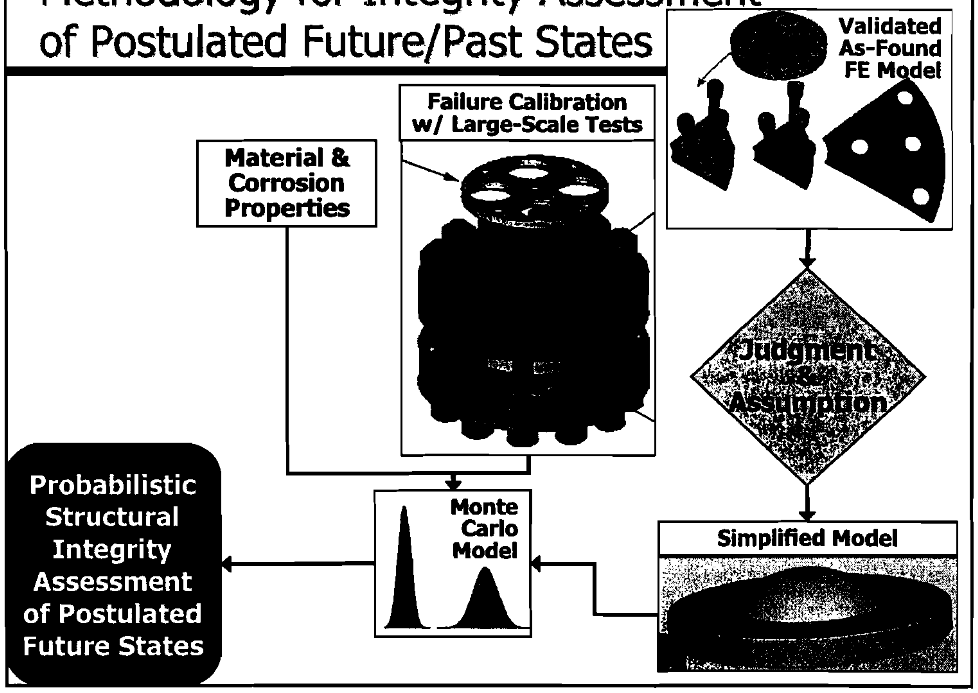


- Our prediction is that
  - Failure did not occur
  - Ductile crack initiation from crack network did not occur

- What actually happened was
  - Failure did not occur
  - Ductile crack initiation from crack network did not occur

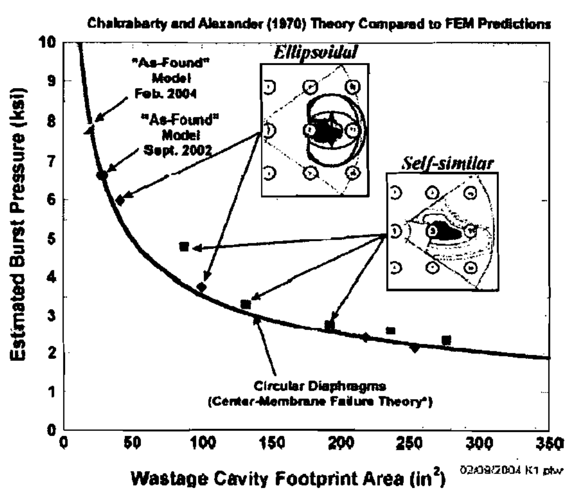
VG 22

Methodology for Integrity Assessment of Postulated Future/Past States



Assumption

- ***The complex cavity shape can be modeled as a circle***
  - **For failure by plastic collapse, un-backed cladding area is much more important than cavity shape**
  - **For failure by ductile tearing, circular assumption is conservative**
  - **State of knowledge was (and is) not adequate to defend a more complex representation of how the cavity shape evolved (or would have evolved) with time**



VG 24

Input Information

- Based on statistical representations of data
  - Toughness properties
  - Strength properties

Discussed already

- Based on engineering judgments
  - LOCA binning rules
  - Statistical fitting of data
- Based on expert opinions benchmarked to data
  - General corrosion properties of the ferritic RPV steel (controls cavity growth rate)
  - Corrosion crack growth properties of the austenitic cladding (controls crack depth growth in cladding)
- Based on expert opinion (for ASP analysis only)
  - Crack depth on 2-16-02 *minus one year*
  - Cavity size on 2-16-02 *minus one year*

VG 25

Engineering Judgments

- LOCA binning rules
  - Small up to 3.5" diameter
  - Medium from 3.5" – 4.8" diameter
  - Large > 4.8" diameter
  - "Conservative" model equates thru-clad cracking with failure of cladding
  - "Best estimate" model assesses stability of thru-clad crack based on standard *J-R* curve analysis techniques
- Statistical fitting of data
  - Illustrated on the following slides

VG 26

Expert Elicitation Information

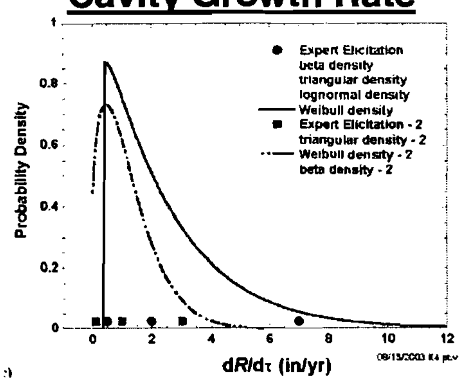
Judgments Obtained from Informal Expert Elicitation

Parameter Description	Parameter Units	Subject Matter Expert ID No.#	Lower Bound	Median (BE) Value	Upper Bound
			Associated Percentiles		
			5%	50%	95%/99.9%
Effective Cavity		#1	0	1.25	2
Radius at TOD-1	(in.)	#2	0	1.125	2.5
R_0		#3	0	1.5	2.25
Effecive Cavity		#1	1	2	7
Wastage Rate	(in./year)	#2	0.5	2	6
$dR/d\tau$		#3	0.75	1.5	(-)
Flaw Initiation		#1	12	36	48
Time w.r.t. TOD	(months)	#2	1	6	60
$\Delta\tau_{flaw\text{-}ini}$		#3	(-)	(-)	(-)
Effective Flaw		#1	0.001	0.01	0.1
Growth Rate	(in./month)	#2	0.001	0.01	0.1
$da/d\tau$		#3	0.004	0.01	0.04

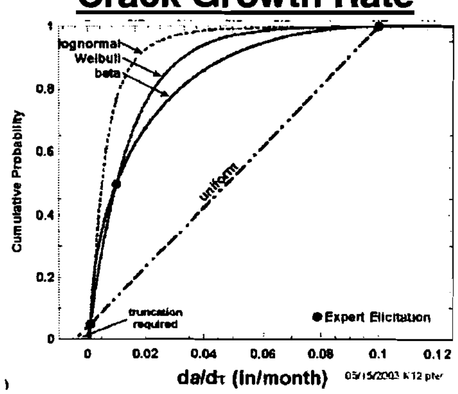
VG 27

Statistical Representation of Expert Elicitation Information (Examples)

Cavity Growth Rate



Crack Growth Rate



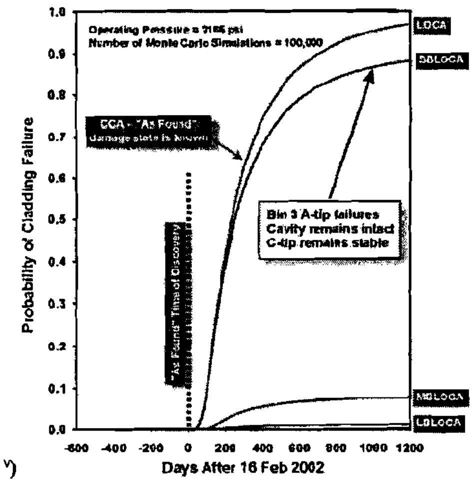
Sampling Distributions Categories	$dR/d\tau$ (in/yr)	$da/d\tau$ (in/month)	R_0 (in)	$\Delta\tau_{flaw\text{-}ini}$ (months)
BE	beta-1	Weibull	beta	Weibull
MC	triangluar	lognormal	triangular	lognormal
LC	Weibull	uniform	normal	triangular

VG 28

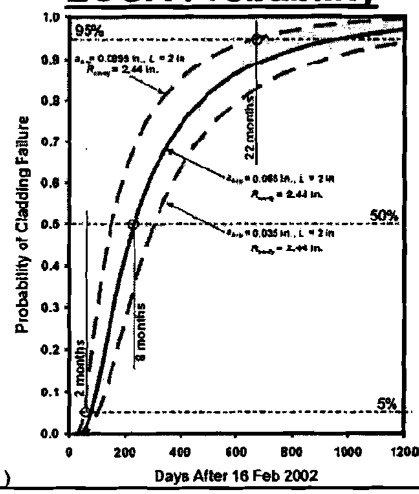
Results of "Forward Looking" Analysis

(as-found state known)

LOCA Size Breakdown



Effect of Assumed Crack Size on Total LOCA Probability



Results of "Forward Looking" Analysis

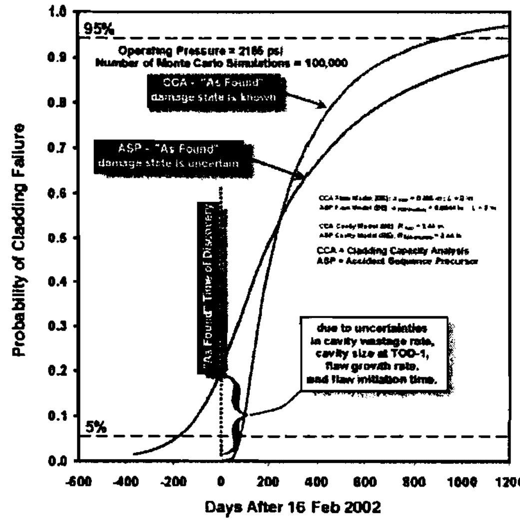
(as-found state known)

- **Based on bounding flaw model (Flaw 3: consistent with ASME practice) between 2 & 22 months of operation beyond 2-16-02 could have taken place before the cladding was compromised**
  - **Best estimate is 5 months**
- **Had the primary pressure boundary been compromised the most likely consequence was a small break LOCA**
  - **Known deep initial flaw depth favors SB-LOCA**

VG 30

Comparison of Forward and Backward Looking Analysis on the Predicted Total LOCA Probability on 2-16-02

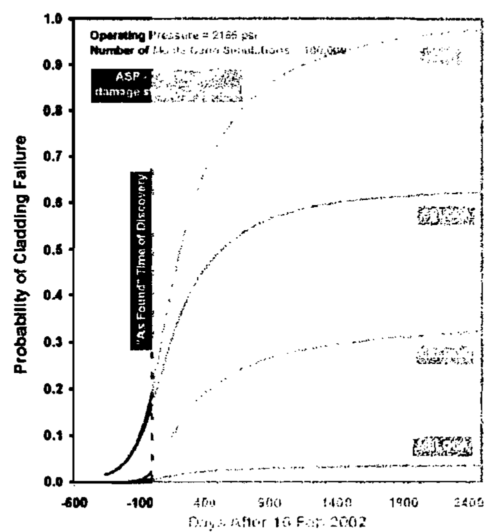
- **Backward looking analysis predicts an ≈20% LOCA probability on 2-16-02 (when, in fact, no LOCA occurred)**
- **This occurs as a direct consequence of uncertainty regarding the initial conditions assumed in the backward looking analysis, i.e.**
  - **Cavity size on 2-16-01**
  - **Depth of flaws in cladding on 2-16-01**



VG 31

Summary of Best-Estimate "Backward Looking" (ASP) Analysis

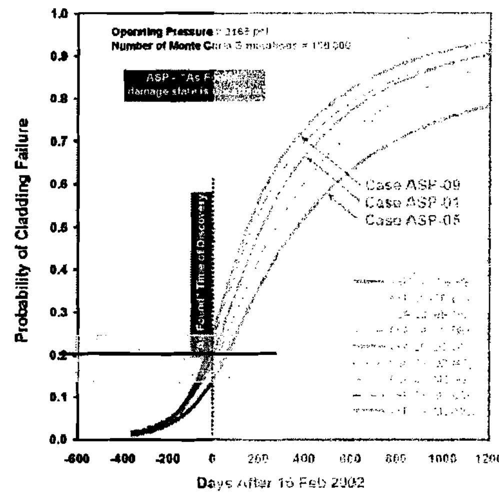
- **ASP analysis only uses predicted LOCA probabilities between 2-16-01 and 2-16-02**
- **LOCA probabilities for dates beyond 2-16-02 shown *for information only***
- **As was the case with the forward looking analysis, SB-LOCA is the most likely outcome had the pressure boundary been compromised**



VG 32

Effect of Uncertainties / Judgments on Total LOCA Probabilities

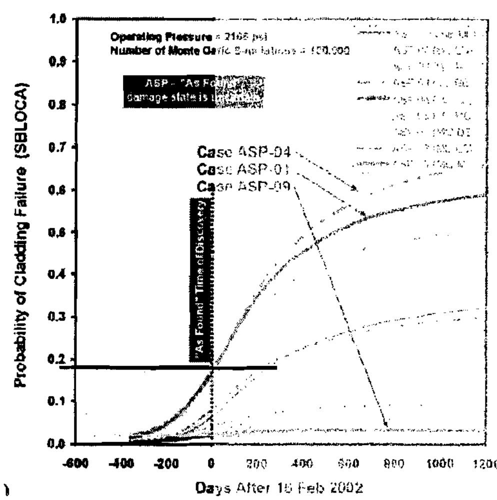
- **Total LOCA probability on 2-16-02**
  - **Min ≈ 14%**
  - **Max ≈ 24%**
  - ***Best estimate ≈ 20%***



VG 33

Effect of Uncertainties / Judgments on SB-LOCA Probabilities

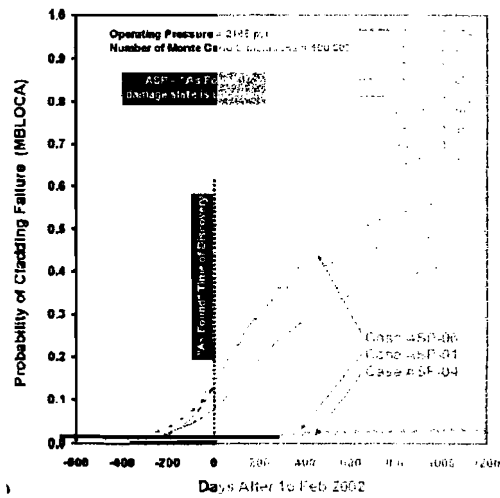
- **SB-LOCA probability on 2-16-02**
  - **Min ≈ 2%**
  - **Max ≈ 18%**
  - ***Best estimate ≈ 18%***



VG 34

Effect of Uncertainties / Judgments on MB-LOCA Probabilities

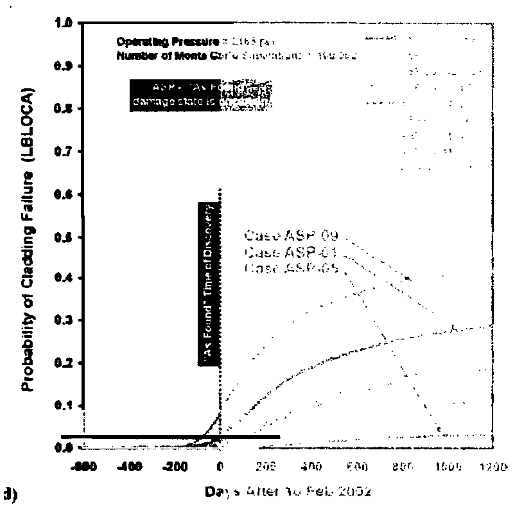
- MB-LOCA probability on 2-16-02
  - Min < 1%
  - Max ≈ 15%
  - ***Best estimate <1%***



VG 35

Effect of Uncertainties / Judgments on LB-LOCA Probabilities

- LB-LOCA probability on 2-16-02
  - Min = 0%
  - Max ≈ 9%
  - ***Best estimate ≈ 3%***



VG 36

Summary

- **As-found analysis**
  - **Forensic exams found no ductile tearing initiated from corrosion assisted flaws – suggests cladding rupture was not imminent**
  - **No crack initiation predicted on day of discovery**
  - **Pressure in excess of relief valve setpoint pressure would have been needed to rupture the cladding**
- **Forward looking analysis (as found condition known)**
  - **2-22 months more operation needed to rupture the cladding ... best estimate is ~5 months.**
  - **Most likely consequence of cladding rupture is SB-LOCA**
- **Backward looking analysis ... ASP (as found condition uncertain)**
  - **~20% (+/-5%) total LOCA probability on day of discovery**
  - **Most likely consequence of cladding rupture is SB-LOCA**

VG 37

(11)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th MEETING
DAVIS-BESSE REACTOR PRESSURE VESSEL HEAD INTEGRITY CALCULATIONS
Rockville, MD
October 6, 2005

- PROPOSED SCHEDULE -

	Topic	Presenter(s)	Time
I	Opening Remarks	J. Sieber, ACRS	2:30-2:40pm
II	Davis-Besse Reactor Pressure Vessel Head Integrity Calculations	A. Hiser, RES M. EricksonKirk, RES	2:40-3:45pm
III	Discussion and Closing Remarks	J. Sieber, ACRS	3:45-4:00pm
	Break		4:00pm

Notes:
- Presentation time should not exceed 50% of the total time allocated for a specific item.
- Number of copies of presentation materials to be provided to the ACRS - 35.

(12)

April 30, 2004

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Ashok C. Thadani, Director /RA/
Office of Nuclear Regulatory Research

SUBJECT: UPDATE ON STRUCTURAL INTEGRITY ASSESSMENT OF DAVIS-BESSE REACTOR PRESSURE VESSEL HEAD WITH CORROSION WASTAGE CAVITY

My memorandum to you dated January 8, 2003, summarized RES activities related to the degradation of vessel head penetration nozzles in pressurized water reactors, including estimates of the pressure necessary to fail the Davis-Besse RPV head in the as-found condition on February 16, 2002, and how long Davis-Besse could have operated before the cladding failed. My memorandum noted several uncertainties in the analyses, including those related to cracks found in the cladding. The attachment to the memorandum stated that the licensee would determine the depth of the cracks in the cladding and that the presence of cracks might necessitate a revision of the calculations and could possibly reduce the pressure margin identified in the original calculations.

This memorandum updates the January 8, 2003, memorandum, specifically addressing the influence of the cracks on the pressure necessary to fail the cladding and how long Davis-Besse could have operated before the cladding failed. Since the original calculations, additional work has been done in the following areas:

- Clad disk tests of samples with simple cavity and clad crack geometries,
- Characterization of cracks in the Davis-Besse cladding,
- Fracture toughness characterization of the Davis-Besse cladding,
- Development of a detailed finite element model of the Davis-Besse wastage cavity and cladding as they were on February 16, 2002, and
- Peer review by an independent external panel to review the experimental activities and the approach for the analytical work.

Our analyses of the pressure necessary to fail the cladding used two representations of the cladding cracks to provide understanding of the sensitivity to crack size. For the longer and deeper of these two crack representations (with length of 2 inches and depth 0.1 inches, consistent with an ASME Code representation for the multiple cracks in the cladding), estimates of the pressure necessary to fail the cladding range from 2700 to 3300 psi (at the 5^{th} and 95^{th} percentiles, respectively), with a median pressure of 3000 psi. For a shorter (0.66 inches) and shallower (0.065 inches) representation, estimates of the pressure necessary to fail the cladding range from 3900 to 6550 psi (at the 5^{th} and 95^{th} percentiles, respectively), with a median pressure of 5250 psi.

Considering the uncertainties in predicting the failure pressure for multiple flaws, in our engineering judgment the ASME Code representation of the cladding cracks is the more appropriate model. Thus, our judgment is that the margin against failure ranges from a factor of 1.2 to 1.5 of the operating pressure, with a median value of 1.4. These estimates are in agreement with the forensic evidence that the operating pressure of 2165 psi was insufficient to produce crack initiation. The margin against failure at the relief valve setpoint (2500 psi) ranges from a factor of 1.1 to 1.3 of the setpoint pressure, with a median value of 1.2.

Finally, we used a simplified model of the cavity geometry in Davis-Besse to estimate how long after February 16, 2002, Davis-Besse could have operated without failure of the stainless steel cladding. For the ASME Code representation of the cladding cracks, this model predicts an operating time of 2 to 13 months (at the 5^{th} and 95^{th} percentiles, respectively), with a median estimate of 5 months. For the shallower depth crack of 0.065 inches (and length of 2 inches), estimates of the operating time range from 3 to 13 months (at the 5^{th} and 95^{th} percentiles, respectively), with a median estimate of 8 months. There are significant uncertainties regarding the rate and direction of cavity expansion (for example, was the cavity continuing to grow? - our analysis assumes that the cavity was growing) and the rate of stress corrosion crack growth in the cladding. With our engineering judgment that the ASME Code representation of the cladding cracks is the more appropriate model, it is our conclusion that Davis-Besse could have operated for 2 to 13 months without failure of the cladding, with a median value of 5 months.

A more complete description of the experimental and analytical work performed is attached to this memorandum. At present we are preparing input for an Accident Sequence Precursor (ASP) analysis of Davis-Besse and finalizing the detailed documentation of this work, including the experimental testing, characterization of the cracks in the Davis-Besse cladding, the analytical modeling efforts, and the external panel review. In accordance with normal Agency process to evaluate the risk significance of operating conditions at nuclear power plants, the ASP analysis will evaluate the risk from the degradation of the reactor vessel head at Davis-Besse. A final engineering and analysis report will be issued when we report on the preliminary results and findings of the ASP analyses in early summer.

Attachment: As stated

Distribution:

MEB RF	LMarsh	WRuland	AMendiola	CLipa	PBaranowsky
GDemoss	WBateman	RTregoning	BSheron	SBurnell	MCheok
TMensah	LGerke	JDyer	JCaldwell	LReyes	DET r/f

DOCUMENT NAME: G:\DET\HISER\DB\DB Updated Analysis Memo.WPD

OAR in ADAMS? (Y or N) Y   ADAMS ACCESSION NO:________   TEMPLATE NO. RES:___

Publicly Available? (Y or N) N   DATE OF RELEASE TO PUBLIC______   SENSITIVE? N

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OFFICE	RES/DET/MEB	RES/DET/MEB, ABC	RES/DET/MEB, BC	
NAME	M. Kirk /RA/	A. Hiser /RA/	N. Chokshi /RA/	
DATE	4/12/04	4 / 12 /2004	4 /12 /2004	

OFFICE	RES/DET	RES, DDIR	RES, DIR	
NAME	M. Mayfield /RA/	J. Strosnider /RA/	A. Thadani /RA/	
DATE	4/20/2004	4 /21 /2004	4 /30 /2004	

ATTACHMENT

STRUCTURAL INTEGRITY ASSESSMENT OF DAVIS-BESSE REACTOR PRESSURE VESSEL HEAD WITH CORROSION WASTAGE CAVITY

A memorandum from A. Thadani (RES) to W. Travers (EDO) dated January 8, 2003, summarized RES activities related to the degradation of vessel head penetration nozzles in pressurized water reactors, including estimates of the pressure necessary to fail the Davis-Besse RPV head in the as-found condition on February 16, 2002, and how long Davis-Besse could have operated before the cladding failed. The memorandum noted several uncertainties in the analyses, including cracks found in the cladding. The attachment to the memorandum stated that the licensee would determine the depth of the cracks in the cladding and that the presence of cracks might necessitate a revision of the calculations and a possible reduction in the pressure margin identified in the original calculations.

Since the original calculations, additional work has been done in the following areas:

- Clad disk tests of samples with simple cavity and clad crack geometries,
- Characterization of cracks in the Davis-Besse cladding,
- Fracture toughness characterization of the Davis-Besse cladding,
- Development of a geometrically accurate finite element model of the Davis-Besse wastage cavity and cladding as they existed on February 16, 2002, and
- Peer review by an independent external panel to review the experimental activities and the approach for the analytical work.

This additional work is described below, along with an update to the calculations using the best available information and modeling available to the staff.

<u>Clad Disk Tests</u>

The failure calculations reported in the January 8, 2003, memorandum were based on a failure model which depended only on the strength of the cladding material, and is characterized as a net-section collapse model. It was chosen based on limited failure testing of thin plate specimens performed by EPRI well before the discovery of the Davis-Besse RPV head degradation. Initial testing was performed at Oak Ridge National Laboratory (ORNL) to confirm that this model was appropriate for cladding material, and in particular for cladding material containing cracks. The initial test results clearly indicated that the failures were dependent on the fracture toughness of the cladding and not just its strength properties. Additional testing was performed to validate this finding. ORNL performed a total of 11 tests of clad disk specimens machined from the pressure vessel of a canceled plant (note this was *not* cladding from the Davis-Besse RPV head). The geometry of these tests was simplified, with a circular 6-inch diameter "cavity" machined through the ferritic steel to provide an exposed cladding surface as the test piece. The circular cavity in these samples was similar in overall size to the cavity at Davis-Besse, but the surface area of exposed cladding in the tests (~ 28.3 square inches) was greater than at Davis-Besse (~16.5 square inches). The difference between the test configuration and the Davis-Besse condition is due to the J-groove weld in the Davis-Besse RPV head, which could not be incorporated into the ORNL tests. However, since the tests were designed to validate the failure model, this difference was not significant. Three of these tests were performed with no cracks in the cladding. The remaining tests had flaws machined into the cladding with a 2-inch length and depths ranging from 10% to 85% of the clad thickness. These tests demonstrated that, in the presence of cracks in the cladding, failure would occur consistent with a ductile tearing fracture mechanics model rather than the net-section collapse model used in the earlier analyses. Therefore, the failure model was revised and additional material property testing was initiated to provide appropriate properties for the Davis-Besse cladding.

Cladding Cracks

Based on work at BWXT Service (initially funded by FirstEnergy and then continued by RES funding), the Davis-Besse cladding was found to contain a complex network of stress corrosion cracks having a total extent on the surface (length) of ~2 inches. The longest contiguous portion of these cracks (0.66 inches in length) was in the central portion of the cracking coincident with the deepest cracking. However, the shorter crack segments were close to this longest segment but slightly offset from the axis of the crack. A maximum depth of 0.1 inch (40% of the cladding thickness) and an average depth of 0.065-in. (26% of the cladding thickness) were measured in this region. The maximum flaw depth occurred in small "fingers" that are characteristic of stress corrosion cracking. The 0.1 inch deep fingers were identified over ~20% of the central 0.66 inch of the crack network.

An additional, and very important, finding of the forensic examination is that the stress corrosion cracks in the Davis-Besse cladding showed no evidence of ductile tearing at the operating pressure (2165 psi), a necessary precursor to cladding failure. This finding provides a reality benchmark for our analysis of pressure margins reported below: a realistic model of the Davis-Besse as-found condition will not predict initiation of a ductile crack at the operating pressure for the conditions that existed on February 16, 2002.

Geometrically Accurate Model

A finite element model was developed that provides a geometrically accurate representation of the as-found Davis-Besse configuration, including the size and shape of the exposed cladding surface, the J-groove weld, and the control rod drive mechanism (CRDM) nozzles in the RPV head. Based on the flaw size information described above, two crack configurations were incorporated in the finite element model. A crack 2 inches long and 0.1 inch deep was adopted to represent the network of flaws in the cladding (essentially an envelope around the cladding cracks) since the shorter crack segments were close to the longest contiguous segment. This characterization of the crack network in the Davis-Besse cladding is consistent with American Society of Mechanical Engineers (ASME) Code rules regarding modeling of multiple cracks and with traditional fracture assessments of cracked components. A separate analysis included a crack 0.66 inch long and 0.065 inch deep based on the dominant crack in the cladding, as described previously. These two characterizations of the cracks were used to address the uncertainty in the failure pressure predictions caused by the multiple cracks. The 2 inch long crack provides a traditional prediction while the 0.66 inch long crack provides a more optimistic prediction.

External Review Panel

To provide an independent perspective on the experimental and analytical work, an external review panel, composed of the following individuals, was formed:

- Dr. William Shack of Argonne National Laboratory and the ACRS. Dr. Shack has expertise in materials analysis and corrosion.

- Dr. Gery Wilkowski of the Engineering Mechanics Corporation of Columbus. Dr. Wilkowski has expertise in fracture testing of both laboratory test specimens and large structural components and in fracture analysis of structural components.
- Professor James Joyce of the United States Naval Academy. Professor Joyce has expertise in fracture analysis and testing.

The review panel met with the staff and ORNL in early December 2003 and had several discussions with the staff after that time. Each reviewer submitted an independent letter to the staff (ADAMS accession ML041030107 and ML041110832), but all reviewers raised the following themes:

- While the clad disk tests provide useful information on the failure characteristics of the cladding, they should not be taken to represent the conditions that existed at Davis-Besse. Estimates of the Davis-Besse structural integrity should be based on a finite element analysis that represents much more closely the geometric conditions that existed at Davis-Besse on February 16, 2002, combined with laboratory data on the strength, toughness, and failure characteristics of the stainless steel cladding.
- The clad disk tests should have additional instrumentation to permit differentiation of crack initiation and failure.
- A better characterization of the crack network that existed in the Davis-Besse cladding is needed to support a realistic assessment of the as-found condition.
- Evidence on the fracture morphology of the cladding cracks does not suggest that failure was imminent on February 16, 2002.

These suggestions were incorporated in the final clad disk tests and analyses described previously.

<u>Updated Estimates of Davis-Besse Failure Conditions</u>

As-Found Condition on February 16, 2003

Ductile tearing fracture analyses were completed for the two crack characterizations, using the geometrically accurate finite element analysis of the cavity. This analysis accounted for the variability in strength and toughness properties of the stainless steel cladding. The variability in material property data was obtained directly from measurements on the Davis-Besse cladding. The strength and fracture toughness properties of the Davis-Besse cladding determined from the testing performed under this program were compared to values obtained for the cladding tested in the clad disk tests and to values previously obtained for archival cladding material. This comparison revealed that the Davis-Besse cladding has similar strength to the archival cladding material and the cladding from the clad disk tests, with the fracture toughness for the Davis-Besse cladding lower than that for the clad disk tests and higher than that for the archival cladding material.

For the ASME Code representation of the cladding cracks, estimates of the pressure necessary to fail the cladding range from 2700 to 3300 psi (at the 5^{th} and 95^{th} percentiles, respectively), with a median pressure of 3000 psi. For the shorter and shallower crack, estimates of the pressure necessary to fail the cladding range from 3900 to 6550 psi (at the 5^{th} and 95^{th} percentiles, respectively), with a median pressure of 5250 psi. Considering the uncertainties in predicting the failure pressure for multiple flaws, in our engineering judgement the ASME Code representation of the cladding cracks is the more appropriate model. Thus, our judgement is that the margin against failure ranges from a factor of 1.2 to 1.5 of the operating pressure, with a median value of 1.4. These estimates are in agreement with the forensic evidence that the operating pressure of 2165 psi was inadequate to produce crack initiation. The margin against failure at the relief valve setpoint (2500 psi) ranges from a factor of 1.1 to 1.3 of the setpoint pressure, with a median value of 1.2.

To provide an independent check on the ORNL analyses, the staff had one of the peer review panel members develop an estimate of the failure pressure. The panel member used an empirical approach that relies heavily on structural integrity assessment procedures developed and validated for ductile fracture by the gas transmission pipeline industry. Those estimates of failure pressures are consistent with the estimates developed by ORNL and reported above.

Continued Operation Beyond February 16, 2002

This analysis accounted for the variability in both the rate of cavity enlargement (assuming that the cavity was continuing to grow) and the rate of stress corrosion crack growth due to the concentrated boric acid solution inside the wastage cavity. To overcome the lack of empirical evidence on the cavity and crack growth rates, expert opinion was used to estimate these parameters and their variability. For the ASME Code representation of the cladding cracks, this model predicts that Davis-Besse could have operated for 2 to 13 month (at the 5^{th} and 95^{th} percentiles, respectively) without failure of the cladding, with a median estimate of 5 months. For a crack with a shallower depth of 0.065 inches (and length of 2 inches), estimates of the operating time range from 3 to 13 months (at the 5^{th} and 95^{th} percentiles, respectively), with a median estimate of 8 months. There are significant uncertainties regarding the rate and direction of cavity expansion (for example, was the cavity continuing to grow? - our analysis assumes that the cavity was growing) and the rate of stress corrosion crack growth in the cladding. With our engineering judgement that the ASME Code representation of the cladding cracks is the more appropriate model, it is our conclusion that Davis-Besse could have operated for 2 to 13 months without failure of the cladding, with a median value of 5 months.

Future Activities

Detailed documentation of this work is under preparation, including the experimental testing, characterization of the cracks in the Davis-Besse cladding, the analytical modeling efforts, and the external panel review. A final engineering and analysis report will be issued in conjunction with the report on the preliminary results and findings of the Accident Sequence Precursor (ASP) analysis in early summer.

STATUS OF BL 2005-02 RESPONSES

Eric Weiss
NSIR/DPR/EPD
(301) 415-3264

X3

Response Summary

- All licensees responded within 30 days
- All licensees plan to modify ECLs and EALs
- All licensees plan to notify the NRC of security events promptly
- All licensees plan to address onsite protective action enhancements
- We expect all licensees will adequately address staff augmentation enhancements
- All licensees support improvements in security-based drills and exercises
- Licensees are in the process of implementing enhancements

Emergency Classification Levels and Emergency Action Levels

- All licensees plan to modify ECLs and EALs consistent with Bulletin information
- Licensees plan to use guidance for NUREG-0654, NUMARC/NESP-007, or NEI 99-01 ECLs/EALs, as applicable to their current or proposed scheme
- All licensees plan to complete ECL/EAL revisions within 180 days, except 1 (210 days)

NRC Notification

- All licensees plan to modify procedures to provide prompt notification to NRC of security events
- 16 sites plan to complete changes within 60 days, 47 sites plan to complete changes within 90 days, 1 has completed changes
  - Bulletin requested "60 days"
- Most sites used a "goal of 15 minutes" in their response.
  - Bulletin did not use the term "goal"

Onsite Protective Actions

- All licensees plan to incorporate enhancements to onsite protective actions
- Many licensees will consider developing a decision making tool to aid control room staff
- All licensees plan to complete revisions within 180 days, except 1 (210 days)

ERO Staff Augmentation

- We expect all licensees to enhance staff augmentation consistent with the Bulletin information
- 19 licensees were contacted to provide clarification to their response
- 1 licensee is not completely consistent with the Bulletin information
  - The licensee is reviewing options to address non-consistent issues
- NRC staff are engaging the licensee, and will provide licensee results in a SECY to the Commission

Security-based Drill and Exercise Program

- All licensees plan to support the inclusion of security-based scenarios into their drill and exercise program
- Most licensees included a condition for FEMA review and acceptance of security-based scenarios in meeting exercise objectives
  - Performance of security-based exercises was contingent on NRC and FEMA endorsing changes to a new evaluated exercise program

Actions Forward

- NRC staff preparation of a SECY "Status of Emergency Preparedness Directorate Activities in the Post 9/11 Environment"
- NRC staff is interacting with licensees not providing consistent response to Bulletin
- Future report to the Commission with recommendations

(14)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
526th MEETING
LICENSEE RESPONSES TO EMERGENCY PREPAREDNESS BULLETIN
Rockville, MD
October 7, 2005

- PROPOSED SCHEDULE -

	Topic	Presenter(s)	Time
I	Opening Remarks (Open session)	M. Bonaca, ACRS	8:35-8:45am
II	Summary of Licensee Responses to Bulletin on EP for Security-Related Events (Open session)	E. Weiss, NSIR G. Casto, NSIR	8:45-9:45am
III	Discussion and Closing Remarks (Closed session if necessary)	M. Bonaca, ACRS	9:45-10:00am
	Break		10:00am

Notes:
- Presentation time should not exceed 50% of the total time allocated for a specific item.
- Number of copies of presentation materials to be provided to the ACRS - 35.

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REGULATORY GUIDE 1.82
REVISION 4

Discussion of ACRS
September 20, 2005 Letter to the EDO

Dr. Brian Sheron
Associate Director for Project Licensing and Technical Analysis

October 7, 2005

SEPTEMBER 20 ACRS LETTER

- Rev. 4 to RG 1.82 should not be issued for public comment at this time
- Revise guidance describing what factors to consider in conservatively calculating containment overpressure to improve its clarity:
  - Section 1.3.1 (PWR)
  - Section 2.1.1 (BWR)

SEPTEMBER 20 ACRS LETTER

- Containment overpressure credit to ensure sufficient NPSH for ECCS and containment heat removal pumps should only be "selectively" granted:
  - Demonstrate no practical alternative
  - Only grant for robust containments with positive indication of containment integrity
  - Limited to a "few" hours

STAFF OBSERVATIONS

- "No practical alternative" criterion was developed during the resolution of the BWR sump issue in the mid 1990s
- Necessary for existing BWR designs to demonstrate compliance with 10 CFR 50.46
- Criterion addressed the immediate need, but was not the preferred regulatory approach

STAFF OBSERVATIONS

- Staff has approved numerous requests from both BWRs and PWRs for containment accident pressure credit
- Recent power uprates have prompted the staff to re-examine the issue and develop a consistent regulatory approach for allowing credit for containment accident pressure

HISTORY

- 25 plants credit some amount of containment accident pressure
- December 1997 ACRS letter agreed with containment accident pressure credit; "…should consider a broad range of accident sequences such as typically found in a PRA…"
- Dresden, Quad Cities, Duane Arnold, and Brunswick extended power uprates, which credited containment accident pressure, received favorable ACRS letters

RISK-INFORMED APPROACH

- Previous staff interactions with ACRS may not have sufficiently addressed the staff's risk-informed approach to this issue
- Risk-informed approach relies on a demonstration by the licensee that the five key principles of risk-informed decisionmaking in RG 1.174 are met when credit for containment accident pressure is requested

RISK-INFORMED APPROACH

- Five key principles from RG 1.174:
  - Meets current regulations
  - Consistent with defense-in-depth philosophy
  - Maintains sufficient safety margins
  - Increases in CDF or risk should be small and consistent with the Commission's Safety Goal Policy Statement
  - Impact should be monitored using performance measurement strategies

RISK-INFORMED APPROACH

- Staff is revising RG 1.82 to clearly describe the elements of a risk-informed approach for crediting containment accident pressure:
  - Defense-in-depth: Licensee to show that, under realistic conditions, credit is either not needed or only needed for relatively short time
  - Safety margins: When credit is requested, analysis should be conservatively calculated
  - Small risk increase: License must submit PRA results to demonstrate they meet RG 1.174 numerical risk acceptance guidelines
  - Performance monitoring strategies: Licensee must describe program to ensure containment integrity

NEXT STEPS

- The staff is revising appropriate sections of RG 1.82 to clarify requirements and describe licensee expectations for submitting risk-informed license amendments to credit containment accident pressure
- The staff will provide ACRS with revised RG 1.82

NEXT STEPS

- Staff will request that the ACRS reconsider its position on this issue in light of the staff's approach to use a risk-informed approach to crediting containment accident pressure
- Staff will work with industry to explore options to develop realistically conservative NPSH calculations

CONCLUSIONS

- Staff believes using a risk-informed approach to determine when credit for containment accident pressure can be given is consistent with Commission policy to risk-inform our regulatory process
- Staff intends to use this approach for future license amendments that propose to credit containment accident pressure

(16)

October 5, 2005 (8:23am)
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INTERNAL USE ONLY

SCHEDULE AND OUTLINE FOR DISCUSSION
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
October 5, 2005

1) 10:00 - 10:15 a.m. Review of the Member Assignments and Priorities for ACRS Reports and Letters for the October ACRS meeting (JTL/SD)

Member assignments and priorities for ACRS reports and letters for the October ACRS meeting are attached (pp. 7-8). Reports and letters that would benefit from additional consideration at a future ACRS meeting will be discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the October ACRS meeting be as shown in the attachment (pp. 7-8).

2) 10:15 - 10:35 a.m. Anticipated Workload for ACRS Members (JTL/SD)

The anticipated workload for ACRS members through December 2005 is attached (pp. 9-10). The objectives are to:

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

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

RECOMMENDATION

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

3) 10:35 - 10:45 a.m. Meeting with the NRC Commissioners (JTL/AT)

The ACRS is scheduled to meet with the NRC Commissioners between 1:00 and 3:00 p.m. on Thursday, December 8, 2005 to discuss items of mutual interest. [**NOTE:** This meeting was previously scheduled to be held between 1:30 and 3:30 p.m.] A list of topics noted below, approved by the Committee during its September 2005 meeting, has been sent to the Office of SECY on September 15, 2005, requesting that the Commissioners select a maximum of four topics and provide feedback on the proposed topics by September 30, 2005.

- Policy Issues Related to New Plant Licensing
- Risk-Informed Alternatives to the Single Failure Criterion
- Early Site Permits
- Proposed Alternative Embrittlement Criteria
- Digital Instrumentation and Control System Research Plan
- Fire Protection Matters

On October 4, 2005, the Commission has approved the following topics:

- Issues Related to New Plant Licensing (including technology Neutral Framework) (TSK/MME)
- Proposed Alternative Embrittlement Criteria in 10 CFR 50.46 (DAP/RC)
- Fire Protection Matters (GEA/JGL)
- Power Uprate Technical Issues (RSD/RC)

In addition to the above topics, the ACRS Chairman will provide an overview. Proposed topics for the overview include:

- License renewal
- Early site permits
- Future ACRS activities

RECOMMENDATION

The Subcommittee recommends that the cognizant members and the ACRS staff prepare draft slides for consideration and approval at the November ACRS meeting. The slides should be finalized at the November meeting in order to have them transmitted to the Commission on November 25, 2005.

4) 10:45 - 10:55 a.m. Proposed ACRS Meeting Dates fo CY 2006 (JTL/SD)

Proposed ACRS meeting dates for CY 2006, which are summarized below and also included in the attached Calendar (pp 13-24), were provided to the members during the September meeting, requesting comments by September 23, 2005.

The members agree with the proposed meeting dates. However, Dr. Bonaca suggests that the July meeting be held on July 6-8, or 5-7, instead of July 12-14.

Meeting No.	Meeting Dates
-	January 2006 (No Meeting)
529	February 9-11, 2006
530	March 9-11, 2006
531	April 6-8, 2006
532	May 4-6, 2006
533	May 31 - June 1-2, 2006 *
534	July 12-14, 2006 *
-	August (No Meeting)
535	September 7-9, 2006
536	October 4-6, 2006 *
537	November 1-3, 2006 *
538	December 7-9, 2006

* Wednesday - Friday

RECOMMENDATION

The Subcommittee recommends that the Committee decide on the July meeting dates, taking into account Dr. Bonaca's suggestion, and approve ACRS meeting dates for CY 2006.

5) 10:55 - 11:05 a.m. ACRS Retreat in 2006 (JTL)

During the September 2005 meeting, the Committee decided to hold a retreat on January 26-27, 2006. The members were requested to propose topics for the retreat by September 23, 2005. Drs. Powers and Apostolakis, Mr. Sieber and Dr. Ransom provided their views (pp. 25-31).

RECOMMENDATION

The ACRS Executive Director supports having a retreat in January 2006. It will be valuable to discuss a number of issues related to Committee operations. These issues include:

(i) How should the ACRS handle any significant workload increase in FY 06 and 07? Adding one or two additional meetings does not seem practical as a few members already approach the maximum allowed 130 days per year now. The

Committee should look at various options, including expanding the ACRS to fifteen (15) members and adding a new subcommittee or adding more Saturday sessions.

(ii) Each Subcommittee Chairman should take a few minutes and talk about their forecast of future work for the coming year and whether or not they foresee any emerging issues of significance.

(iii) The Committee should take some time and discuss what technical expertise is needed on the ACRS in the future. Also the ACRS should be more proactive in the search for future members and find ways to have these individuals auditioned prior to recommending for membership on the ACRS. Maybe their should be a standing Subcommittee for potential new ACRS members.

(iv) The Committee should take some time to discuss the upcoming Quadripartite Meeting in October '06, including the presentations and planned events.

The Subcommittee recommends that the Committee discuss the issues raised by Drs. Powers and Apostolakis and decide on a course of action.

6) 11:05 - 11:25 a.m. Staff Requirements Memorandum on Policy Issues Related to New Plant Licensing (JTL/MME)

In a Staff Requirements Memorandum (SRM) dated September 14, 2005, the Commission states that the ACRS should provide its views on the two policy issues (SECY-05-0130) related to new plant licensing, including the feasibility of alternatives to the QHOs as technology-neutral risk objectives. The staff should then consider ACRS comments in developing a subsequent notation vote paper addressing these policy issues.

The Committee issued a report to the Commission on these policy issues in September 2005. However, the Committee did not explicitly address the issues raised by the Commission in the SRM.

RECOMMENDATION

The Subcommittee recommends that Dr. Kress propose a course of action for addressing the issues raised by the Commission

7) 11:25 - 11:35 a.m. Candidates for Potential Membership on the ACRS (JTL) (Closed)

On June 28, 2005, the Screening Panel met to discuss 42 applications received in response to the solicitation for the current vacancies on the ACRS. The Panel selected six applicants in the areas of plant operations and materials and metallurgy. These candidates were interviewed by the members and the Screening

Panel during the September ACRS meeting. The Panel is in the process of preparing a report to the Commission recommending a slate of candidates to fill the vacancy in the area of materials and metallurgy. The Screening Panel will continue to look for qualified candidates to fill the vacancies on the Committee in the areas of thermal-hydraulics and plant operations.

RECOMMENDATION

The Planning and Procedures Subcommittee recommends that the ACRS Executive Director keep the Committee informed of further developments on this matter.

8) 11:35 - 11:45 a.m. Response to Ms. Nancy Burton, Connecticut Coalition Against Millstone Regarding Millstone Units 2 and 3 License Renewal Application (JTL/CS)

In a letter to Dr. Wallis, ACRS Chairman, dated September 7, 2005, Ms. Nancy Burton, Connecticut Coalition Against Millstone, requested that the ACRS defer its decision regarding the Millstone license renewal application until after the State of Connecticut has had an opportunity to provide its input. She also made statements during the Plant License Renewal Subcommittee meeting on April 6, 2005 and the full Committee meeting on September 8, 2005. In her letter, she listed several issues that were not addressed in the staff's final SER (FSER) related to Millstone Units 2 and 3 license renewal application.

It has been the Committee's practice to respond to individuals who sent letters to the ACRS Chairman. During the September meeting, Mr. Santos, ACRS staff engineer, informed the Committee about sending a response to Ms. Burton. A draft response to Ms. Burton will be distributed during the meeting. Since she raises other issues related to the adequacy of the staff's FSER, it will be appropriate to refer those issues to the EDO for possible action. A draft memo to the EDO will be provided during the meeting. Mr. Sieber provided his views with regard to the nature of the response that should be sent to Ms. Burton.

RECOMMENDATION

The Subcommittee recommends that the Committee decide on the nature of the response that should be sent to Ms. Burton and also on the need for referring the other issues raised by Ms. Burton to the EDO.

9) 11:45 - 11:55 a.m. Summary Matrix of ACRS Reports and Letters (JTL)

As directed by the Commission, we need to submit a summary matrix of ACRS reports and letters issued in FY 05 along with the Operating Plan. The Operating Plan and the summary matrix are due to the Commission on December 30, 2005. In order to preclude violation of the ACRS Bylaws, the Committee should authorize the ACRS Executive Director and/or his designee to summarize the ACRS reports and letters issued in FY 05.

RECOMMENDATION

The Subcommittee recommends that the Committee authorize the ACRS Executive Director and/or his designee to summarize the ACRS reports and letters issued during FY 05.

10) 11:50 - 11:55 a.m. Member Issues (JTL)

NRR Office Instruction on Risk-Informed Review Process for Emergent Issues (JTL/JHF)

NRR has recently issued for trial use an internal Office instruction on risk-informed review process for emergent issues. This process was developed in response to the GAO recommendations included in its May 2004 report on NRC's handling of reactor vessel head corrosion at Davis-Besse. In its report, GAO stated that NRC should improve its use of PRA estimates in decisionmaking by:

- Ensuring that the risk estimates, uncertainties, and assumptions made in developing the estimates are fully defined, documented, and communicated to NRC decisionmakers

- Providing guidance to decisionmakers on how to consider the relative importance, validity, and reliability of quantitative risk estimates in conjunction with other quantitative safety-related factors.

Drs. Apostolakis and Denning would like to know more about this NRR process. As tasked by Dr. Larkins, Dr. Flack will provide a brief presentation to the Subcommittee and the full Committee during their October meetings.

RECOMMENDATION

The Subcommittee recommends that Drs. Apostolakis and Denning propose a course of action after hearing Dr. Flack's presentation.

ANTICIPATED WORKLOAD
OCTOBER 6-8, 2005

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis		Lamb	Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" **[STATUS REPORT]**	—	—	—
Bonaca	—	Santos/Lamb	Interim Review of the License Renewal Application for Browns Ferry Units 1, 2, and 3	A	To provide ACRS views	—
		Thornsbury/Savio	Licensees' Responses to Bulletin on Emergency Preparedness and Response Actions for Security-Based Events	—	—	—
Powers	All Members	Nourbakhsh/ Duraiswamy	Draft ACRS Report to the Commission on the NRC Safety Research Program	Report to be finalized in December	To respond to SRM. Due date March 15, 2006	—
	Apostolakis/ Sieber/ Ransom	Nourbakhsh/ Thornsbury/Santos/ Caruso	Results of the Quality Assessment of the NRC Research Projects (SPAR Models, SG Tube Integrity Program at ANL, Thermal-Hydraulic Testing at Penn State)	A	To support staff schedule	—
Ransom	—	Caruso	NRC Staff's Response to the ACRS Letter on the Proposed Revision 4 to Reg. Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a LOCA"	—	—	—

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ANTICIPATED WORKLOAD
OCTOBER 6-8, 2005 (Cont'd)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Sieber	—	Lamb	Proposed Recommendations for Resolving GSI-80, "Pipe Break Effects on CRD Hydraulic Lines in the Drywells of Mark I and II Containments"	A	To support staff schedule	—
	—	Lamb	Davis-Besse Reactor Pressure Vessel Integrity Calculations **[Information Briefing]**	—	—	—
	Bonaca	Lamb/Santos	Subcommittee Report-Browns Ferry Unit 1 Restart Activities (Subc. Mtg 9/21/05)	—	—	—

ANTICIPATED WORKLOAD
NOVEMBER 3-5, 2005

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Thornsbury	Digital I&C Research Plan and Related Matters	A	To provide ACRS views	—
Bonaca	—	Santos	Final Review of the License Renewal Application for Point Beach Nuclear Plant	A	To support staff schedule	—
Denning	—	Lamb	Draft Final Rule on Post-Fire Operator Manual Actions	A	To support staff schedule	—
Kress	—	El-Zeftawy	General Description of ESBWR Design/NRC Staff's Review Schedule **[INFORMATION BRIEFING]**	—	—	—
Powers	All Members	Nourbakhsh/ Duraiswamy	Draft ACRS Report to the Commission on the NRC Safety Research Program	Report to be finalized in December	To respond to SRM. Due date March 15, 2006	—
Sieber	—	Lamb	Draft Final Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and Operability of Offsite Power"	A	To support staff schedule	—
Wallis	All Members	Larkins/Thadani/Scott	Preparation for Meeting with the Commissioners, Dec. 8, 2005	—	—	—

ANTICIPATED WORKLOAD
DECEMBER 8-10, 2005

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	Apostolakis	Flack	Staff's Response to SRM (SECY-04-0111) Regarding Staff Actions on Agency Guidance in the Areas of Safety Conscious Work Environment and Safety Culture **[INFORMATION BRIEFING]**	—	—	—
Denning	—	Lamb	Draft Final Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance With Fire Protection Regulations"	A	To support staff schedule	—
	Wallis	Caruso	Final Review of the Vermont Yankee Power Uprate Application and the Final SER	A	To support staff schedule	—
Powers	—	El-Zeftawy	Final Review of the Grand Gulf Early Site Permit Application and the Final SER	A	To support staff schedule	—
	All Members	Nourbakhsh/ Duraiswamy	NRC Safety Research Program Report	A	To respond to SRM. Due date March 15, 2005	—
Wallis	All Members	Larkins/Thadani/ Scott	Meeting with the NRC Commissioners [1:00 - 3:00, December 8, 2005]	—	—	—

ACRS Items Requiring Committee Action

1 **Review of FERRET Reactor Vessel Fluence Methodology**

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

Estimated Time: 2 hours

Purpose: Possible Review & Comment

Priority: High

Requested by: NRR L. Lois

Westinghouse has submitted a topical report describing the FERRET methodology, which is used to predict the fluence on the reactor vessel wall due to neutron leakage from the core. This value is used in evaluating the embrittlement of the reactor vessel. The staff issued its safety evaluation on August 30, 2005, and the Committee should determine whether it wants to review this topical report and SER.

2 **Review of Proposed Rulemaking on Safety/Security Interface**

Member: Mario Bonaca **Engineer:** Eric Thornsbury

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR J. Birmingham, NRR

The staff is proposing to add a new section in 10 CFR Part 73, requiring holders of licenses for operating nuclear power plants to assess changes to the facility or to the implementation of the security plan for potential adverse interaction by taking appropriate compensatory or mitigative measures commensurate with existing requirements.

The draft proposed rule resulted, in part, from a Petition for Rulemaking, (PRM-50-80), dated April 28, 2003, from the Union of Concerned Scientists and the San Luis Obispo Mothers for Peace. Part of the petition requested that the regulations establishing conditions of licenses and requirements for evaluating proposed changes, tests, and experiments for nuclear power plants be amended to require licensee evaluation of the effect of the proposed change, test, or experiment caused protection against radiological sabotage to be decreased. The NRC Petition Review Board (PRB) evaluated the petition and decided that existing regulations do not explicitly require an evaluation of changes to the facility or to implementation of the security plan for potential adverse interaction prior to implementing those changes and therefore this part of the petitioners request should be considered for rulemaking.

The staff is requesting that the Committee defer its review of the proposed rule until the final rulemaking package has been prepared. The proposed rule is scheduled to be submitted to the Commission in February 2006.

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