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

December 10, 2004

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

SUBJECT: SUMMARY REPORT - 517th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, NOVEMBER 4-6, 2004 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

Dear Chairman Diaz:

During its 517th meeting, November 4-6, 2004, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letters, and memoranda:

REPORTS:

Reports to Nils J. Diaz, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Draft Proposed Rule on Post-Fire Manual Operator Actions, dated November 19, 2004
- Report on "An Overview of Differences in Nuclear Safety Regulatory Approaches and Requirements Between United States and Other Countries," dated November 2, 2004

LETTERS:

Letters to Luis A. Reyes, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Lessons Learned from the ACRS Review of the AP1000 Design dated November 18, 2004
- Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria," dated November 17, 2004

MEMORANDA:

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

- Draft Regulatory Guide 1127, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," dated November 8, 2004
- Proposed Rule: Fitness for Duty (FFD) Programs, 10 CFR Part 26, dated November 8, 2004

OTHER

Letter to Carl Paperiello, Director, Office of Nuclear Regulatory Research, from Mario V. Bonaca, Chairman, ACRS, Subject: ACRS Assessment of the Quality of Selected NRC Research Projects, dated November 18, 2004

HIGHLIGHTS OF KEY ISSUES

1. Proposed Rule Language for Risk-Informing 10 CFR 50.46

The Committee met with representatives of the NRC staff to review proposed rule language for a risk-informed alternative to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." The briefing focused on the proposed rule language and conforming changes, ECCS analysis requirements, and the process for approval of plant changes based upon the new design basis LOCA.

Committee Action:

The Committee decided to review and comment on the proposed rule in December 2004 after the NRC staff completes the associated Statement of Considerations and Regulatory Analysis.

2. Proactive Materials Degradation Assessment Program

The Committee heard presentations by and held discussions with the industry and the staff of the Office of Research (RES) regarding the status of the proactive materials degradation assessment program.

The NRC presentation was made by Joe Muscara, RES. The Industry presentations were made by Robin Dyle, Southern Nuclear Corporation for NEI and Robin Jones, EPRI.

The purpose of the presentations was to provide an update of the plans to develop programs that address proactively environmentally–assisted materials degradation in light-water reactors.

Materials degradation phenomena will continue to be an important factor in future operational safety issues. The approaches taken by the NRC and industry in developing capabilities to manage the materials degradation issues differ in timing and focus, but are complementary in achieving the same overall objective.

RES indicated that the objective of the NRC program is to provide a foundation for regulatory actions to help prevent materials degradation from adversely impacting safety. The work is being done in two phases. The first phase is to identify susceptible materials and locations where degradation can be expected in LWRs in the future. The second phase is to develop and implement an international cooperative program for the components and degradation of interest.

The industry described the policy and practices that they will follow in managing materials aging issues in NEI 03-08, "Guideline for the Management of Materials Issues." They gave a description of the technical approaches that will be taken to meet defined objectives and how these approaches will affect the year-by-year prioritization of materials degradation issues and the development of adequate mitigation and inspection strategies.

Committee Action

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

3. Proposed Rule on Post-Fire Operator Manual Actions

The Committee met with representatives of the NRC staff, the industry, and general public to review the proposed rule on post-fire operator manual actions. The staff discussed the development of the rule and key provisions that were included to ensure adequate safety. If promulgated, the proposed rule would codify the use of certain operator manual actions to satisfy the requirements of Appendix R to 10 CFR 50. To obtain credit for the use of operator manual actions under this rule, licensees would have to show that the manual actions meet certain acceptance criteria and the affected fire areas must be equipped with fixed detection and automatic suppression systems. The staff stated that this proposed rule should result in significant cost reductions for the NRC and licensees.

Representatives of the nuclear industry and the general public expressed differing opinions regarding the effectiveness of the proposed rule change. The industry stated the proposed rule did not provide enough flexibility and few licensees would benefit from this rulemaking. The members of the public questioned the NRC's motives for pursuing the rulemaking and expressed concern that the staff was attempting to replace the clear and crisp language currently included in Appendix R with vague and unenforceable guidance. Other members of the public were concerned that the staff was sending the wrong message to those licensees that made the necessary investments in their plants to fully comply with Appendix R to 10 CFR 50.

Committee Action

The Committee issued a report to the NRC Chairman on November 19, 2004, recommending that the proposed rule be published in the Federal Register for public comment. The ACRS report was accompanied by additional comments from an individual ACRS member, Mr. Stephen Rosen. In his comments, Mr. Rosen suggested that the Commission pursue a more risk-informed and performance-based approach to the control of fire risk. He also suggested that the rule would be more effective if the requirement for automatic suppression in the area of the postulated fire was removed.

4. Grid Reliability Issues and Related Significant Operating Events

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding grid reliability and offsite power issues.

The NRC staff presentation was made by John Lamb and Tom Koshy of NRR, and Dale Rasmuson and Bill Raughley of RES. The purpose of the presentations was to hear the staff's views of issues associated with grid reliability and offsite power issues.

The NRC staff identified 48 concerns with the reliability of offsite power to nuclear power plants that needed to be resolved. These concerns are divided into three groups. Group One contains 10 concerns that the staff has determined need to be addressed in the short-term. Group Two has 21 concerns that are beyond NRC's statutory authority. They are within the purview of the Federal Energy Regulatory Commission (FERC) and the North American Electric Reliability Council (NERC). Group Three has 17 remaining concerns not addressed by the other two approaches.

A three pronged approach was developed for the group one concerns. First, the staff raised awareness of the concerns by developing and issuing a Regulatory Issues Summary. Secondly, the staff assessed the licensees' readiness to manage any degraded or losses of offsite power through inspections and interviews. Lastly, the staff monitored and reviewed conditions and events through the summer of 2004 and determined that the operational readiness of offsite power systems for nuclear power plants would be assured.

Group three concerns were consolidated into four topical areas: (1) Offsite Power System Availability, (2) Station Blackout Review, (3) Risk Insights, and (4) Interactions with External Stakeholders. In the Offsite Power System Availability and the Station Blackout Review topical areas, the staff is considering a generic communication. In the Risk Insights topical area, the staff is reviewing the RES report, "Loss of Offsite Power Frequency and Duration Analyses." In the Risk Insights topical area, the staff will also review RES report "Loss of Offsite Power Risk Analyses and Emergency Diesel Generator Reliability. In the Risk Insights Topical area, RES is working on, (1) "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1986 – 2003," and (2) "Status of the Investigation of Grid Operating Data for Signs of Change and Potential Vulnerabilities."

In the Interactions with External Stakeholders topical area, the NRC staff signed Memoranda of Agreement (MOA) between the NRC and NERC and the NRC and FERC on August 27, and September 1, 2004, respectively.

Committee Action

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

5. Status of Early Site Permit Reviews

The Committee heard presentations by the NRC staff regarding the status of the review of the early site permit (ESP) applications. The NRC received three ESP applications in September and October 2003 from Dominion Nuclear North Anna, LLC (Dominion), for the North Anna ESP site; Exelon Generation Company, LLC (Exelon), for the Clinton ESP site; and System Energy Resources, Inc. (SERI), for the Grand Gulf ESP site.

The NRC staff has developed a review standard for ESP applications to provide guidance to the NRC staff on the process and criteria for reviewing an ESP application. Due to the absence of reactor types to be built on an ESP site, the applicants are proposing to use a plant parameters envelope (PPE) approach. The PPE is a set of design parameters that are expected to bound the characteristics of a reactor or reactors that might be deployed at a site. Currently, the NRC staff plans to issue draft safety evaluation reports (SERs) for Dominion, Exelon, and SERI on their ESP applications in late December 2004, February 2005, and April 2005, respectively.

Issues that have arisen from the NRC staff's review include tornado wind speed, seismic analysis, and emergency planning. The NRC staff plans to compile lessons learned for future reviews.

Committee Action

This briefing was provided for information only. The Committee plans to review the draft SERs when they are available.

6. Assessment of the Quality of Selected NRC Research Projects

The Office of Nuclear Regulatory Research (RES) is required to have an independent evaluation of the quality of its research programs. This evaluation is mandated by the Government Performance and Results Act (GPRA) and needs to be in place during the next fiscal year. The Committee has agreed to assist RES in assessing the effectiveness and utility of the NRC research programs. The Committee has previously approved the strategy for the review of the quality of selected research projects. This strategy is to be tried during FY 2004 and refined in FY 2005. During the November 2-4, 2004 ACRS meeting, the Committee discussed its report on Assessment of the Quality of Selected NRC Research Projects.

Committee Action

The Committee approved the letter transmitting the report on Assessment of the Quality of Selected NRC Research Projects to the Director of RES. The Committee is now poised to undertake review of four additional research projects during fiscal year 2005.

7. Plant License Renewal Subcommittee Report

The Chairman of the Plant License Renewal Subcommittee provided a report to the Committee summarizing the results of the November 3, 2004, subcommittee meeting with the NRC staff and representatives of the Southern Nuclear Operating Company (SNC) to review the NRC staff's Draft Safety Evaluation Report (SER) related to the License Renewal Application for the Farley Units 1 and 2. The current operating licenses for Units 1 and 2 expire on June 25, 2017, and March 31, 2021, respectively. The applicant has requested approval for continued operation of each unit for a period of 20 years beyond the current license expiration dates. During the meeting SNC described Farley's operating history, major equipment replacements, their application of the GALL report, and other industry issues. The staff issued fewer request for additional information requests during the review of this application because of the new review process and on-site audits. The staff has determined that there are no open or confirmatory items regarding this application. The Draft SER listed 3 license conditions and concluded that the license renewal application meets the requirements of 10 CFR Part 54.

Committee Action

The Committee will review the final SER and hold discussions with the staff and applicant during the April 2005 ACRS meeting.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the response from the EDO dated October 7, 2004, to the ACRS report dated July 19, 2004, concerning the Proposed Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at PWRs. The Committee decided that it was satisfied with the EDO's response.

  The staff committed to continue confirmatory research consistent with ACRS recommendations. The staff also committed to keeping the ACRS informed of the staff's activities relative to resolving GSI-191.

- The Committee considered the response from the EDO dated October 26, 2004, to the ACRS report dated September 16, 2004, concerning the Report on the Safety Aspects of the License Renewal Application for the Dresden 2 and 3 and Quad Cities 1 and 2 Nuclear Power Stations. The Committee decided that it was satisfied with the EDO's response.

  The staff committed to further clarify that Extended Power Uprate (EPU) applications are reviewed for aging effects for their current license operating period. The staff also committed to discuss the status of improved guidance regarding the synergistic effects of plant operations at EPU levels on plant structures and components.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from October 7, 2004 through November 3, 2004, the following Subcommittee meetings were held:

- Fire Protection - October 27, 2004

The Subcommittee heard presentations from the staff on the proposed Fire Protection Rulemaking to allow for the use of operator manual actions in lieu of installed fire barriers. The staff intends to provide the proposed rule to the Commission December 2004. If accepted by the Commission, the proposed rule will be released for public comment.

- Regulatory Policies and Practices - October 28-29, 2004

The Subcommittee reviewed the proposed rule package for risk-informing 50.46.

- Plant License Renewal - Joseph M. Farley Nuclear Station - November 3, 2004

The Subcommittee reviewed the License Renewal Application and associated draft SER for the Joseph M. Farley Nuclear Station.

- Planning and Procedures - November 3, 2004

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 plans to review the proposed rule for risk-informing 10 CFR 50.46 during its December 2004 meeting.

- The Committee plans to review the draft final 10 CFR Part 26 on drug testing and worker fatigue.

- The Committee plans to review the license renewal application for the Farley Nuclear Plant and the associated staff's final SER in April 2005.

- The Committee plans to review the draft final rule on post-fire operator manual actions after reconciliation of public comments.

- The Committee plans to review the staff's draft SERs on early site permit applications when available.

PROPOSED SCHEDULE FOR THE 518th ACRS MEETING

The Committee agreed to consider the following topics during the 518th ACRS meeting, to be held on December 2-4, 2004:

- Expert Elicitation on Large-Break LOCA Frequencies
- Proposed Rule for Risk-Informing 10 CFR 50.46
- Technical Basis for Potential Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule
- Draft Commission Paper on Technology Neutral Framework for Future Plant Licensing - Policy Issues

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

Date Issued: 12/15/2004
Date Certified: 12/23/2004

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NOVEMBER 4-6, 2004

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APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
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CERTIFIED

MINUTES OF THE 517th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NOVEMBER 4-6, 2004
ROCKVILLE, MARYLAND

The 517th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on November 4-6 2004. Notice of this meeting was published in the *Federal Register* on October 26, 2004 (65 FR 62466) (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. Mario V. Bonaca (Chairman), Dr. Graham B. Wallis (Vice Chairman), Mr. Stephen L. Rosen, (Member-at-Large), Dr. George E. Apostolakis, Dr. F. Peter Ford, Dr. Thomas S. Kress, Dr. Victor H. Ransom, Dr. William J. Shack, and Mr. John D. Sieber. Dr. Dana A. Powers 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. Mario V. Bonaca, 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. Proposed Rule for Risk-Informing 10 CFR 50.46 (Open)

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

The Committee met with representatives of the NRC staff to review proposed rule language for a risk-informed alternative to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." The briefing focused on the proposed rule language and conforming changes, ECCS analysis requirements, and the process for approval of plant changes based upon the new design basis LOCA.

Committee Action:

The Committee decided to review and comment on the proposed rule in December 2004 after the NRC staff completes the associated Statement of Considerations and Regulatory Analysis.

III. Proactive Materials Degradation Assessment Program (Open)

[Note: Mrs. Maggalean W. Weston was the Designated Federal Official for this portion of the meeting.]

Mr. John D. Sieber, Chairman of the Plant Operations Subcommittee introduced this topic to the Committee. The Committee heard presentations by and held discussions with representatives of the Nuclear Energy Institute (NEI), the Electric Power Research Institute (EPRI), and the staff of the Office of Research (RES) regarding the status of the proactive materials degradation assessment program.

Industry and NRC Staff Presentations

The NRC presentation on the materials degradation assessment was made by Joe Muscara, RES. The Industry presentations were made by Robin Dyle, Southern Nuclear Corporation for NEI and Robin Jones, EPRI.

The purpose of the presentations was to provide an update of the plans to develop programs that address proactively environmentally–assisted materials degradation in Light Water Reactors.

Examples of such degradation phenomena include:

- Stress corrosion cracking of PWR steam generator tubing.
- Intergranular stress corrosion cracking of BWR piping.
- Irradiation-assisted cracking of stainless steel core components in BWRs.
- Stress corrosion cracking of nickel-base alloy primary piping in PWRs.
- Stress corrosion cracking of nickel-base alloy vessel penetrations in PWRs.

- Flow accelerated corrosion of carbon-steel piping in both PWRs and BWRs.
- Boric acid wastage of low-alloy PWR pressure vessel steel.

Materials degradation phenomena will continue to be an important factor in future operational safety issues. Both the NRC and industry have instituted programs to manage the materials degradation issues so that they are predicted and inspection and/or mitigating actions are developed well before the problems present a significant safety or financial burden. The approaches taken by the NRC and industry in developing these capabilities differ in timing and focus, but are complementary in achieving the same overall objective.

RES Approach

The objective of the NRC program is to provide a foundation for regulatory actions to help prevent materials degradation from adversely impacting safety. T he work is being done in two phases. The first phase, which started in August 2004, is to identify susceptible materials and locations where degradation can be expected in LWRs in the future.

The method being used in the first phase is a modified Phenomena Identification and Ranking Table (PIRT) process to identify materials and locations in representative PWR and BWR designs. The PIRT panel members are drawn from the USA, Japan, and France, are qualified in a wide range of relevant disciplines, and are well-versed in the operating history, practicalities and theories related to materials degradation issues. The panel will examine the operating history of plant components and individually evaluate the potential for future environmentally-assisted degradation. Components at various degrees of risk of future environmentally-assisted degradation will be identified, and may be prioritized in terms of evaluating the necessary "control" developments associated with inspection, mitigation, etc., that are addressed in the second phase of this program.

The second phase is to develop and implement an international cooperative program for the components and degradation of interest. This cooperative program will address:

- Materials and degradation mechanisms
- Mitigation
- Repair and replacement
- Non-destructive evaluation

Industry Approach

In May, 2003, the industry described in NEI 03-08, "Guideline for the Management of Materials Issues," the policy and practices that will be followed in managing materials aging issues. These include:

- A process for managing the resources (currently $55M/year) expended by the industry on materials degradation issues.
- Prioritization of the materials issues.
- Development of proactive approaches for materials issues.
- Oversight of implementation of mitigation and inspection strategies.

During this meeting a description was given of the detailed technical approaches taken by EPRI. The outlined technical approaches will meet the above objectives and will affect the year-by-year prioritization of materials degradation issues and the development of adequate mitigation and inspection strategies.

The staff and the industry plan to brief the Committee in the future regarding any additional work.

Committee Action

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

IV. Proposed Rule on Post-Fire Operator Manual Actions (Open)

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

The Committee met with representatives of the NRC staff, the industry, and the general public to review the proposed rule on post-fire operator manual actions. The staff discussed the development of the rule and key provisions that were included to ensure adequate safety. If promulgated, the proposed rule would codify the use of certain operator manual actions to satisfy the requirements of Appendix R to 10 CFR 50. To obtain credit for the use of operator manual actions under this rule, licensees would have to show that the manual actions meet certain acceptance criteria and the affected fire areas must be equipped with fixed detection and automatic suppression systems. The staff stated that this proposed rule should result in significant cost reductions for the NRC and licensees.

Representatives of the nuclear industry and the general public expressed differing opinions regarding the effectiveness of the proposed rule change. The industry stated the proposed rule did not provide enough flexibility and few licensees would benefit from this rulemaking. The members of the public questioned the NRC's motives for pursuing the rulemaking and expressed concern that the staff was attempting to replace the clear and crisp language currently included in Appendix R with vague and unenforceable guidance. Other members of the public were concerned that the staff was sending the wrong message to those licensees that made the necessary investments in their plants to fully comply with Appendix R to 10 CFR 50.

Committee Action

The Committee issued a report to the NRC Chairman on November 19, 2004, recommending that the proposed rule be published in the Federal Register for public comment. The ACRS report was accompanied by additional comments from an individual ACRS member, Mr. Stephen Rosen. In his comments, Mr. Rosen suggested that the Commission pursue a more risk-informed and performance-based approach to the control of fire risk. He also suggested that the rule would be more effective if the requirement for automatic suppression in the area of the postulated fire was removed.

V. Grid Reliability Issues and Related Significant Operating Events (Open)

[Note: Mrs. Maggalean W. Weston was the Designated Federal Official for this portion of the meeting.]

Mr. John D. Sieber, Chairman of the Plant Operations Subcommittee introduced this topic to the Committee. The Committee heard presentations by representatives of the NRC staff regarding grid reliability and offsite power issues.

NRC Staff Presentations

The NRC staff presentation was made by John Lamb and Tom Koshy of the Office of Nuclear Reactor Regulation (NRR), and Dale Rasmuson and Bill Raughley of RES.

The purpose of the presentations was hear the staff's views of issues associated with grid reliability and offsite power issues.

Background

In 1998 and 1999, the NRC staff evaluated the impact of deregulation on the reliability of the electric grid. This evaluation led to recommendations to confirm the licensing basis of the nuclear power plants and to reevaluate the underfrequency protection trip settings.

In 2000, the NRC asked NEI and other industry representatives to address the adequacy of reliable offsite power to nuclear power plants. As a result, the Institute of Nuclear Power Operations (INPO) issued a Significant Operating Experience Report on the "Loss of Grid," in December 1999. In that report, INPO called for the establishment of communication protocols between the nuclear power plant operator and the grid operator.

The power blackout event on August 14, 2003, highlighted the fact that the Nation's electric grid is no longer being operated in the manner that it was considered when it was designed and constructed. An unreliable grid cannot ensure the availability of the offsite power system (preferred power supply), which is essential to ensure the safe operation of nuclear power plants (NPPs).

In December 2003, the NRC Chairman directed the NRC Executive Director for Operations (EDO) to conduct a review of the issues raised in a report entitled "State of U.S. Power Grid from a Nuclear Power Plant Perspective." Following deterministic and risk evaluation, it was concluded that it was important to address those significant issues manifested by the August 14, 2003, event.

Discussion

The NRC staff identified 48 concerns with the reliability of offsite power to nuclear power plants that needed to be resolved. These concerns have been divided into three groups. Group One contains 10 concerns that the staff has determined need to be addressed in the short-term. Group Two has 21 concerns which are beyond the statutory authority of the NRC and fall within Federal Energy Regulatory Commission's (FERC's) and the North American Electric Reliability Council's (NERC's) purviews. Group Three has 17 remaining concerns not addressed by the other two approaches.

Group One

To resolve the concerns in Group One, the staff developed a three-pronged approach.

First, the staff raised awareness of the concerns by developing and issuing a Regulatory Issue Summary (RIS) 2004-05, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," highlighting the significance of grid reliability with respect to the operability of the offsite power system for nuclear power plants.

Second, the staff assessed the licensees readiness to manage any degraded or losses of offsite power through inspections and interviews using Temporary Instruction TI 2515/156, "Offsite Power System Operational Readiness."

Lastly, the staff monitored and reviewed conditions and events through the summer of 2004 and determined that the operational readiness of offsite power systems for nuclear power plants would be assured during the summer of 2004.

Group Two

Group Two had 21 concerns which were beyond the statutory authority of the NRC and fall within NERC's and FERC's purview. The NRC staff continues to follow the activities of NERC and FERC.

Group Three

The staff consolidated the concerns in group three, into four topical areas: (1) Offsite Power System Availability, (2) Station Blackout Review, (3) Risk Insights, and (4) Interactions with External Stakeholders.

1. Offsite Power System Availability

The issues in this area concern (1) offsite power stability, timing, and reliability, (2) communication protocols between the nuclear power plant operator and its transmission system operator, (3) technical specifications limiting conditions for plant operation related to the operability of offsite power, (4) engineering assessment of offsite power assumptions in accident analyses, and (5) updating the licensing bases for offsite power systems.

2. Station Blackout (SBO) Review

The issues in this area concern (1) the underlying assumptions for assessing nuclear power plant's coping duration and recovery of offsite power, (2) unavailability of emergency diesel generators (EDGs), and (3) calculation of SBO risks with updated Standardized Plant Analysis Risk (SPAR) models.

3. Risk Insights

The issues in this area primarily relate to (1) loss of offsite power (LOOP) probability, (2) allowed outage time extension for online EDG maintenance, (3) risk assessment of offsite power assumptions in accident analyses, (4) maintenance risk assessment before and during switchyard work, and (5) assessment of cumulative risk impacts of combined LOOP events at multiple units and sites. In addition, this topical area encompasses the effort to predict the likelihood of future blackout events using grid operational data obtained from the NERC.

4. Interactions with External Stakeholders

The issues in this area concern interactions with external stakeholders to address grid concerns such as (1) nuclear power plant underfrequency trip settings, (2) containment of cascading power blackout, (3) collection of grid operational data, and (4) cyber security.

The following key information is driving the staff actions:

1. Based on NRC inspections and assessments of the results of the audits conducted by NERC, the staff believes effective actions are being taken to enhance the availability of offsite power for safe nuclear power plant operation.

2. Nuclear power plant operators need to be aware of the offsite power needs of the nuclear power plant and know when these needs cannot be met.

3. The NRC staff found considerable variability and uncertainty among licensees regarding the responses to the three key questions in Temporary Instruction (TI) 2515/156, "Offsite Power System Operational Readiness."

   The three key questions from the TI are:

   a. Do agreements between NPP and TSOs specify voltage range and post-trip load requirements?
   b How often is post-trip voltage contingencies calculated?
   c. What notification occurs before a reactor trip would result in inadequate NPP voltage?

4. The cooperation of the transmission system operator may have to be enlisted to ensure that offsite power will be available and switchyard voltages will be adequate following a trip of the plant.

5. Because of the inconsistency in how the industry is addressing the need to ensure the availability of offsite power following a unit trip, a generic communication may be needed to ensure readiness to cope with an event similar to the August 14, 2003, power outage and to ensure that regulatory requirements will continue to be met.

In the Offsite Power System Availability and the Station Blackout Review topical areas, the staff is considering a generic communication.

In the Risk Insights topical area, the staff is reviewing the RES report, "Loss of Offsite Power Frequency and Duration Analyses." In the Risk Insights topical area, the staff will also review the RES report "Loss of Offsite Power Risk Analyses and Emergency Diesel Generator Reliability.

In the Risk Insights Topical area, RES is working on, (1) "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1986 – 2003," and (2) "Status of the Investigation of Grid Operating Data for Signs of Change and Potential Vulnerabilities."

In the Interactions with External Stakeholders topical area, the NRC staff signed a Memoranda of Agreement (MOA) between the NRC and NERC, and the NRC and FERC on August 27, and September 1, 2004, respectively.

<u>Committee Action</u>

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

VI. Status of Early Site Permit Reviews (Open)

[Note: Dr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

Dr. Thomas S. Kress, Future Plant Designs Subcommittee Chairman, stated that the purpose of this meeting was to hear a presentation by the NRC staff regarding the status of the staff's review of the early site permit (ESP) applications. The NRC received three ESP applications in September and October 2003 from Dominion Nuclear North Anna, LLC (Dominion), for the North Anna ESP site; Exelon Generation Company, LLC (Exelon), for the Clinton ESP site; and System Energy Resources, Inc. (SERI), for the Grand Gulf ESP site.

Mr. Michael Scott, NRR, stated that NRC staff has developed a review standard for ESP applications (RS-002, Processing Applications for Early Site Permits) to provide guidance to the NRC staff on the process and criteria for reviewing an ESP application. RS-002 consolidates existing guidance, updates the guidance to reflect the ESP licensing process, and identifies the scope of the ESP review. The standard also informs the stakeholders regarding the information the staff expects to be provided in an ESP application.

Currently, in the absence of reactor types to be built on an ESP site, the applicants are proposing the use of a plant parameters envelope (PPE) approach. The PPE is a set of design parameters that are expected to bound the characteristics of a reactor or reactors that might later be deployed at a site. In terms of the safety reviews, this means that the design characteristics will be no more demanding from a site suitability perspective than the bounding design parameters listed in the PPE tabulation.

As a result of earlier and current efforts, appropriate design parameters have been identified for inclusion in the PPE through a systematic review of regulatory criteria and guidance, ESP application content requirements, and experience with previous site suitability studies. The plant parameters are used to characterize: the functional or operational needs of the plant from the site's natural or environmental resources; the plant's impact on the site and surrounding environs; and the site imposed requirements on the plant. For example, water used for cooling may be from a natural site resources, or plant site environs, and site wind conditions could impose a force on plant structures.

Design parameters are the postulated features of the reactor or reactors that could be built and values are chosen to bound a range of possible future facilities. The PPE values are generally based on certified design information and the best available information for as yet uncertified designs. The site parameters are the postulated physical, environmental, and demographic features of an unspecified site.

The staff plans to issue draft safety evaluation reports for Dominion/North Anna, Exelon/Clinton, SERI/Grand Gulf on December 20, 2004, February 10, 2005, and April 7, 2005, respectively.

During the staff's review, challenging "first-of-a-kind" issues had arisen. These included tornado wind speed, seismic analysis, and emergency planning. All design certifications to date have assumed a wind speed of 300 mph. Currently, the staff is re-evaluating the maximum tornado wind speed based on new data. The staff recommends the development of a risk-informed approach. North Anna and Clinton have proposed a performance-based approach for determining safe-shutdown earthquakes (SSE). The staff has not reviewed the acceptability of the new approach. SSE at rock sites may exceed certified plant design SSEs at high frequencies. The applicants for a combined license or a construction permit will need to be addressed. All three applicants seek acceptance of major features of emergency plans. The industry had concerns regarding the finality associated with such acceptance and the level of detail in the staff's review relating to previously filed information. State and local plans will be reviewed when an applicant seeks approval of major features related to offsite emergency planning. The NRC staff plans to compile lessons learned for future reviews.

Committee Action

This briefing was for information only. The Committee plans to review the draft SERs when they are available.

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

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

The Office of Nuclear Regulatory Research (RES) is required to have an independent evaluation of the quality of its research programs. This evaluation is mandated by the Government Performance and Results Act (GPRA) and needs to be in place during the next fiscal year. The Committee has agreed to assist RES in assessing the effectiveness and utility of the NRC research programs. The Committee has previously approved the strategy for the review of the quality of selected research projects. This strategy is to be tried during FY 2004 and refined in FY 2005. During the November 2-4, 2004 ACRS meeting, the Committee discussed its report on Assessment of the Quality of Selected NRC Research Projects.

Committee Action

The Committee approved the letter transmitting the report on Assessment of the Quality of Selected NRC Research Projects to the Director of RES. The Committee is now poised to undertake review of four additional research projects during fiscal year 2005.

VIII. Plant License Renewal Subcommittee Report (Open)

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

The Chairman of the Plant License Renewal Subcommittee provided a report to the Committee summarizing the results of the November 3, 2004, subcommittee meeting with the NRC staff and representatives of the Southern Nuclear Operating Company to review and discuss the NRC's Draft Safety Evaluation Report (SER) related to the License Renewal Application for the Farley Units 1 and 2. The current operating licenses for Units 1 and 2 expire on June 25, 2017, and March 31, 2021, respectively. The applicant has requested approval for continued operation of each unit for a period of 20 years beyond the current license expiration dates.

The Joseph M. Farley Nuclear Power Station consists of two, 3-loop Westinghouse pressurized water reactor units each rated at 2775 MWt, with a gross electrical output of approximately 910 MWe. In 1998 power was uprated by 123 MWt for each unit. The steam generators for both units have been replaced and the reactor vessel heads for both units will be replaced by 2005. Inspections have been performed at both Farley units in response to the industry events of cracking in bottom mounted instrumentation nozzles and cracking in the VC summer hot leg. No degradation was evident from these inspections.

The staff's review of Farley's license renewal application was the first to implement the consistency with GALL audits. This new review process for the aging management programs and aging management reviews reduced the number of requests for additional information issued compared to previous reviews of other license renewal applications. The staff stated that no open items must be resolved before the staff can make a determination on the application. The draft SER issued on October 15, 2004, listed three license conditions and concluded that the license renewal application meets the requirements of 10 CFR Part 54.

Committee Action

The Committee will review the final SER and hold discussions with the staff and applicant during the April 2005 ACRS meeting.

X. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations/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 these recent ACRS reports:

- The Committee considered the response from the EDO dated October 7, 2004, to the ACRS report dated July 19, 2004, concerning the Proposed Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at PWRs. The Committee decided that it was satisfied with the EDO's response.

  The staff committed to continue confirmatory research consistent with ACRS recommendations. The staff also committed to keeping the ACRS informed of the staff's activities relative to resolving GSI-191.

- The Committee considered the response from the EDO dated October 26, 2004, to the ACRS report dated September 16, 2004, concerning the Report on the Safety Aspects of the License Renewal Application for the Dresden 2 and 3 and Quad Cities 1 and 2 Nuclear Power Stations. The Committee decided that it was satisfied with the EDO's response.

  The staff committed to further clarify that Extended Power Uprate (EPU) applications are reviewed for aging effects for their current license operating period. The staff also committed to discuss the status of improved guidance regarding the synergistic effects of plant operations at EPU levels on plant structures and components.

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 November 3, 2004. The following items were discussed:

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

Member assignments and priorities for ACRS reports and letters for the November 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 February 2005 were addressed. The objectives were:

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

During this session, the Subcommittee also discussed and developed recommendations on items included the Future Activities list.

Expanded Meeting of the Planning and Procedures Subcommittee

During the October 2004 ACRS meeting, the Committee decided to hold an expanded meeting of the Planning and Procedures Subcommittee to discuss certain process and regulatory issues. The Committee agreed to hold this meeting at the ACRS Conference Room on January 27-28, 2005. A list of topics proposed by the Subcommittee and the member assignments are provided below. Additional assignments for the members and the staff will be made by the ACRS Chairman and the ACRS Executive Director.

Process Issues

- Reviewing fewer issues and spend more time on each (TENTATIVE) (GEA)
- Documenting some ACRS members' concerns regarding the quality of science and engineering that goes into regulations more importantly into the regulatory process (DAP)
- Role of the cognizant Subcommittee Chairman during full Committee meetings (MVB)
- Effectiveness of preparing ACRS reports during full Committee meetings (WJS)

Regulatory Issues

- Extended power uprate issues (MVB/GBW/TSK)
  - Use of containment overpressure credit for NPSH calculations - Impact on defense in depth/safety margins
  - Uncertainty associated with determining the adequacy of the NPSH
  - Should there be a limit on containment overpressure credit to be allowed?
  - Adequacy of the risk methodology to allow a risk-informed decision on the percentage or amount of overpressure credit to be allowed.
  - Generic Implications of extended power uprates
  - Issues proposed by Dr. Kress (pp. 14-15)

  Issues raised by Dr. Bonaca and a list of questions and technical issues, associated with power uprates, prepared by the ACRS staff are also attached (pp. 16-20).

- Role of design-basis accident concept in future reactors (DAP)

Election of Officers for CY 2005

During the December 2004 ACRS meeting the Committee will elect a Chairman, a Vice Chairman, and a Member-at-Large for the Planning and Procedures Subcommittee. In accordance with the ACRS Bylaws, those members who do not wish to be considered for all or any of the Offices should inform the ACRS Executive Director in writing two weeks (November 19, 2004) prior to the election.

Christmas Party

Each year, the members sponsor a Christmas party for the ACRS/ACNW Office staff during the December meeting. The members need to decide whether they want to continue with this tradition and sponsor a Christmas party to the ACRS/ACNW Office staff this year. If decided to hold a party, the ACRS/ACNW Executive Director suggests that the party be held at the Greenfields restaurant in Rockville.

Anonymous Letter Concerning the TRACE Computer Code Development and Review Practices

During the October 2004 ACRS meeting, the Committee considered the anonymous letters sent to Drs. Wallis and Ransom and directed the ACRS Executive Director to send these letters to the EDO for disposition. Accordingly, the ACRS Executive Director sent these letters to the EDO on October 14, 2004. In a memorandum dated October 19, 2004, Dr. Paperiello, RES Director, transmitted the anonymous letter to the Inspector General (IG) and stated that the issues raised in the letter appear to be technical in nature and will handle them accordingly. He also stated that the letter was being sent to the IG for any further action.

C. <u>Future Meeting Agenda</u>

Appendix IV summarizes the proposed items endorsed by the Committee for the 518th ACRS Meeting, December 2-4, 2004.

The 518th ACRS meeting was adjourned at 1:00 pm on November 6, 2004.

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

December 23, 2004

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

FROM: Mario V. Bonaca
Chairman

SUBJECT: CERTIFIED MINUTES OF THE 517th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), NOVEMBER 4-6, 2004

I certify that based on my review of the minutes from the 517th 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.

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

December 15, 2004

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador Sherry
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 517th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - NOVEMBER 4-6, 2004

Enclosed are the proposed minutes of the 517th meeting of the ACRS. This draft is being provided to give you an opportunity to review the record of this meeting and provide comments. Your comments will be incorporated into the final certified set of minutes as appropriate, which will be distributed within six (6) working days from the date of this memorandum.

Attachment:
As stated

7:30 a.m. and 4:15 p.m. (e.t.). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: October 19, 2004.

John H. Flack,

Acting Branch Chief, ACRS/ACNW.

[FR Doc. 04–23901 Filed 10–25–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Subcommittee Meeting on Safeguards and Security; Notice of Meeting

The ACRS Subcommittee on Safeguards and Security will hold a closed meeting on November 3, 2004, Room T–8E8, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be closed to public attendance to protect information classified as national security information and safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3).

The agenda for the subject meeting shall be as follows:

Wednesday, November 3, 2004—8:30 a.m. until 11:30 a.m.

The Subcommittee will hear presentations from the NRC staff, NRC staff consultants, and representatives of the industry regarding safeguards and security issues. The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

FOR FURTHER INFORMATION CONTACT: Mr. Richard K. Major (telephone: 301–415–7366) or Dr. Richard P. Savio (telephone: 301–415–7362) between 7:30 a.m. and 4:15 p.m. (e.t.).

Dated: October 19, 2004.

John H. Flack,

Acting Branch Chief, ACRS/ACNW.

[FR Doc. 04–23902 Filed 10–25–04; 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 November 4–6, 2004, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, November 21, 2003 (68 FR 65743).

Thursday, November 4, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10:30 a.m.: Proposed Rule for Risk-Informing 10 CFR 50.46. (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed rule for risk-informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and related matters.

10:45 a.m.–12:15 p.m.: Proactive Materials Degradation Assessment Program (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the status of the Proactive Materials Degradation Assessment Program.

1:15 p.m.–2:45 p.m.: Proposed Rule on Post-Fire Operator Manual Actions (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed rule on post-fire operator manual actions and related matters.

3 p.m.–4:30 p.m.: Grid Reliability Issues and Related Significant Operating Events (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding their activities associated with grid reliability, significant operating events related to grid stability, and other related matters.

4:45 p.m.–7 p.m.: Preparation of ACRS Reports (Open/Closed)—The Committee will discuss proposed ACRS reports on matters considered during this meeting. In addition, the Committee will discuss proposed reports on: Response to the August 25, 2004, EDO response to the May 21, 2004, ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG–1740, "Voltage-Based Alternative Repair Criteria;" AP1000 Lessons Learned Report; and Safeguards and Security Matters (Closed).

Friday, November 5, 2004, 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.: Status of Early Site Permit Reviews (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the status of the staff's review of the early site permit applications.

10:15 a.m.–11:45 a.m.: Assessment of the Quality of Selected NRC Research Projects (Open)—The Committee will discuss the preliminary results of the cognizant ACRS members' assessment of the quality of the NRC research projects on Sump Blockage and on MACCS code.

12:45 p.m.–1 p.m.: Plant License Renewal Subcommittee Report (Open)—The Committee will hear a report by the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim review of the license renewal application for the Farley Nuclear Plant.

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

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

2:30 p.m.–7 p.m.: Preparation of ACRS Reports (Open/Closed)—The Committee will discuss proposed ACRS reports.

Saturday, November 6, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

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

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 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) Pub. L. 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 as well as safeguard 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., e.t.

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

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

Dated: October 20, 2004.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 04–23903 Filed 10–25–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Federal Register Notice

DATES: Weeks of October 25, November 1, 8, 15, 22, 29, 2004.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of October 25, 2004

There are no meetings scheduled for the week of October 25, 2004.

Week of November 1, 2004—Tentative

There are no meetings scheduled for the week of November 1, 2004.

Week of November 8, 2004—Tentative

Monday, November 8, 2004

9 a.m. Briefing on Plant Aging and Material Degradation Issues—Part One (Public Meeting) (Contact: Steve Koenick, 301–415–1239)

1:30 p.m. Briefing on Plant Aging and Material Degradation Issues—Part Two (Public Meeting) (Contact: Steve Koenick, 301–415–1239)

This meeting (both parts) will be webcast live at the Web address—*http://www.nrc.gov.*

Week of November 15, 2004—Tentative

Tuesday, November 16, 2004

1:30 p.m. Briefing on Threat Environment Assessment (Closed—Ex. 1) (New time)

Thursday, November 18, 2004

1:30 p.m. Discussion of Security Issues (Closed—Ex. 1) (New date and time)

Week of November 22, 2004—Tentative

There are no meetings scheduled for the week of November 22, 2004.

Week of November 29, 2004—Tentative

There are no meetings scheduled for the week of November 29, 2004.

* The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Dave Gamberoni, (301) 415–1651.

"Briefing on Reactor Safety and Licensing Activities (Public Meeting)," originally scheduled for 9:30 a.m. on Tuesday, November 9, 2004, is being rescheduled for a later date.

The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/what-we-do/policy-making/schedule.html.*

The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (*e.g.* braille, large print), please notify the NRC's Disability Program Coordinator, August Spector, at 301–415–7080, TDD: 301–415–2100, or by e-mail at *aks@nrc.gov*. Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: October 21, 2004.

Dave Gameroni,

Office of the Secretary.

[FR Doc. 04–24010 Filed 10–22–04; 10:12 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to section 189a.(2) of the Atomic Energy Act of 1954, as amended

APPENDIX II

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

October 18, 2004

SCHEDULE AND OUTLINE FOR DISCUSSION
517th ACRS MEETING
NOVEMBER 4-6, 2004

THURSDAY, NOVEMBER 4, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:30 A.M. Proposed Rule for Risk-Informing 10 CFR 50.46 (Open) (WJS/MRS)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed rule for risk-informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and related matters.

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

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

3) 10:45 - 12:15 P.M. Proactive Materials Degradation Assessment Program (Open) (JDS/MWW)
- 3.1) Remarks by the Cognizant ACRS Member
- 3.2) Briefing by and discussions with representatives of the NRC staff regarding the status of the Proactive Materials Degradation Assessment Program.

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

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

4) 1:15 - 2:45 P.M. Proposed Rule on Post-Fire Operator Manual Actions (Open) (SLR/MDS)
- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed rule on post-fire operator manual actions and related matters.

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

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

5) 3:00 - 4:30 P.M. Grid Reliability Issues and Related Significant Operating Events (Open) (JDS/MWW)
- 5.1) Remarks by the Subcommittee Chairman
- 5.2) Briefing by and discussions with representatives of the NRC staff regarding their activities associated with grid reliability, significant operating events related to grid stability, and other related matters.

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

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

6) 4:45 - 7:00 P.M. Preparation of ACRS Reports (Open/Closed)
Discussion of proposed ACRS reports on:
- 6.1) Proposed Rule for Risk-Informing 10 CFR 50.46 (WJS/MRS)
- 6.2) Proposed Rule on Post-Fire Operator Manual Actions (SLR/MDS)
- 6.3) Grid Reliability Issues and Related Significant Operating Events (JDS/MWW)
- 6.4) Response to the August 25, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria" (GBW/FPF/CS)
- 6.5) AP1000 Lessons Learned Report (TSK/MME)
- 6.6) Safeguards and Security Matters (CLOSED) (MVB/RKM/RPS)

FRIDAY, NOVEMBER 5, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - 10:00 A.M. Status of Early Site Permit Reviews (Open) (TSK/MME)
- 8.1) Remarks by the Subcommittee Chairman
- 8.2) Briefing by and discussions with representatives of the NRC staff regarding the status of the staff's review of the early site permit applications.

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

9) 10:15 - 11:45 A.M. Assessment of the Quality of Selected NRC Research Projects (Open) (GEA/SLR/TSK/HPN/SD)
- 9.1) Remarks by the Cognizant ACRS Member
- 9.2) Discussion of the results of the Cognizant ACRS Members' assessment of the quality of the NRC research projects on Sump Blockage and on MACCS code.

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

10) 12:45 - 1:00 P.M. Plant License Renewal Subcommittee Report (Open) (MVB/CS)
Report by the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim review of the license renewal application for the Farley Nuclear Plant.

11) 1:00 - 2:00 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)
- 11.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
- 11.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

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

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

13) 2:30 - 7:00 P.M. Preparation of ACRS Reports (Open/Closed)
Discussion of the proposed ACRS reports on:
- 13.1) Proposed Rule for Risk-Informing 10 CFR 50.46 (WJS/MRS)
- 13.2) Proposed Rule on Post-Fire Operator Manual Actions (SLR/MDS)
- 13.3) Grid Reliability Issues and Related Significant Operating Events (JDS/MWW)
- 13.4) Response to the August 25, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria" (GBW/FPF/CS)
- 13.5) AP1000 Lessons Learned Report (TSK/MME)
- 13.6) Assessment of the Quality of Selected NRC Research Projects (GEA/SLR/TSK/HPN/SD)
- 13.7) Safeguards and Security Matters (CLOSED) (MVB/RKM/RPS)

SATURDAY, NOVEMBER 6, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

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

517TH ACRS MEETING
NOVEMBER 4-6, 2004

NRC STAFF (11/4/2004)

G. Kelly, NRR	J. Bongarra, NRR	C. Ader, RES
R. Landry, NRR	A. Klein, NRR	A. Lee, RES
B. Sheron, NRR	S. Wong, NRR	J. Muscara, RES
J. Vora, RES	R. Gallucci, NRR	M. Switzer, RES
T. Collins, NRR	J. Yerokim, RES	A. Hiser, RES
S. Black, NRR	B. Thomas, NRR	A. Wilson, RES
E. McKenna, NRR	E. Lois, RES	N. Chokshi, RES
G. Hammer, NRR	D. Trimble, NRR	D. Diec, NRR
M. Rubin, NRR	M. Johnson, NRR	M. Kaltman, NRR
W. Lyon, NRR	J. Lamb, NRR	S. Weerakkods, NRR
J. Wermiel, NRR	T. Koshy, NRR	R. Dipert, NRR
M. Barillas, NRR	J. Calvo, NRR	J. Jolicieur, OEDO
R. Taylor, NRR	J. Segala, RES	J. Hannon, NRR
R. Barrett, NRR	G. Lanik, RES	N. Iqbal, NRR
M. Mitchell, NRR	D. Rasmuson, RES	B. Radlinski, NRR
R. Dudley, NRR	J. Lazevnick, NRR	R. Rasmussen, NSIR
F. Orr, NRR	R. Roughly, RES	P. Lain, NRR
M. Tschiltz, NRR	R. Jenkins, NRR	Y. Orechioc, NRR
D. Franklin, NRR	S. Dinsmore, NRR	C. Jackson, OCM

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

C. Brinkman, Westinghouse	V. Gilbert, NEI	H. Leake, Palo Verde
J. Riley, NEI	R. Moye, SNC	
R. Dyle, SNC	S. Chun, SCE	
R. Jones, EPRI	S. Eodi, INEEL	
P. Negus, GE	D. Gladey, PPL	
K. Sakamoto, JNES	P. Brady, PPL	
N. Chapman, SERCH/Bechtel	W. Johns, EPRI	
S. Dolley, McGraw Hill	J. Gyrath, Exelon	
F. Emerson, NEI	T. Yamada, JNES	

<u>NRC STAFF</u> (11/5/2004)
R. Anand, NRR
N. Gilles, NRR
L. Dudes, NRR
W. Beckner, NRR
B. Musico, NSIR
J. Lee, NRR
D. Szware, NRR
G. Imbro, NRR
A. Murphy, RES
C. Munson, NRR
G. Bagchi, NRR
B. Harvey, NRR
D. Barss, NSIR
J. Wilson, NRR
D. Johnson, NSIR
M. Evans, RES
J. Mitchell, RES
M. Johnson, NRR
D. Solorio, NRR

<u>ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC</u>
J. Hegner, Dominion
S. Routh, Bechtel
C. Berger, Energetics
T. Miller, DOE
B. Hoffman, Public Citizen
R. Bell, NEI

APPENDIX IV: FUTURE AGENDA

INSERT A COPY OF THE NEXT MEETING, TYPE APPENDIX IV IN THE RIGHT HAND CORNER

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

November 17, 2004

SCHEDULE AND OUTLINE FOR DISCUSSION
518th ACRS MEETING
DECEMBER 2-4, 2004

THURSDAY, DECEMBER 2, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:00 A.M. Expert Elicitation on Large-Break LOCA Frequencies (Open) (GEA/MRS)
   - 2.1) Remarks by the Cognizant Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding draft predecisional NUREG-XXX, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," and the conclusions and recommendations of the Expert Elicitation Panel.

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

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

3) 10:15 - 11:45 A.M. Proposed Rule for Risk-Informing 10 CFR 50.46 (Open) (WJS/MRS)
   - 3.1) Remarks by the Cognizant Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed rule for risk-informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

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

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

4) 12:45 - 2:45 P.M. Technical Basis for Potential Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule (Open) (WJS/HPN/CS)
   - 4.1) Remarks by the Cognizant Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the NRC staff regarding the technical basis for potential revision of the PTS screening criteria in the PTS rule.

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

5) 3:00 - 4:30 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
5.1) Expert Elicitation on Large-Break LOCA Frequencies (GEA/MRS)
5.2) Proposed Rule for Risk-Informing 10 CFR 50.46 (WJS/MRS)
5.3) Technical Basis for Potential Revision of the PTS Screening Criteria (WJS/HPN/CS)

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

6) 4:45 - 7:00 P.M. Safeguards and Security Matters (Closed) (MVB/RPS/RKM)
Discussion of Safeguards and Security matters.

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

FRIDAY, DECEMBER 3, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - 9:00 A.M. Peer Review Comments on the Technical Basis for Revising the PTS Screening Criteria (Open) (WJS/HPN/CS)
8.1) Remarks by the Cognizant Subcommittee Chairman
8.2) Briefing by and discussions with the Chairman of the Peer Review Panel, as needed, regarding the Panel's comments on the technical basis for potential revision of the PTS screening criteria.

9) 9:00 - 10:30 A.M. Draft Commission Paper on Technology Neutral Framework for Future Plant Licensing - Policy Issues (Open) (TSK/MME)
9.1) Remarks by the Cognizant Subcommittee Chairman
9.2) Briefing by and discussions with representatives of the NRC staff regarding the draft Commission Paper on "Regulatory Structure for New Plant Licensing, Part 1: Technology Neutral Framework - Policy Issues."

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

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

10) 10:45 - 11:00 A.M. Subcommittee Report (Open) (GEA/MRS)
Report by the Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment regarding the status of development of the draft NUREG document on Treatment of Uncertainties.

11) 11:00 - 11:15 A.M. Subcommittee Report (Open) (MVB/CS)
Report by the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim review of the license renewal application for the Arkansas Nuclear One, Unit 2 Nuclear Power Plant.

12) 11:15 - 11:45 A.M. Election of ACRS Officers for CY 2005 (Open) (MVB/JTL/SD)
The Committee will elect Chairman and Vice Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee for CY 2005.

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

13) 1:45 - 2:45 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/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) 2:45 - 3:00 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB, 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.

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

15) 3:15 - 7:00 P.M. Preparation of ACRS Reports (Open/Closed)
Discussion of the proposed ACRS reports on:
15.1) Expert Elicitation on Large-Break LOCA Frequencies (GEA/MRS)
15.2) Proposed Rule for Risk-Informing 10 CFR 50.46 (WJS/MRS)
15.3) Technical Basis for Potential Revision of the PTS Screening Criteria (WJS/HPN/CS)
15.4) Technology Neutral Framework for Future Plant Licensing - Policy Issues (Tentative) (TSK/MME)
15.5) Safeguards and Security Matters (Tentative) (Closed) (MVB/RPS/RKM)

<u>SATURDAY, DECEMBER 4, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

16) 8:30 - 12:30 P.M. <u>Preparation of ACRS Reports</u> (Open/Closed)
Continue discussion of proposed ACRS reports listed under Item 15.

17) 12:30 - 1:00 P.M. <u>Miscellaneous</u> (Open) (MVB/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

<u>NOTE:</u>

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

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

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
517TH ACRS MEETING
NOVEMBER 4-6, 2004

[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 November 4-6, 2004

2 Proposed Rule for Risk-Informing 10 CFR 50.46
  2. Risk Informing 50.46 ECCS Acceptance Criteria presentation by B. Sheron, NRR [Viewgraphs]
  3. Regulatory Structure of Proposed Rule Risk-Informed 10 CFR 50.46 presentation by R. Dudley, NRR [Viewgraphs]
  4. ECCS Analysis Requirements presentation by R. Landry, NRR [viewgraphs]
  5. Risk-Informed Evaluation of the Acceptability of Proposed Plant Modifications presentation by G. Kelly, NRR

2 Proactive Materials Degradation Assessment Program
  6. Proactive Materials Degradation Assessment presentation by Dr. Peter Ford, ACRS Member (working with the Office of Nuclear Regulatory Research) and J. Muscara, RES [Viewgraphs]
  7. Update to ACRS on Industry Materials Initiative presentation by R. Dyle, Southern Nuclear Services [Viewgraphs]
  8. Defining Materials Degradation Vulnerabilities presentation by R. Jones, EPRI [Viewgraphs]

4 Proposed Rule on Post-Fire Operator Manual Actions
  9. Post-Fire Operator Manual Actions Rulemaking presentation by NRR [Viewgraphs]
  10. Industry Views Manual Actions Rulemaking: An Update presentation by F. Emerson, NEI [Viewgraphs]

5 Grid Reliability Issues and Related Significant Operating Events
  11. Electrical Grid Reliability presentation by J. G. Lamb, NRR [Viewgraphs]
  12. Overview of LOOP Frequency and Duration Update presentation by D. Rasmuson, RES [Viewgraphs]
  13. Loss of Offsite Power Events presentation by T. Koshy, NRR [Viewgraphs]

8 Status of Early Site Permit Reviews
  14. Early Site Permit (ESP) Review Status presentation by NRR [Viewgraphs]

15. Incident at the Davis Besse nuclear power plant (USA) rated at INES Level 3 of 6 March 2002, "Boric Acid Corrosion on the Reactor Vessel Head" and lessons learned for German plants [Handout from Dr. George Apostolakis, ACRS Member]

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

16. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - November 3, 2004 [Handout #11.1]

12 Reconciliation of ACRS Comments and Recommendations

17. Reconciliation of ACRS Comments and Recommendations [Handout #12.1]

Appendix V
517th ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Proposed Rule for Risk-Informing 10 CFR 50.46
	1. Table of Contents
	2. Proposed Schedule
	3. Status Report
	4. Report dated April 27, 2004, from Mario V. Bonaca, Chairman, ACRS, to Nils J. Diaz, Chairman, NRC, Subject: SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to LBLOCA Break Size and Plans for Rulemaking on LOCA With Coincident Loss-of-Offsite Power"
	5. Staff Requirements Memorandum dated July 1, 2004, from Annette L. Vietti-Cook, SECY, NRC, to Luis A. Reyes, EDO, NRC, Subject: Staff Requirements - SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant-Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power"
	6. Draft Rule Conceptual Basis and Draft Rule Language for Proposed Risk-Informed Revision to Emergency Core Cooling Requirements (10 CFR 50.46), dated July 27, 2004
	7. Memorandum dated October 14, 2004, from Catherine Haney, Program Director, Policy and Rulemaking Program, NRR to John T. Larkins, Executive Director, ACRS, Subject: Review of Risk-Informed 10 CFR 50.46 Proposed Rule Executive Summary and Draft Proposed Rule Language (Pre-Decisional for Internal ACRS Use Only)
3	Proactive Materials Degradation Assessment Program
	8. Table of Contents
	9. Proposed Schedule
	10. Status Report
4	Discussion on Post-Fire Operator Manual Action Rulemaking
	11. Table of Contents
	12. Proposed Meeting Schedule
	13. Status Report
	14. Memorandum from Catherine Haney, NRR, to John Larkins, ACRS, Subject: Review of Post-Fire Operator Manual Actions Proposed Rule
	15. Letter from Chairman Diaz to the Honorable Edward Markey, U.S. House of Representatives
	16. Letter from Chairman Diaz to the Honorable John D. Dingell, Ranking Member Committee on Energy and Commerce, U.S. House of Representatives
5	Grid Reliability Issues and Related Significant Operating Events
	17. Table of Contents
	18. Proposed Schedule
	19. Status Report

8 <u>Status of Early Site Permit Reviews</u>

20. Table of Contents
21. Proposed Schedule
22. Status Report
23. ACRS Report to the Honorable Richard A. Meserve, NRC Chairman, from Mario V. Bonaca, ACRS Chairman, dated March 12, 2003, Draft Review Standrd, RS-002: "Processing Applications for Early Site Permits"

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY
517TH FULL COMMITTEE MEETING

NOVEMBER 4-6, 2004

<u>NOVEMBER 4, 2004</u>
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING
PLEASE PRINT

NAME	NRC ORGANIZATION
Glenn Kelly	Special Proj DEDO
RALPH LANDRY	NRR/DSSA
BRIAN SHERON	NRR/ADPT
Jit VORA	RES/DET/MEB.
Tim Collins	NRR/ADPT
SUZANNE BLACK	NRR/DSSA
Eileen McKenna	NRR/DRIP
Gary Hammer	NRR/DE
Mark Rubin	NRR/DSSA
WARREN LYON	NRR/DSSA
Jerry Wermiel	NRR/DSSA/SRXB
Martha Barillas	NRR/DSSA/SRXB
Robert Taylor	NRR/DSSA/SRXB
RICHARD BARRETT	NRR/DE
Matthew L Mitchell	ROPMS
Richard Dudley	NRR/DRIP
FRANK ORR	NRR/SRXB
[illegible]	NRR/DSSA/SPSB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY
517TH FULL COMMITTEE MEETING

NOVEMBER 4-6, 2004

NOVEMBER ~~5~~ 4, 2004
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING
PLEASE PRINT

NAME	NRC ORGANIZATION
Daniel M Frumkin	NRC/NRR/DSSA/SPLB
J. BONGARRA	NRC/NRR/DIPM/IROB
Alex Klein	NRC/NRR/DSSA/SPLB
See-Meng Wong	NRC/NRR/DSSA/SPSB
Ray Gallucci	NRR/DSSA/SPLB
Jimi Yerokun	RES/DRAA
Brian Thomas	NRR/DRIP
ERASMIA LOIS	RES/DRAA
Dave Trimble	NRR/DIPM/IROB
Michael Johnson	NRR/DSSA
JOHN LAMB	NRR/DE
Thomas Koshy	NRR/DE
José Calvo	NRR/DE/EEIB
John Segala	RES/DSARE/ARREB
George Lanik	RES/DSARE/ARREB
Dale Rasmuson	RES/DRAA/OERAB
Jim Lazevnick	NRR/DE/EEIB
Bill Raughley	RES/DSARE/AAREB
Wayne	
Ronaldo V. Jenkins	NRR/DE/EEIB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY
517TH FULL COMMITTEE MEETING

NOVEMBER 4-6, 2004

NOVEMBER 4, 2004
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING
PLEASE PRINT

NAME	NRC ORGANIZATION
Stephen Dinsmore	NRR/DSSA
Charles Ader	RES/DRAA
ANDREA LEE	RES/DET
Joe MUSCARA	RES/DET
Mike Switzer	RES/DET
Allen Hiser	RES/DET
Adam Wilson	RES/DET
Nilesh Chokshi	RES/DET
David Diec	NRR/RPRP
MIKE KALTMAN	NRR/DRIP
Sunil Weerakkody	NRR/DSSA
RICHARD DIPERT	NRR/DSSA
JOHN JOLICOEUR	OEDO
John Hannon	NRR/DSSA
Naeem Iqbal	NRR/DSSA/SPLB
Bob Radlinski	NRR/DSSA/SPLB
Rick Rasmussen	NSIR/DNS/RSS
PAUL LAI	NRR/DSSA/SPLB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

517th FULL COMMITTEE MEETING
NOVEMBER 4-6, 2004

November 4, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Yuri Orechwa	NRR
Christopher Jackson	OCM
Charles Brinkman	Westinghouse
Jim Riley	NEI
Robin Dyle	SNC/RWKVIP
Robin Jones	EPRI
Paige Negus	GE
Kazunobu Sakamoto	JNES
Nancy Chapman	SERCH/Bechtel
Steve Dolley	McGraw Hill/Inside NRC
FRED EMERSON	NEI
VINCE GILBERT	NEI
Robert Moye	SNC
STEPHEN CHUN	SCE
Steve Eide	INEEL
David Gladey	PPL Susquehanna
Philip Brady	PPL Susquehanna
Wayne Johnson	EPRI
John Gyrath	EXELON NUCLEAR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

517th FULL COMMITTEE MEETING
NOVEMBER 4-6, 2004

November 4, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Tomoko Yamada	JNES
Harvey Leake	Palo Verde Nuclear Gen. Sta.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY
517TH FULL COMMITTEE MEETING

NOVEMBER 4-6, 2004

NOVEMBER 5, 2004
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING
PLEASE PRINT

NAME	NRC ORGANIZATION
RAJ ANAND	NRC/NRR/DRIP
Nanette Gilles	NRR/DRIP/RNRP
Laura Dudes	NRR/DRIP/RNRP
WILLIAM BECKNER	NRR/DRIP/RNRP
Bruce Musico	NSIR/DPR/EPD
Jay Lee	NRR/DSSA/SPSB
Dariusz Szwarc	NRR/DRIP/RNRP
Gene Imbro	NRR/DE/EMEB
Andrew Murphy	RES/DET/ERAB
Cliff Munson	NRR/DE/EMEB
GOUTAM BAGCHI	NRR/DE/EMEB
Brad Harvey	NRR/DSSA/SPSB
DAN BARSS	NSIR/DRP/EPD
JERRY WILSON	NRR/RNRP
Don Johnson	NSIR/DRP/EPD
Michele Evans	RES/ERAB/DET
Jocelyn Mitchell	RES/DSARE
Michael Johnson	NRR/DSSA
Dave Solorio	NRR/DSSA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

517th FULL COMMITTEE MEETING
NOVEMBER 4-6, 2004

November 5, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Joe Hegner	Dominion
Steve Routh	Bechtel
Carl Berger	Energetics
TOM MILLER	DEPARTMENT OF ENERGY
Brendan Hoffman	Public Citizen
Russ Bell	NEI

(1)

ITEMS OF INTEREST

517th ACRS MEETING

NOVEMBER 4-6, 2004

Introductory Statement by the ACRS Chairman

517th Meeting - November 4-6, 2004

The meeting will now come to order. This is the **first** day of the 517th meeting of the Advisory Committee on Reactor Safeguards. During today's meeting, the Committee will consider the following:

(1) Proposed Rule Language for Risk-Informing 10 CFR 50.46
(2) Proactive Materials Degradation Assessment Program
(3) Proposed Rule on Post-Fire Operator Manual Actions
(4) Grid Reliability Issues and Related Significant Operating Events
(5) Preparation of ACRS Reports

A portion of the meeting will be closed to discuss safeguards and security matters.

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act.

Dr. John T. Larkins is the Designated Federal Official for the initial portion of the meeting.

We have received no written comments from members of the public regarding today's sessions. We have received requests from NEI for time to make oral statements regarding proposed rule language for risk-informing 10 CFR 50.46, and the proposed rule on post-fire operator manual actions. A transcript of portions of the meeting is being kept, and it is requested that the speakers use one of the microphones, identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

I will begin with some items of current interest.

Introductory Statement by the ACRS Chairman

517th Meeting - November 4-6, 2004

The meeting will now come to order. This is the **second** day of the 517th meting of the Advisory Committee on Reactor Safeguards. During today's meeting, the Committee will consider the following:

(1) Status of Early Site Permit Reviews
(2) Assessment of the Quality of Selected NRC Research Projects
(3) Plant License Renewal Subcommittee Report
(4) Future ACRS Activities/Report of the Planning and Procedures Subcommittee
(5) Reconciliation of ACRS Comments and Recommendations
(6) Preparation of ACRS Reports

A portion of the meeting will be closed to discuss safeguards and security matters.

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act.

Mr. Sam Duraiswamy is the Designated Federal Official for the initial portion of the meeting.

We have received no written comments or requests for time to make oral statements from members of the public regarding today's sessions. A transcript of a portion of the meeting is being kept and it is requested that the speakers use one of the microphones, identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
517th MEETING
November 4-6, 2004, 2004

Page

STAFF REQUIREMENT MEMORANDUM

COMSECY

SPEECHES

CORRESPONDENCE

Significant Enforcement Actions

INSIDE NRC ARTICLES

IN RESPONSE, PLEASE
REFER TO: M041013A

October 22, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director of Operations

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

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON DECOMMISSIONING ACTIVITIES AND STATUS, 9:30 A.M., WEDNESDAY, OCTOBER 13, 2004, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the staff on the NRC's decommissioning activities and status which are now encompassed by the Comprehensive Decommissioning Program. In addition, three stakeholders who represented, respectively, an industry organization, a State regulator, and a corporation engaged in decommissioning complex sites, commented on their experiences with the NRC's decommissioning process.

At the next annual briefing on decommissioning activities staff should discuss its progress in capturing lessons learned and best practices from recent experience which can instruct both licensees and NRC staff undertaking current and future decommissioning projects.

Staff should continue to address salient issues that affect the efficiency and effectiveness of NRC's decommissioning program, including: improving radiological monitoring; establishing measures to provide finality in the decommissioning process; improving consistency among State and Federal regulators; and enhancing guidance to better address issues of flexibility in decommissioning approaches and institutional controls for restricted release scenarios.

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

October 21, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: COMSECY-04-0059 - STATUS OF DECOMMISSIONING PROGRAM - 2004 ANNUAL REPORT

The Commission has approved publication of the NUREG on the status of the NRC decommissioning program as contained in COMSECY-04-0059, subject to the following comments and edits.

1. This paper did not ask for any decision from the Commission. In future policy papers that are not provided for information only, the staff should clearly identify what they are asking the Commission to approve.

2. An additional appendix should be added to the report which lists sites by state, with the sites listed alphabetically within each state. Each site listed should have a cross reference to the appropriate page which contains the detailed site status.

3. The discussions for the research and test reactors are inconsistent in describing the status of the fuel. All the research and test reactors should have a brief discussion of the status of both fresh and spent fuel on the site. For example, the discussion of the University of Buffalo reactor states that there is no firm date for DOE to accept shipment of the fuel. It is the Commission's understanding that all fresh fuel has been shipped to another site and they are waiting on DOE to accept the spent fuel.

4. The title for section 10.3 (page 35) should be clarified to more clearly describe the actions in this section for someone outside the NRC organizational structure. These facilities are undergoing partial site decommissioning and will remain an active NRC licensee. The title for section 10.3 should be something similar to "Fuel Cycle Facility Decommissioning at Active Facilities". The discussion at the start of 10.3.1 should state that some active sites are undergoing partial decommissioning and will remain as active licensees. These sites remain the responsibility of the Division of Fuel Cycle Safety and Safeguards. This change will clarify the discussion for someone not familiar with the NRC organization.

5. The word "Sioux" (as in Sioux Falls, SD) is spelled incorrectly twice - see item number 28 on page 28 and at the top of page C-30).

6. The discussion for Millstone - Unit 1 (page A-9) and for Saxton (page A-14) should include a brief discussion on the fact that these units are under the responsibility of NRR and when (if ever) they are anticipated to be transferred to NMSS. The other facilities remaining under NRR responsibility have such language and the report should be consistent.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
OGC
CFO
OCA
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No. S-04-014

Remarks of Chairman Nils J. Diaz
Prepared for the
Americas Nuclear Energy Symposium (ANES)
Miami Beach, Florida
October 5, 2004

Thank you, Mr. White, for your kind introduction. It is my great pleasure and privilege to be able to address this third bi-annual Americas Nuclear Energy Symposium. I had the honor of speaking with you at the first ANES in 2000, and again two years ago during the second ANES. During the 2000 ANES, I spoke about safety and economics, and then in 2002 about energy security and national security. Today, I'd like to talk with you about some of the lessons the U.S. Nuclear Regulatory Commission (NRC) has learned, what we are doing today to ensure safety and security, and my thoughts on what else should we do. Of course, I didn't come here to just talk about the NRC, but also about the many common points of interest there are among the various countries who are here at this symposium.

Over the past two days, we have been engaged in discussing a variety of issues facing the safe and secure usage of radioactive materials and nucleo-electric power across the Americas. In particular, the Opening Plenary's panel discussion on major nuclear policy initiatives for the various countries in this hemisphere was especially noteworthy. It appears that several of the policy initiatives, as well as their associated implementation, that I have worked both as an NRC Commissioner and now as Chairman, fit in neatly with those discussed.

I am sure that everyone in this audience agrees that all of us -- the regulatory authorities, the regulated industries, the various vendors, and academia -- must continue to work effectively to ensure the safety and security of radioactive materials and nucleo-electric power. As I have often stated, our actions, both those of the regulators and the regulated, must be consistent, predictable, realistic, and *appropriately* conservative, with an unconditional commitment to safety, security, and preparedness. This, after all, is the nuclear business.

While the need for safety and security is certainly obvious, I believe that everyone here will also agree that we must stay adequately prepared for *credible* emergencies in order to continue to protect the public health and safety and the environment. In fact, being prepared isn't just the hallmark of the Boy Scouts, it is what conscientious scientists and engineers strive for when they thoughtfully and *realistically* analyze likely outcomes of their experiments and designs, including potential consequences, and then build in *appropriate* safety margins to mitigate and manage these scenarios.

However, counter-intuitive as it may initially sound, designing and regulating for highly *improbable* possibilities does not necessarily get you to the desired end state of increased safety and security. In some cases, designing and building to counter Murphy's axiom of "whatever can go wrong, will," may even shift the focus away from a more timely response to issues that have greater probabilities of being truly safety significant, like the early focus on fairly implausible large-break loss-of-coolant-accidents (LOCAs) over what we now know are the much more credible small-break LOCAs.

This can be demonstrated by examining a bit of history of the commercial nuclear power industry. Early on, the state of nuclear knowledge was quite limited by today's standards, especially the science of integrated risk assessments. As such, in order to deal with the many uncertainties that are inherent with any new technology, both the industry and first the Atomic Energy Commission (AEC), and then its successor the NRC, relied heavily on large engineering safety margins and what we now know were very unlikely design-basis-accidents (DBAs). This prescriptive, deterministic approach gave the U.S. and the world decades of safe and secure electric power generation performance, without injury to the public, with the exception of Chernobyl. It did not incorporate the state-of-the-art know-how into everyday regulation or even everyday operation.

This merits the question, what do the regulator and the regulated industry need to do to make things better? Shouldn't we be prepared, with the right tools, to face the challenges of a more technologically advanced and a more energy demanding world?

I would be remiss if I do not briefly address the issue of security by describing some of the key actions we have taken since 9-11.

The NRC has further strengthened security requirements at nuclear power plants and enhanced our coordination with federal, state, and local organizations.

We have ordered plants to take into account a more challenging adversarial threat; we are requiring tighter access controls and vehicle checks at greater stand-off distances; we have significantly improved force-on-force exercises to test the capabilities of plant defenders; we are demanding better readiness by plant security forces; and we have enhanced liaison with the intelligence community, and federal, state and local authorities responsible for protecting the national critical infrastructure through integrated response training.

In addition, the NRC has conducted research-based studies which concluded that a significant radiological release affecting public health and safety is unlikely from a terrorist attack, including a large commercial aircraft. And those studies show that time is available to protect the public in the unlikely event of a radiation release. Nuclear power plants have been and are even more so now among the most well protected elements of our national civilian infrastructure.

The NRC has undertaken several significant safety initiatives to make its regulatory activities more risk-informed and performance-based, as opposed to being prescriptive. A salient and functioning example of this is the transformation of the Inspection and Enforcement program into the risk-informed Reactor Oversight Process (ROP) and Significance Determination Process (SDP). The ROP, which is continuing to evolve as we gain experience, aims for objectivity over subjectivity, performance over prescription, and risk insights over design basis concerns. The objectives in developing and implementing this new oversight process were to provide the tools for inspecting and assessing licensee performance in a manner that was more risk-informed, objective, predictable, and understandable than the previous oversight processes, and that ensures the agency's performance goals are being met.

I am championing risk-informing the NRC's regulations to ensure that these requirements continue to make sense and that they are effective in focusing our programs, practices, and resources -- as well as the industry's -- on those activities that are most important to the public's safety. Specifically, the flexibility inherent in risk-informed regulations enables us to implement requirements that are appropriate to the risk presented by postulated hazards, and to do so with the use of state-of-the-art technology.

Let me reemphasize what I mean when I refer to "realistic conservatism," which is a term that I've been using for over a year now to describe my regulatory philosophy. Simply put, technical and regulatory decisions are informed by the *real world* -- utilizing advancing scientific knowledge, improving technological capabilities, and the lessons that have been learned through decades of operating experience -- in order to preserve *appropriate* and *prudent* safety margins. This allows regulatory authorities, such as the NRC, to provide oversight in a manner that corresponds to the *actual* risk presented, and not to an aphysical set of assumptions. I am confident that risk-informed and performance-based regulations can provide the quantitative edge to make realistically conservative decisions.

With over 10,000 reactor-years of operational experience internationally, and billions of dollars spent globally on research and development, we now know much more than we did early on, and thus do not need to continue to add <u>excessive</u> conservatism to nuclear power plant designs in order to ensure their safety. As such, we are developing an integrated, coordinated, and realistically conservative risk-informed and performance-based set of regulations.

I have covered a bit of the past and the present; let me touch on the future. There is an increasing need for energy security through diverse energy sources to continue the improvements in life brought out by economic development; this is especially true in the Americas. The future contribution of nuclear power generation depends on a complex of factors -- technological developments, business judgments, and regulatory actions all play a role. Experience has clearly shown, however, that nuclear power generation can be a valuable asset and an important component in a nation's energy mix. It can contribute to energy supply, improved energy security and environmental stewardship, year after year, now and in the future. The NRC, as a regulator, is ready to do its part in ensuring that nuclear technology continues to be a safe and reliable source of power that contributes significantly to the well-being of the people it serves.

Looking ahead to the new technologies that may be employed for nuclear power generation, I recognize that some, perhaps many, of our current regulations may not be directly applicable. This implies that there will need to be a regulatory framework to adequately address design and operational issues associated with future reactors that may be distinctly different from current light water reactor (LWR) designs. The NRC's present regulations were originally written for LWRs. However, I believe that, in the long-term, future reactors will be a mixture of evolutionary, or even revolutionary, LWRs and non-LWR technologies, such as the high-temperature gas-cooled reactor (HTGR) and others. I am convinced that improvements in efficiency are needed and, for generating power, efficiency depends on temperature. Reliable high-temperature reactors will eventually be developed which will provide these greater efficiencies.

To address the regulatory infrastructure of the future, the NRC has developed a performance-based, risk-informed, technology-neutral, design certification process under 10 CFR Part 52 that allows for enhanced safety and the early resolution of licensing issues, irrespective of the type of reactor. It provides a more stable and predictable licensing process, and resolves safety and environmental issues before authorizing construction.

I have advocated that nations that share common interests, like, for example, those involved in the development of Generation IV reactors, establish an internationally acceptable regulatory framework certifying the reactor design and safety analysis such that participating nuclear vendors and utilities could utilize this in designing and building new power plants. By doing so, we can substantially increase our ability to address safety and security matters in an international context, and increase the acceptability of these reactor designs to a variety of nations around the world.

I believe that ensuring the safety of nuclear power generation, making it more reliable, and potentially increasing its global availability -- and the benefits of improved energy security -- are issues we all need to address. We need the industry, vendors, and academia to continue to collaborate in ensuring that thoughtful consideration of safety, security, and preparedness is ingrained into everything we do.

And what do we want to achieve? Let me quote from the NRC's new Strategic Plan:

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

This statement embodies the principles of regulation that the agency believes are needed to be responsive to the needs of our society. I hope that you find them useful and I wish you well.

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No. S-04-017

The Honorable Nils J. Diaz
Chairman
Nuclear Regulatory Commission

at the

Nuclear Safety Research Conference
Washington, D.C.

October 25, 2004

Design Certification Across Borders

Good morning. I join Dr. Paperiello in welcoming you today. I'd like to discuss my thoughts on how regulators, such as the U.S. Nuclear Regulatory Commission (NRC), should further *enable* the safe and secure civilian use of nuclear energy and materials today and in the future. My remarks represent my personal views and are not intended to represent the Commission's views.

Before I continue, I must admit that I struggled over what I wanted to talk about today, especially since there are so many interesting topics available. For instance, I had considered discussing materials aging and degradation issues, or issues in developing and applying computer codes to analyze reactor systems. I also contemplated discussing radiation protection, emergency preparedness, or nuclear fuel issues for the next forty-five minutes. You may be glad to hear that one topic I did not consider discussing was nuclear security. Finally, I decided that I needed to use this forum to discuss a solution to one of the critical interfaces between regulators: the regulatory acceptability of reactor designs across borders prior to their full licensing, a solution that can foster regulatory cooperation and predictability.

I have said on several occasions that regulations should keep pace with technology. When nuclear technology was in its infancy, the regulations dealt with the various uncertainties by being prescriptive and conservative, usually *overly* prescriptive and *overly* conservative. Now, given the advances in technology, coupled with more than 10,000 reactor-years of operational experience internationally and the billions of dollars spent globally on research and development, we now know better where to focus our efforts and requirements. Uncertainty has been significantly reduced and both regulators and owners need to use the lessons learned. Therefore, the NRC is developing, and using as we develop, integrated and realistically conservative regulations that are both risk-informed and performance-based, and that are consistent and upgradable to the present level of knowledge and capabilities.

We, the regulators, also need to deal better with one of the realities of nuclear power -- its ever increasing "internationalization." Vendors all around the world supply the thousands of components and ideas that comprise a nuclear power plant, such as advanced reactor designs from the U.S. and Europe, steam generators from Spain, reactor vessels from Japan, and turbines from Germany. Given these circumstances, I believe that it is time for regulators around the world to *multilaterally* adopt a common safety framework for reactor designs.

There is no doubt that the right thing to do is to keep national licensing and regulatory authorities strong and responsible for making the decisions, but there are key parts of regulations that are amenable to "internationalization." We are familiar with the IAEA's Safety Standards, an internationally usable product, but I propose working toward a more specific regulatory product, maybe to the point that it could be considered a "commodity." Safety will be better served when certified designs can be accepted across borders as a commodity, fully respecting property rights. Therefore, I am convinced that regulators should seek to develop the tools needed to *certify* new reactor designs, as well as to certify the related research programs used to validate these designs, using multilateral agreements. The bottom line is that safety and regulatory decisions would be facilitated globally.

Earlier this year, I was asked by the Generation IV International Forum (GIF) and the Department of Energy (DOE) to talk about improving the regulatory framework for new generation reactors, and during the GIF meeting in Paris, I proposed that the development and international adoption of a regulatory framework to utilize safety assessments, compatible with the ongoing evolutionary nature of today's nuclear technologies, should include the certification of new reactor designs. This internationally-acceptable framework could establish a consistent set of appropriate requirements that nuclear vendors and regulators could utilize in designing and reviewing new power plants. Specifically, I offered the NRC's design certification process as a usable model.

We do not need to wait for Generation IV reactors. For already certified designs, the NRC would facilitate adoption of these certifications by other regulators by making a broad range of expertise, research results, and other resources available. For future design certification efforts, the NRC would encourage international participation by other regulators, in both the technical reviews and the related research efforts that support the certification, at the front end. It would be expected that other countries would do likewise, and regulatory consortia would be formed.

I am not advocating international licensing; licensing should remain each country's responsibility. I am advocating certifying reactor designs in a manner that *facilitates* their licensing. I believe that it is time for the world's nuclear regulators to begin building an internationally-acceptable regulatory reactor design certification to facilitate reactor licensing and regulatory decisions by individual countries.

The NRC developed a design certification process that provides a *stable* and *predictable* safety review for new nuclear power plant designs. This certification process resolves safety and environmental issues before authorizing construction, thus reducing licensees' financial risk while allowing for timely and meaningful public participation. Further, by placing the approved designs under a restrictive change process which applies to both the regulator and the applicant for design certification, we have reduced licensing uncertainty by ensuring that the safety issues already resolved will not be *needlessly* reconsidered during the plant licensing process. It should be noted that, when necessary, changes *can* be made by a *disciplined* certification amendment process.

The certification process examines:
(1) an essentially complete design, thus facilitating standardization;
(2) the final design information, which is equivalent to the information in a Final Safety Analysis Review (FSAR);
(3) the postulated site parameters;
(4) interface requirements; and,
(5) inspections, tests, analyses, and acceptance criteria (ITAAC).

To be clear, the certification process does not review site-specific safety issues, like seismology, environmental impact issues, operational programs, site-specific design features, or selected design areas. Site-specific issues are bounded to allow for separation of siting reviews from the design reviews. If a globally-acceptable certification process is developed based on a model similar to the NRC's, each country's regulatory authority should then have the bases for more efficient and effective licensing decision-making and greater resources for the resolution of country-specific issues in accordance with its own regulatory framework.

I have confidence in the design certification process; it has been tested and has been proven to be effective. Using it, the NRC has issued rules certifying three standard designs -- the Advanced Boiling Water Reactor (ABWR), the System 80+, and the AP-600. The AP-1000 design, which has received a safety evaluation report and final design approval, is now in the rulemaking phase of the certification process.

I mentioned earlier that acceptability could extend beyond new reactor designs. This concept can be used also used for major research projects used to validate those designs, or other significant research and development (R&D) issues. The NRC's Office of Nuclear Regulatory Research (RES) has been actively involved in seeking out opportunities to collaborate internationally on research activities. In fact, almost one hundred bilateral and multilateral agreements are in place to conduct research into activities as diverse as fuel issues, materials degradation, code development, and new reactor designs. In fact, this list looks suspiciously like the agenda for this conference. This is an area which could be more formalized, such that the participating regulators could take either the data from this collaborative research and make use of it in an individual way, much as is done today, or they could jointly analyze the data and produce a peer-reviewed report to support regulatory positions. This would provide greater regulatory consistency globally, while conserving the regulators' resources.

Let me be clear - I am *not* suggesting that this is the *only* way that various international nuclear regulatory authorities can successfully cooperate in developing an acceptable regulatory framework. Nor am I suggesting that this framework could, or even should, supplant any national regulations. I am proposing the establishment of an international framework for regulatory cooperation that will allow for resolving major regulatory issues, with common safety objectives, to better serve our countries.

Conclusion

The future contribution of nuclear power generation to the global energy mix depends on a variety of factors; technological developments, business judgments, and regulatory actions all play a role. Experience has clearly shown that nuclear power generation -- when well designed, constructed, operated, and regulated -- can be a valuable asset and an important component in a nation's energy mix and can contribute to environmental stewardship.

An important component of the nuclear energy business is its international activities. Regulatory activities need to keep pace with international developments. I am advocating another step in that direction by offering the U.S. NRC's design certification process as a model for cross-border regulatory cooperation. By doing so, we can substantially increase the regulators' ability to consistently address safety matters and contribute to the assurance of the safety of reactor designs for a variety of nations around the world.

As I said a week ago during an IAEA international conference, it is time to move forward from "a nuclear accident anywhere is a nuclear accident everywhere," to "a nuclear safety improvement anywhere is a nuclear safety improvement everywhere."

Have a great and fruitful conference. Thank you.

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No. S-04-016

The Honorable Jeffrey S. Merrifield
Commissioner
U.S. Nuclear Regulatory Commission

at the

NSRC Conference
Washington, D.C.

October 26, 2004

Introduction

As many of you may know, I have dedicated many speeches to encouraging our staff to remain on the cutting edge of innovations in technology in the areas of safety and security. I want to take this opportunity to highlight the need to remain just as vigilant in the area of emergency preparedness. Rudolph Giuliani has said that relentless preparation develops a culture of responsibility and awareness. The Nuclear Regulatory Commission plays a significant role in this country's preparedness and response in the unlikely event of a nuclear emergency. Therefore, we have a responsibility to ensure that we devote the resources and staff effort necessary to ensure that our country is prepared should that unlikely event ever materialize.

Big Problems Call for Big Solutions

To be clear, our efforts in the area of emergency planning are substantial. Our staff is well informed, dedicated, and does an outstanding job communicating with other federal agencies responsible for coordinating the nation's emergency response. That could not always be said of the NRC. Until the late 1970's emergency preparedness and response activities were fragmented among many federal, state, and local agencies. The NRC was one of them. During the event at Three Mile Island (TMI), however, the Nation became acutely aware of our shortcomings in responding to an event at a nuclear facility. Our communications with the State and media were abysmal and helped make a serious situation even worse. There was so much confusion over whether to order an evacuation that the public became distrustful of our ability to ensure their safety.

After Three Mile Island, the National Governors' Association requested that President Jimmy Carter centralize emergency functions, which he eventually did. Through an executive order he created, among other agencies, a new Federal Emergency Management Agency (FEMA). The NRC coordinated all emergency planning at nuclear facilities with FEMA. Over the years our agencies worked together to ensure that the communities in the areas of nuclear facilities were well prepared to respond to an event at a nuclear facility. Then the events of 9/11 put an even finer point on the need to be prepared. In the aftermath of 9/11 FEMA, along with 22 other federal agencies, programs and offices, became part of the Department of Homeland Security (DHS). With this change, our agency faces new challenges. We enhanced our emergency preparedness capabilities since 9/11, but I believe we should continually ask ourselves whether our emergency preparedness and response capabilities are up-to-date and adequate to respond to a radiological emergency.

Emergency preparedness has been, and will continue to be, an area on which we must maintain our focus. However, at virtually every regulatory body I know, including our own here at the NRC, the number of staff dedicated to this task is typically quite low, and in some cases not elevated to a level of importance equivalent to the technically oriented roles.

It is obvious to me that the level of attention that our Agency has historically directed toward emergency preparedness is not fully commensurate with its vitally important role of protecting public health and safety and the environment. I say this not merely as a regulator, but one whose early career was strongly influenced by this important concern. As a junior staff member for a New Hampshire senator in 1986, I spent more than half of my time on issues surrounding the licensing of Seabrook Station's nuclear power plant. Seabrook, which is located less than ten miles from the border between New Hampshire and Massachusetts, was the subject of significant controversy when then-Massachusetts Gov. Michael Dukakis refused to certify the Massachusetts portion of the ten-mile emergency planning requirements for Seabrook. This controversy nearly led President Reagan to issue an executive order reducing the NRC ten-mile EPZ requirement for Seabrook to two miles. Ultimately, the NRC adopted an alternate approach to certifying the adequacy of the emergency plan—an approach that ultimately eliminated Massachusetts' ability to block the licensing.

Preparation Boosts Public Confidence

For me, what was most critical about the Seabrook decision was the degree to which the public embraced the importance of emergency planning. It is key to providing the public confidence that the plant is safe. Unlike technical staff, who are more comfortable with discussing the likelihood of core damage frequency based on certain postulated events, most members of the public want to be assured that in the event of a radiological emergency, they and their families can be protected or evacuated.

The lesson borne out of the event at TMI, and following through the more recent concern in New York about the Indian Point site, is that the public is searching for an answer to a very simple question: If something happens, will my family be safe? To answer that simple question, one cannot provide a technically driven answer based on probabilistic assumptions postulating the likelihood of failure of key structures, systems, or components. Instead, robust emergency preparedness is the key to enhancing public confidence, and getting to the one-word answer: "yes."

At the very least, we need to ask ourselves if we have corrected all of the emergency response issues that surfaced during the TMI event. I am not sure that we have accomplished all that we should in this regard. We must be confident that our processes for making decisions on appropriate protective actions in the event of an emergency are adequate. Otherwise, the public will not follow our instructions and the benefits of the protective measures will be compromised. The TMI experience clearly illustrates this point.

On Friday, March 30, 1979, in the height of the confusion over the conditions at the plant, the NRC staff recommended to Governor Thornburgh an evacuation within five miles of the facility. Concern was in part sparked by radiation measurements taken on site. If properly interpreted, the measurements should not have signaled that there would be significant offsite releases because measurements of this magnitude had been seen in the previous days and did not result in significant offsite releases. Nevertheless, this measurement set off a crisis. While much of the panic can be blamed on a lack of a command and control structure at the NRC, a matter which I am completely confident has been corrected, some of the confusion stemmed from a lack of data, imprecise measurements, and lack of standards for determining when to take certain protective actions.

The public reacted just as one would expect. They distrusted the utility and the NRC as they sensed that there was too much confusion to place any confidence in the government's ability to protect their families. First the Governor had recommended sheltering in place, then evacuation of women and pre-school children. The response was that approximately 3500 pregnant women and pre-school children evacuated, roughly 83% of that population, but in addition 70,000 others evacuated as well.[1]

There will always be panic in the midst of a crisis, but a better, more credible and predictable system for measuring offsite releases would have eliminated some of the factors that fueled the increased panic during the TMI event.

Key Initiatives Following 9/11

Earlier this year, following a reexamination of our emergency response capabilities after 9/11, a unanimous Commission agreed to reorganize our emergency preparedness and response programs and enhance our emergency equipment. By almost tripling our staff in this area, and by placing this organization more harmoniously within the Office of Nuclear Security and Incident Response, the Commission has embraced Chairman Diaz's idea that we should be considered a safety, security, and emergency preparedness agency. I heartily believe this was a long-needed and valuable change. I further recommended, and the Commission agreed, to update and modernize our incident response center in Rockville.

Similarly, the federal government initiated a number of activities after 9/11 to better coordinate emergency response and preparedness. Creating DHS was perhaps the most substantial change. DHS has a number of initiatives underway, including responding to the President's direction to develop a new National Response Plan to align the various Federal resources into a unified, all-discipline and all-hazards approach to domestic incident management. This plan ties together the complete spectrum of federal incident management activities, including the prevention of, preparedness for, response to, and

[1] J. Samuel Walker, *A Nuclear Crisis in Historical Perspective Three Mile Island,* University of California Press 2004, pp. 138-139.

recovery from an act of terrorism, a major natural disaster or other major emergency, such as a radiological release from a nuclear facility.

Nonetheless, I believe for our part that increasing our staff, enhancing our organizational effectiveness, and updating our equipment are merely the first steps to meeting this important public challenge. I believe we need to do more.

Borrowing and Conducting Research

One of the benefits of my travel abroad is the ability to benchmark NRC capabilities with those of our counterparts. What is clear to me is that we have much to learn in this regard. Earlier this year I traveled to South Korea which included a visit to the Korea Institute for Nuclear Safety. I saw meteorological and plume release information that was more detailed, timely, and realistic than our own. Similarly, in a recent visit to our counterpart HSK in Switzerland, I saw an encapsulation of emergency response actions, including both evacuation and sheltering that were fully protective of public health, presented in a way that was not only useful for regulators, but also easily understandable for the general public.

Clearly we need to learn and share more. We need to work with other federal agencies including DHS, FEMA, and the Department of Energy, to examine whether we can improve our offsite radiation dose data collection systems, models for predicting where a plume may travel, and processes for determining appropriate levels of protective actions.

I believe that further research is needed in this area. Regulators around the world have used their unique experience to develop new, innovative and frequently economical methods of enhancing their effectiveness at protecting public health and safety. For the benefit of the American public, as well as for citizens of countries worldwide, we must more effectively share our best practices in this regard.

Having made that statement, do not mistake it as a call for significantly new levels of research funding. Indeed, my New England Yankee sensibility restrains me from doing so. Instead, at the NRC we need to more effectively utilize our bilateral relationships to identify methods, techniques, models and equipment that will allow us to enhance our emergency capabilities.

Furthermore, like the recent update of our incident response center, we need to identify new readily available off-the-shelf technologies to enhance our response capabilities.

Let me give you some specific examples. Offsite monitoring can be more predictable. In Switzerland, they have offsite monitors in the vicinity of their plants that feed real-time data to centralized computers that can be checked at any time to judge the conditions outside the site boundary. These measurements are publicly released on the Internet. Radiation detection and plume modeling equipment is more effective and cheaper than it was in years past. Approximately 15 years ago widespread use of radiation monitors was expensive, with per unit costs running in the hundreds of thousands of dollars. Today, portable devices can be deployed for significantly less. Similarly, enhancements in modeling using readily available information can provide more site specific determinations of plume dispersion that can more realistically consider the effects of weather and site specific geographical conditions. Many of our counterparts have taken this step and we should follow their lead.

Better real-time data in the event of an emergency would allow for more effective and timely decision-making. Our counterparts in Switzerland relying on these systems that provide precise data have developed dose thresholds for determining when to order a graduated level of emergency responses. For example at a particular dose, the public would be directed to shelter in place, with more significant dose thresholds calling for evacuation. These precise standards are supported by their state-of-the-art dose modeling equipment. What is impressive about this system is that it is easy for the public to understand, which should lead to more rational decisions by the public in response to an emergency. This is especially important when sheltering in place is the recommended response. It is difficult to convince the public to remain in place under changing conditions. We learned this lesson during TMI as 70,000 people decided to ignore the government's recommendation and evacuated anyway.

I am not recommending that we simply implement these programs. What is feasible for a small country with relatively few reactor sites may not be appropriate for our country. Switzerland has four reactor sites, we have sixty-seven. We should consider the approaches of other foreign regulators, especially those with larger nuclear programs, to determine if there are any insights that would be useful. To be consistent with our current regulatory principles, we should consider a risk-informed approach to emergency preparedness and response enhancements. For example, it may be that we should first consider radiation detection enhancements based on the proximity of a facility to a large urban area. Finally, we need to work with other federal and State agencies who may be responsible for implementing any enhancements to emergency preparedness programs.

Conclusion

As we meet this week to discuss the results of our past research efforts and seek to identify areas that need further effort, I urge you to look beyond the technical issues with which we may be most comfortable. It is clear to me, and I hope to you as well, that the public believes that we must not only do what is necessary to make certain that the plants we regulate are safe, but that we must also do what is necessary to ensure we are prepared for any contingency. Emergency preparedness, which had long been treated as an afterthought by this agency and others, has been given a full seat at the table with the recent staffing and organizational changes. Now we must make certain that this organization is fed and nurtured in a way that will assure the public that it is a strong and equal member of our public health family. Continued research and benchmarking can and should play a key role in making this happen. The public demands this focus and we must meet it.

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No. S-04-015

Remarks of Chairman Nils J. Diaz

to the

IAEA International Conference on Topical Issues in Nuclear Installation Safety: Continuous Improvement of Nuclear Safety in a Changing World
Beijing, China

October 18, 2004

Thank you. It is my great pleasure and privilege to speak with you today on a subject that I believe needs to receive much more attention -- how nuclear regulators have, and will continue to make vital contributions in ensuring nuclear safety, security, and preparedness. Nuclear regulators need to be recognized and supported in order that they can continue to enable the safety and security of nuclear power as a significant component of the world's energy supply. Regulators use a variety of frameworks to implement their activities, but all share common objectives.

Eight years ago, almost to the day, the Convention on Nuclear Safety (CNS), a key framework for regulators, was approved and entered into force for the signatory nations with civilian nuclear power programs which recognized the importance of ensuring that the safe use of nuclear energy needs to be well regulated and environmentally sound. The key CNS' objectives are, and I quote:

- "to achieve and maintain a high level of nuclear safety worldwide;
- to establish and maintain effective defenses in nuclear installations against potential radiological hazards; and,
- to prevent accidents with radiological consequences and to mitigate such consequences should they occur."

The CNS' framers were interested in promoting a high level of nuclear safety and an effective nuclear safety culture globally. The Convention entailed a commitment to the application of fundamental safety principles for nuclear installations rather than to detailed safety standards.

However, this original intent to give regulators and operators helpful directions on how to be more safety conscious has entered into the world of the 21st Century. As such, the Convention of Nuclear Safety is now even more important to the commonwealth of nations that have committed to it.

We need to recognize that there are threats we did not foresee ten years ago, that there are not just the possibilities of operator error, but of malevolent actions being taken against nuclear facilities that could cause significant consequences. Therefore, the concept of "safety" has undergone a significant revision in that we now recognize that safety also includes security and preparedness.

During the G-8 Nuclear Regulators Conference in Moscow in June of this year, a "Statement on Guidelines for Nuclear Safety and Security Regulatory Authorities" was developed. The objective for it is to complement the Convention on Nuclear Safety in supplementing the regulator's responsibilities. It is the G-8's intent that these Guidelines are available to all countries with civilian nuclear programs so that they may consider them as they enhance their regulatory framework, both for nuclear power plants and other nuclear installations. This is especially necessary in countries that are undergoing changes in their political infrastructure and when the legal and practical authority of the regulators need to be clearly defined. The G-8 Guidelines stated that, in order to accomplish the mission of being strong, effective, credible, transparent, and independent protectors of the public health and safety, security, and the environment, the nuclear regulator needs the necessary infrastructure and expertise, including the power to:

- regulate nuclear facilities and types of activities associated with the use of nuclear energy and utilization of radioactive materials,
- develop, and after approval to issue rules, regulations or other requirements to ensure safety and the protection of the environment,
- conduct a licensing process and to perform independent safety evaluations, as necessary,
- enforce the regulations,
- perform analysis to support the development of such rules and regulations and other requirements,
- require operators using nuclear energy and radioactive materials for civilian purposes to provide the information and reports about their activities,
- inspect the activities dealing with nuclear energy and radioactive materials,
- require compliance with license conditions and fulfillment of regulatory decisions, as well as to require remedial action for violation of regulatory requirements and to impose penalties, including suspension of operation,
- secure resources to conduct its activities effectively, and to attract and maintain a highly competent and respected technical staff,
- require the operator fulfills its primary responsibility and maintains competence for ensuring safety, and
- require appropriate emergency preparedness and response capabilities.

These are not new, yet together they form a simple yet compelling set of the authority and responsibility needed to exercise the mandate to protect the public and the environment from the regulated uses of nuclear materials.

The U.S. Nuclear Regulatory Commission is addressing these safety, security, and preparedness needs both in our day-to-day activities and in our revised Strategic Plan, which states that the NRC's mission is to:

License and Regulate the Nation's Civilian Use of Byproduct, Source, and Special Nuclear Materials to Ensure Adequate Protection of Public Health and Safety, Promote the Common Defense and Security, and Protect the Environment.

This is further captured in the Strategic Goals that we use to establish quantitatively how we are achieving our mission:

I. Safety: Ensure Protection of Public Health and Safety and the Environment.

II. Security: Ensure the Secure Use and Management of Radioactive Materials.

III. Openness: Ensure Openness in Our Regulatory Process.

IV. Effectiveness: Ensure That NRC Actions Are Effective, Efficient, Realistic, and Timely.

V. Management: Ensure Excellence in Agency Management to Carry Out the NRC's Strategic Objective.

The CNS has affirmed that the responsibility for nuclear safety rests with the State having jurisdiction over a nuclear installation, in the form of a properly constructed and authorized regulator. I agree and believe that the primary responsibility for nuclear safety resides with both the operator and the regulator. As I acknowledged during the "Global Threat Reduction Initiative (GTRI) Partners Conference" in Vienna, Austria, last month, the various national nuclear regulators may approach and resolve safety issues in different ways, but we understand that these differences do not equate to different goals or results. All of us are focused on ensuring adequate safety and security for nuclear power plants and radioactive materials of concern. However, I believe that we should, to the extent practicable, share information, expertise, and operating experience lessons learned to better allow all of us to achieve our mutual goals of safety, security, and preparedness.

Regulators historically have the expertise and have been capable of conducting the activities needed to address safety, security, and preparedness concerns in this post-9/11 era. Independent regulators can be centers of disciplined change, but only if they have, as the CNS states, adequate financial resources to support the safety of each nuclear installation and sufficient numbers of qualified staff with appropriate education and training. An independent and credible regulator with sufficient resources is a tremendous asset to both their nation and the international community, an asset that needs to be recognized and appropriately utilized by their nation.

As nuclear regulators, our primary responsibility is to provide, consistently and unmistakably, adequate protection from radiological hazards, including those resulting from terrorist acts, while preserving the benefits that the nation accrues from the use of nuclear materials and radioactive materials. We are also part of a well-established international network centered on the civilian uses of radiation, with well-known communications links, that is continuously addressing matters of importance to our nations and to the international community. These elements make nuclear regulators natural partners, and these are also the reasons that regulators need to be recognized and appreciated for the necessary work they do, day in and day out.

We need to be prepared, with the right tools, to face the challenges of a more technologically advanced and a more energy demanding world. By giving regulators the necessary legal authority and the appropriate resources, and by encouraging that they work closely with their international counterparts to share knowledge, expertise, and to develop internationally acceptable standards and regulations, we will be better able to ensure the safety and security of this essential component of the

Twenty-First Century energy mix.

The NRC is ensuring that we have in place appropriate and realistic regulations and processes that will enable this next generation of reactors to be safely and securely built and operated. As such, we have developed a design certification process that provides a stable and predictable licensing process for new nuclear power plant designs. This process resolves safety and environmental issues before authorizing construction, thus reducing licensees' financial risk while allowing for timely and meaningful public participation. However, we have retained the capability to effect changes to insert technological advances via a disciplined license amendment process. Further, by placing the approved designs under a restrictive change process, that applies to both the regulator and the applicant for design certification, we have reduced licensing uncertainty. The Commission assures license applicants who reference a certified design that the safety issues already resolved will not be needlessly reconsidered during the plant licensing process. The NRC has issued rules certifying three standard designs -- the Advanced Boiling Water Reactor (ABWR), System 80+, and the AP-600 -- and the AP-1000 design, which has received a safety evaluation report and final design approval, is now in the rulemaking phase of the certification process.

Earlier this year, I proposed to the Generation IV International Forum (GIF) meeting in Paris, France, that the development and international adoption of a regulatory framework that can establish the appropriate safety requirements, compatible with the ongoing evolutionary nature of today's nuclear technologies, is the logical next step. This internationally acceptable framework could put into place a consistent set of regulatory requirements that any nuclear vendor and utility could utilize in designing and building new power plants. Specifically, I offered the NRC's design certification process as a starting point for the world's nuclear regulators to use in starting to build an internationally-acceptable regulatory framework.

The IAEA has a tremendous job to do in supporting and advocating safety and reliability, and that includes advocating regulators' capabilities and expertise, and, in doing so, they will be championing nuclear safety, security, and preparedness worldwide. It is time to move forward from "a nuclear accident anywhere is a nuclear accident everywhere," to "a nuclear safety improvement anywhere is a nuclear safety improvement everywhere," and that is everyone's job.

Thank you.

October 14, 2004

The Honorable Christopher Shays, Chairman
Subcommittee on National Security,
Emerging Threats, and International Relations
Committee on Government Reform
United States House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I want to express my appreciation to you and the other Members of the Subcommittee on National Security, Emerging Threats, and International Relations for holding the September 14, 2004 oversight hearing to examine the NRC's security enhancements for nuclear power facilities. The Commission was pleased to have Mr. Luis A. Reyes, NRC's Executive Director for Operations, and Mr. Roy Zimmerman, Director of the Office of Nuclear Security and Incident Response, update the Subcommittee on recent actions the NRC has taken to enhance the security of NRC-regulated nuclear facilities and radioactive materials. However, the Commission would like to take this opportunity to address some of the Members' concerns on this vitally important subject.

Nuclear security is a top priority for the NRC. As the Government Accountability Office (GAO) pointed out in its testimony, following the September 11, 2001 terrorist attacks, the NRC immediately placed nuclear power plants and other facilities at the highest level of alert. NRC responded quickly and decisively to strengthen existing security at these facilities. A very important point, which is not specifically addressed in the GAO testimony, is the fact that the NRC inspected every nuclear plant to verify licensee implementation of agency Orders requiring enhanced security measures. Subsequently, additional Orders were issued regarding site access authorization, the design basis threat, and security guard training and qualification. The NRC continues to inspect licensee compliance with these Orders under the revised baseline inspection program. I can assure you that the NRC's oversight of nuclear plant security involves a lot more than just a "paper review"; in fact, it is a hands-on, day-in and day-out inspection and assessment process.

The NRC conducts on-site inspections by security specialists assigned to the Regional Offices and, by 2003, this direct inspection effort had increased more than 50 percent beyond the effort extended annually before the terrorist attacks. This does not include the significantly enhanced force-on-force exercises discussed below. In addition, there are at least two NRC resident inspectors at each of the nuclear power plant sites who maintain daily vigilance over matters of nuclear safety as well as other regulatory activities at the site, including security. Inspection findings are reported to the Regional Offices for followup and coordination with Headquarters when necessary.

The NRC has continued to improve its security performance evaluation program (i.e., force-on-force exercises), which the Commission considers to be an important element for ensuring protection of the Nation's critical infrastructure. In February 2003, the NRC resumed the force-on-force exercises in the form of a pilot program to test recent enhancements. In February 2004, the NRC began a transition force-on-force program, which incorporated the lessons learned during the pilot program. The transition program follows the same format as the pilot program; however, the "mock adversary" force now uses the characteristics of the Design Basis Threat (DBT), as enhanced and supplemented by Orders, to prepare for resumption of the full security performance assessment program in November 2004. Under that program, the NRC will conduct approximately 22 force-on-force exercises per year so that each site's security will undergo an NRC-evaluated exercise at least once every three years. This represents a significant increase in the exercise frequency; in addition, each plant is required to conduct independent exercises at least once each year.

During the pilot program, the NRC identified the need to improve the offensive capabilities, consistency, and effectiveness of the exercise adversary force. The Commission addressed this need by directing the staff to develop a training standard for a Composite Adversary Force (CAF). The NRC staff is responsible for selecting the mock attack scenarios, overseeing the performance of the CAF, and evaluating the adequacy of the site's security. The CAF for a given NRC-evaluated force-on-force exercise will be comprised of security officers from various nuclear power facilities (excluding the site being evaluated) and will have been trained in offensive, rather than defensive, skills to perform the adversary function. During the hearing, some Subcommittee Members expressed concern regarding a potential conflict of interest on the part of the contractor selected by the industry (Wackenhut Corporation) to supply the CAF members. The Commission shares these concerns and for that reason has directed the staff to take appropriate actions to ensure the independence of the CAF. It is important to emphasize that the CAF members do not evaluate site security. Their role is to provide a credible adversary force that meets standards for training, fitness, and tactical skills that have been established by the NRC. In addition, administrative controls have been established within the industry's CAF contractor to ensure that the CAF includes members from sites not protected by Wackenhut, that CAF members will not participate in NRC-evaluated exercises at the site from which they came, and that the CAF remains organizationally independent of the portion of Wackenhut that provides security services to the sites.

Another question that Members of the Subcommittee raised during the hearing dealt with NRC's treatment of non-cited violations. Both cited and non-cited security violations are documented in NRC's inspection reports. Security violations associated with the implementation of the recently issued Orders are reviewed by an NRC panel to determine their significance and priority for followup. NRC inspectors follow up on all violations that are considered significant. In addition to security-related inspections, NRC inspectors routinely evaluate the adequacy of each licensee's program for identifying and resolving plant problems. This includes samples in each cornerstone of the inspection program throughout the year, as well as a broad overview of each licensee's problem identification and resolution program conducted biennially. In these inspections, the NRC appropriately focuses on the issues of safety and security significance. I can assure you that NRC will take prompt and appropriate enforcement action if these inspections reveal programmatic issues with the licensee's corrective actions to address previously identified violations.

The Commission is confident that nuclear power plants continue to be among the best protected private sector facilities in the Nation, and the NRC is absolutely committed to ensuring strong security at these facilities. As Mr. Luis A. Reyes emphasized at the hearing, there are several legislative proposals which would grant the NRC the statutory authority for steps that the Commission believes should be taken to enhance further the protection of the country's nuclear infrastructure and prevent malevolent use of radioactive material. The support of your Subcommittee in helping to enact these legislative proposals would be greatly appreciated. The details on the needed legislative proposals were provided to you with the NRC's written testimony.

I appreciate the Subcommittee's continued interest in the NRC's oversight of the nuclear power facilities. If you need further information, please do not hesitate to contact me.

Sincerely,

/RA/

Nils J. Diaz

cc: Representative Dennis J. Kucinich, Ranking Member
Representative Michael R. Turner, Vice Chairman

October 1, 2004

The Honorable Edward J. Markey
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Markey:

On behalf of the U. S. Nuclear Regulatory Commission (NRC), I am responding to your letter of August 23, 2004, regarding changes in force-on-force exercises and information security at nuclear power plants licensed by the NRC.

The NRC staff has prepared answers to your questions, which are enclosed. If you need additional information, please contact me.

Sincerely,

/RA/

Nils J. Diaz

Enclosure: As stated

NRC RESPONSE TO QUESTIONS FROM CONGRESSMAN EDWARD MARKEY DATED AUGUST 23, 2003 REGARDING FORCE-ON-FORCE EXERCISES AT NRC-LICENSED FACILITIES

Question 1: Please fully describe the Commission's re-vamped force-on-force (FOF) program. How often will FOF exercises be conducted? Who is responsible for designing the FOF exercises? Who will conduct them? How will they be evaluated? What enforcement actions can be taken if a licensee "fails" an FOF exercise? What are the criteria for "passing " an FOF exercise?

Answer:

The NRC's enhanced FOF program is the result of over 2 years of program redesign and pilot testing at almost two-thirds of the nuclear power reactors in the country. An expanded table-top exercise program conducted during 2002 and an expanded FOF exercise program (EFOF) conducted during 2003 evaluated the bases for revised FOF procedures and the impacts of compensatory measures and expanded adversary characteristics associated with the Orders issued on February 25, 2002, by the Commission. A transitional FOF program (TFOF) conducted during 2004 focused on implementation of the lessons learned in the EFOF program and refinement of the FOF procedures and guidance. During the TFOF period, the NRC increased staffing levels, established initial qualifications, and conducted training for its FOF staff. At the conclusion of the current TFOF program in October 2004, the program will enter a full regulatory oversight program with an NRC-evaluated FOF exercise at each licensee site once every 3 years.

Beginning in November 2004, NRC-evaluated FOF exercises will be fully integrated with the baseline inspection program for physical protection and material control and accounting for power reactors and Category I fuel facilities. The NRC inspection program is designed to verify compliance with the regulatory requirements. The NRC conducts inspections that evaluate the effectiveness of security program performance and include observations of the security force members and their supervisors. Only the NRC team is responsible for NRC-evaluated FOF exercises, including designing exercises and evaluating licensees' performance. The NRC will use an industry-supplied adversary force in FOF exercises, but the adversary force will be directed by the NRC team. Regardless of whether the plant security force is composed of licensee employees or contractor employees, the NRC holds licensees, not contractors, accountable for security performance.

NRC-evaluated FOF exercises are not pass/fail inspections. At the conclusion of an FOF exercise, NRC's evaluators assess their observations and findings in accordance with the Physical Protection Significance Determination Process (PPSDP). The PPSDP comprises two parts, a baseline portion and an FOF portion. Security-compliance findings identified during triennial FOF activities that are not directly related to an FOF exercise outcome are assessed using the baseline portion of the PPSDP. Outcome-related findings are assessed using the FOF portion of the PPSDP. The FOF PPSDP assesses the significance of a licensee's demonstrated performance relative to preventing significant core damage or spent fuel sabotage resulting in a radiological release.

The FOF PPSDP considers the defense of target sets (combinations of components that could result in significant core damage) in conjunction with preventing a radiological release path as the measure of significance. The loss of a single target set with no radiological release represents the least significant finding and the loss of multiple target sets with the creation of a radiological release path represents the most significant finding. For any exercise result that indicates a significant weakness in the licensee's protective strategy, the NRC team remains on site until appropriate compensatory measures are in place.

Throughout the EFOF and TFOF pilot programs in 2003 and 2004, respectively, the NRC observed modified enforcement guidance under which enforcement discretion was applied to findings which were determined to be related to either the FOF process itself or the April 2003 supplemental requirements that implement the enhanced Design Basis Threat (DBT). Beginning in November 2004, the supplemental requirements will be fully effective in accordance with the April 2003 Orders from the Commission. NRC-evaluated FOF exercises will then be conducted using exclusively the supplemental requirements and the licensees' approved Security and Contingency Plans. The enforcement discretion guidance of the pilot programs will no longer apply, and licensee performance will be subject to enforcement action consistent with PPSDP findings and the provisions of NRC's Enforcement Manual.

Question 2: Why isn't the NRC providing its own dedicated mock terrorist force to conduct FOF exercises at nuclear reactors, or making arrangements with other federal agencies with experience in this area, rather than allowing NEI -- the trade association and lobbying arm of the nuclear industry -- to perform this function?

Answer:

Since the inception of the force-on-force (FOF) security exercise program in the early 1980's, there has been essentially no change in the practice of using security officers from the facility being evaluated, other nuclear power facilities, or local law enforcement officers to serve as mock adversaries. During pilot program FOF exercises aimed at strengthening the program in 2003, the NRC identified a need to improve the offensive abilities, consistency, and effectiveness of the exercise adversary force. Staff evaluated several options, including continuing under the established process or establishing a dedicated adversary force (the dedicated adversary force considered the use of NRC staff, other federal personnel, or industry personnel). Staff evaluated the impacts and benefits of each option and provided a recommendation to the Commission. The staff recommended and the Commission approved the establishment of adversary force standards and guidelines for the industry as a group. The industry would select and train a pool of personnel for the adversary force cadre. The Commission directed the staff to develop a training standard for a Composite Adversary Force (CAF), which will travel from site to site to serve as the mock adversary. The CAF for a given NRC-evaluated FOF exercise will be comprised of security officers from various nuclear power facilities (excluding the licensee being evaluated) and will have been trained in offensive, rather than defensive, skills to perform the adversary function. As a result of this initiative, a significant problem of a lack of uniformity in the quality of adversary forces has been resolved. For the first time, NRC staff and contractors will have available a uniform, high quality adversary force trained to NRC standards at all force-on-force exercises conducted by NRC starting later this year.

CAF members do not perform an evaluative function. The NRC and its subject matter expert (SME) contractors evaluate the performance of each licensee during FOF exercises, and the NRC will continue to evaluate the abilities, consistency, and effectiveness of the exercise adversary force.

The industry has selected Wackenhut to manage the CAF. Wackenhut also provides protective services to a substantial number of operating power reactors. The NRC recognizes that some may perceive a conflict of interest with respect to the same contractor providing both the protective services to some individual sites and staffing some members of the adversary force used for exercises. The Commission has directed the staff to ensure that there is a separation of functions, including appropriate management and administrative controls in place within the Wackenhut organization to provide adequate independence between the CAF and the nuclear reactor guard force. In addition, the NRC will continue to assess the performance of the adversary force and require improvements if appropriate, up to and including developing an NRC-contracted adversary force. However, one of the benefits of an industry adversary force is the feedback of an adversary's perspective into enhancement of site protective strategies and security officer training at his or her normally assigned facility, as well as improving the quality of FOF exercises conducted by the licensees annually for training.

The NRC staff considered the aspect of possible conflicts of interest in the exercise program, and the Commission deliberated on the issue before deciding that the industry could be permitted to use its own employees as mock adversaries. Potential drawbacks to NEI's decision, including questions about objectivity, are outweighed by the opportunity to promptly field adversary forces that are better trained and dedicated to the role assigned them.

It is important to note that licensee employees have been used in the role of the mock adversary since the earliest force-on-force testing was initiated in 1982. Sometimes, the teams were made up of security force members from the site being tested, sometimes they were security force members from other licensee facilities within the same corporate structure, and sometimes they were from other security forces altogether, including other licensees and law enforcement agencies. The CAF is a distinct improvement over those practices. NEI's selection of a contractor with an extensive history of training and qualifying security officers for the nuclear industry should ensure that they will bring a high level of skill to bear on the exercises. In addition, their familiarity with nuclear power plant design should make them a more worthy adversary for licensees responding to the exercises. The NRC has issued standards for physical fitness, training, and knowledge of attack strategies to ensure that the CAF will be better trained than previous adversary forces. NEI's selection of a contractor with an extensive history of training and qualifying security officers for the nuclear industry should ensure that they will bring a high level of skill to bear on the exercises.

Further, NEI, in a letter dated September 10, 2004, has made a commitment to the NRC that: (a) the manager of the CAF will report directly to the Chief Executive Officer for Wackenhut, (b) CAF members will be selected from all sites, including those where security is provided by Wackenhut's competitors, and (c) a CAF member will not participate in exercises at his or her home site.

Question 3: The nuclear industry has long resisted most efforts to increase security at nuclear reactors, and has even challenged the Commission's authority to perform FOF tests in the first place (see NRC email cited in http://www.house.gov/markey/issues/iss_nuclear_ltr990708.pdf). Don't you think there would be a disincentive for any mock terrorist force paid for by the nuclear industry to conduct FOF exercises in a rigorous manner that could uncover systemic weaknesses in security at nuclear reactors? If not, why not?

Answer:

The NRC will independently evaluate licensee performance in FOF exercises at each site on at least a triennial basis, using the CAF to provide a consistent, capable, and effective adversary. The CAF will not perform an evaluative role in the exercises. CAF members will arrive on site at about the same time that the NRC evaluation team arrives and will be coordinating closely with the NRC evaluation team and the NRC's subject matter expert contractors before and during the exercises. Any indication that CAF members are not performing rigorously in their role will be promptly identified and addressed by the NRC. The NRC routinely reassesses the effectiveness and efficiency of its FOF evaluation program and has mechanisms in place to revise or improve its FOF processes and procedures as needed. Should the industry be unable to maintain an adequate and objective CAF that meets the standards mandated by the NRC, the NRC will take the necessary actions to ensure the effectiveness of the FOF evaluation program.

Question 4: Wackenhut is responsible for nuclear reactor security at 30 of 64 nuclear power plants in the U.S. Don't you think that there would be a disincentive for the Wackenhut mock terrorist force to rigorously test security at power plants at which Wackenhut also provided the security forces as rigorously as it would at power plants at which Wackenhut's competitors provided the security guard forces? If not, why not?

Answer:

As discussed in answers 2 and 3, above, the NRC recognizes that a perceived conflict of interest exists regarding the industry's selection of a CAF management organization that provides protective services to a large portion of the operating power reactors. Because the CAF does not establish the exercise objectives, boundaries, or timelines, and because the CAF's performance is subject to continual observation by NRC's staff and contractors, the NRC can control the exercise. The commitments by NEI in its letter of September 10, 2004, provide additional assurance that the CAF will conduct exercises at all sites with equal rigor.

Question 5: Was the NRC aware that NEI planned to hire Wackenhut to conduct the FOF tests, even though Wackenhut is responsible for security at 30 of 64 nuclear power plants? If so, why did the NRC allow a contract that poses such a blatant conflict of interest to proceed?

Answer:

The NRC was aware that NEI was considering Wackenhut among other suppliers of CAF member personnel. At the time, the NRC expressed concern and understood that, if Wackenhut were to be selected, the CAF would be managed by a separate business entity within the Wackenhut organizational structure. NRC's published standards for CAF members are focused on performance and qualification standards.

NEI, in a letter dated September 10, 2004, has made a commitment to the NRC that: (a) the manager of the CAF will report directly to the Chief Executive Officer for Wackenhut, (b) CAF members will be selected from all sites, including those where security is provided by Wackenhut's competitors, and (c) a CAF member will not participate in exercises at his or her home site.

Question 6: The Commission recently announced that it would no longer provide any information regarding the assessment of security at nuclear reactors or enforcement actions taken regarding security at nuclear reactors to the public. Please explain why this decision was made. Why can't any information, even information that is not specific to particular reactor vulnerabilities, be publicly available?

Answer:

The Commission deliberated for many months on how to balance its commitment to openness with the concern that some key information is sensitive and might be misused by those who wish us harm. While we have worked diligently to share sensitive information with licensees, Federal agencies, and State and local governments to enhance protection of the public, we have also redoubled our efforts to ensure that we do not release information that could be exploited by adversaries in sabotaging nuclear facilities or stealing nuclear materials. As you recognize, the Commission has overall responsibility for public health and safety and the common defense and security with regard to the utilization of commercial nuclear material. Therefore, we must weigh the information that is made public in that context. Accordingly, the Commission determined that security findings in the Reactor Oversight Process and similar programs for other facilities will no longer be made public. However, the staff plans to develop a publicly available report that would provide some general information about plant performance assessment in the physical protection and security area without revealing any site specific details or compromising security.

Question 7: Please provide copies of all correspondence, emails, memoranda, and telephone logs in the possession of the NRC, including those received by and sent to representatives of the nuclear industry, regarding the decision to withhold this information from the public.

Answer:

The material requested is being prepared. It will be forwarded to you under separate cover.

EA-04-038 - Point Beach 1 & 2 (Nuclear Management Company, LLC)

September 29, 2004

EA-04-038

Mr. Dennis Koehl
Site-Vice President
Point Beach Nuclear Plant
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241-9516

SUBJECT: EXERCISE OF ENFORCEMENT DISCRETION
[NRC OFFICE OF INVESTIGATIONS REPORT NO. 3-2001-033]

Dear Mr. Koehl:

This refers to information provided to the U.S. Nuclear Regulatory Commission (NRC) on July 6, 2001, by a representative of Nuclear Management Company (NMC) concerning alleged employment discrimination at the Point Beach Nuclear Plant. The NMC Employee Concerns Program (ECP) received information indicating that a General Foreman, employed by Day and Zimmerman Nuclear Power Systems (D&Z), a contractor at the Point Beach Plant, submitted the name of a D&Z electrician for lay-off on May 4, 2001, following the electrician's protected activities associated with a work package on March 27 and 28, 2001. The matter was investigated by NMC and the NMC investigator concluded that employment discrimination had occurred. The NRC Office of Investigations (OI) also investigated the matter and the information obtained by OI indicated that an apparent deliberate violation of 10 CFR 50.7, "Employee Protection," occurred when the D&Z General Foreman submitted the name of the D&Z electrician for lay-off after the electrician engaged in protected activities. A copy of the synopsis from the OI report was sent to you on April 1, 2004.

Based on the information developed during investigations by NMC and OI and information contained in a letter from NMC dated May 10, 2004, the NRC determined that a violation of NRC requirements occurred. Specifically, on March 27 and 28, 2001, a D&Z electrician and a coworker found that the required signatures of the duty shift supervisor and reviewing engineer were missing from a work package. The electrician and a co-worker stopped work on the assigned project and notified a D&Z supervisor of the problem. A coworker of the electrician prepared a Condition Report on the subject. A D&Z General Foreman learned of the electrician's activities on March 27 and 28, 2001, and on March 30, 2001, that General Foreman threatened to terminate the employment of the electrician for stopping work. The General Foreman stated that his intention on March 30, 2001, was not to terminate the electrician or his coworker, but to separate the two employees from each other because of the excessive number of breaks they were taking. About April 30, 2001, the electrician was told that he would not be laid-off during a May 2001 reduction in force and he would be retained until the end of the project later that summer. However, the General Foreman submitted the electrician's name for lay-off on May 4, 2001, in part, because the electrician engaged in protected activities on March 27 and 28, 2001. By submitting the electrician's name for lay-off, the General Foreman changed the compensation, terms, conditions, or privileges of the electrician's employment in violation of 10 CFR 50.7. Additionally, the General Foreman allowed two other electricians laid-off on May 4, 2001, to "hover" (remain immediately eligible for reemployment by D&Z without returning to the local union hall for reassignment). However, the General Foreman did not extend the offer to "hover" to the complainant in this matter. The NRC considered the General Foreman to be a first-line supervisor or other low-level manager within the D&Z organization; therefore, the violation is categorized in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600 (Enforcement Policy), at Severity Level III. The NRC staff recognizes that the General Foreman was promoted to that position shortly before the violation of 10 CFR 50.7 occurred. Available information indicated that the General Foreman had not received sufficient training in employee protection or maintaining a safety conscious work environment at the time of the promotion. Therefore, the NRC staff concluded, that while the actions of the General Foreman caused NMC and D&Z to be in violation of 10 CFR 50.7, the General Foreman's actions were not deliberate in nature and the NRC is not taking a separate enforcement action against the General Foreman for violating the NRC regulation prohibiting deliberate misconduct, 10 CFR 50.5.

The NRC considered whether credit was warranted for Identification and Corrective Action in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. Credit was warranted for the Identification factor because the violation was identified and investigated by NMC. An investigation was conducted by NMC and NMC notified the NRC of the results of the NMC investigation. Credit was also warranted for the Corrective Action factor. Corrective actions consisted of, but were not limited to: (1) taking disciplinary action against the General Foreman by the employer; (2) reaching a settlement agreement between the employer and the complainant; and (3) conducting surveys of the overall work environment to ensure that no residual effects existed in the safety conscious work environment following the May 4, 2001, employment action. Other actions are described in the previously referenced letter from NMC on May 10, 2004. The NRC acknowledges that the actions by NMC were prompt, comprehensive, addressed the specific situation and the overall work environment for raising safety concerns, and were done without intervention by the NRC.

Therefore, to encourage prompt identification and correction of violations without the intervention of the NRC, I have been authorized, after consulting with the Director, Office of Enforcement, and the Deputy Executive Director for Reactor Programs, to exercise discretion in accordance with Section VII.B.5 of the Enforcement Policy and not issue a Notice of Violation or civil penalty in this matter. Any future violation of 10 CFR 50.7 will be considered for full application of the Enforcement Policy.

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 letter from NMC dated May 10, 2004. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, please provide your response within 30 days of the date of this letter. Your response should be marked "Response to EA-04-038" and sent to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator and Enforcement Officer NRC Region III, and a copy to the Resident Inspector at the Point Beach Nuclear Power Plant. If you contest this enforcement action, you should also provide a copy of your response, with the basis of your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you have any questions, please contact Julio Lara, Chief, Electrical Engineering Branch, at (630) 829-9731.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and your response, if your choose to respond, 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, your response, if you choose to respond, 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/ Geoffrey E. Grant for

James L. Caldwell
Regional Administrator

Dockets No. 50-266; 50-301
Licenses No. DPR-24; DPR-27

cc:

R. Kuester, President and Chief
Executive Officer, We Generation
J. Cowan, Executive Vice President
Chief Nuclear Officer
D[illegible]per, Senior Vice President, Group Operations
D[illegible]ver, Nuclear Asset Manager
Plant Manager
Regulatory Affairs Manager
Training Manager
J. Rogoff, Vice President, Counsel & Secretary
K. Duveneck, Town Chairman
Town of Two Creeks
Chairperson
Public Service Commission of Wisconsin
J. Kitsembel, Electric Division
Public Service Commission of Wisconsin
State Liaison Officer

EA-04-105 - Day and Zimmerman Nuclear Power Systems

September 29, 2004

EA-04-105

Mr. Michael P. McMahon
President
Day and Zimmerman Nuclear Power Systems
1866 Colonial Village Lane, Suite 101
Lancaster, PA 17601

SUBJECT: EXERCISE OF ENFORCEMENT DISCRETION
[NRC OFFICE OF INVESTIGATIONS REPORT NO. 3-2001-033]

Dear Mr. McMahon:

This refers to information provided to the U.S. Nuclear Regulatory Commission (NRC) on July 6, 2001, by a representative of Nuclear Management Company (NMC) concerning alleged employment discrimination at the Point Beach Nuclear Plant. From information obtained during investigations by NMC and the NRC Office of Investigations (OI), an apparent violation of 10 CFR 50.7, "Employee Protection," occurred when a General Foreman, employed by Day and Zimmerman Nuclear Power Systems (D&Z), a contractor at the Point Beach Plant, submitted the name of a D&Z electrician for lay-off on May 4, 2001, following the electrician's protected activities associated with a work package on March 27 and 28, 2001. A copy of the synopsis from the OI report was sent to NMC on April 1, 2004 (NRC Document Control No. ML040920294[1]).

Based on the information developed during the NMC and OI investigations and information contained in a letter from NMC on May 10, 2004 (NRC Document Control No. ML041330278), the NRC concluded that a violation of NRC requirements occurred. Specifically, on March 27 and 28, 2001, a D&Z electrician and a coworker found that the required signatures of the duty shift supervisor and reviewing engineer were missing from a work package. The electrician and a co-worker stopped work on the assigned project and notified a D&Z supervisor of the problem. A coworker of the electrician prepared a Condition Report on the subject. A D&Z General Foreman learned of the electrician's activities on March 27 and 28, 2001, and on March 30, 2001, that General Foreman threatened to terminate the employment of the electrician for stopping work. The General Foreman stated that his intention on March 30, 2001, was not to terminate the electrician or his coworker, but to separate the two workers from each other because of the excessive number of breaks they took. About April 30, 2001, the electrician was told that he would be retained until the end of the project later in the Summer 2001. The General Foreman subsequently submitted the electrician's name for lay-off on May 4, 2001, in part, because the electrician engaged in protected activities on March 27 and 28, 2001. As a result, the General Foreman changed the compensation, terms, conditions, or privileges of the electrician's employment in violation of 10 CFR 50.7. Additionally, the General Foreman allowed two other electricians laid-off on May 4, 2001, to "hover" (remain immediately eligible for reemployment by D&Z without returning to the local union hall for reassignment). However, the General Foreman did not extend the offer to "hover" to the complainant in this matter. The NRC considered the General Foreman to be a first-line supervisor or other low-level manager within the D&Z organization and categorized the violation in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600 (Enforcement Policy), at Severity Level III. The current Enforcement Policy is included on the NRC's Web site at www.nrc.gov:select What We Do, Enforcement, then Enforcement Policy. The NRC staff recognizes that the General Foreman was promoted to that position shortly before the violation of 10 CFR 50.7 occurred. Available information indicated that the General Foreman had not received sufficient training in employee protection or maintaining a safety conscious work environment at the time of the promotion. Therefore, the NRC staff concluded, that while the actions of the General Foreman caused NMC and D&Z to be in violation of 10 CFR 50.7, the General Foreman's actions were not deliberate in nature and the NRC is not taking a separate enforcement action against the General Foreman for violating the NRC regulation prohibiting deliberate misconduct, 10 CFR 50.5.

The NRC is concerned with the initial approach that D&Z management took in this matter. The initial investigation by D&Z was not aggressive and did not identify that employment discrimination had occurred, attributing the reason for the layoff as [redacted]sonality dispute and absenteeism. Upon subsequent investigation by NMC, these reasons were found to be un[redacted]ded and information developed during the NMC investigation indicated that the lay-off was for discriminatory reasons. It is fortuitous for D&Z that NMC conducted a separate, independent investigation and found that employment discrimination was at the root of the issue. Therefore, civil penalty [2] credit for Identification was warranted to NMC because NMC identified and investigated the matter. Had NMC solely relied on the results of the D&Z investigation, and not conducted its own investigation, credit for the Identification factor would not have been possible. Credit for the Corrective Action factor is warranted because of the combined actions of both NMC and D&Z. Corrective actions by D&Z consisted of, but were not limited to: (1) taking disciplinary action against the General Foreman; (2) conducting surveys to ensure that no residual effects existed in the D&Z safety conscious work environment; (3) reaching a settlement between D&Z and the electrician; and (4) establishing a D&Z Employee Advocate Program. Other actions are described in the previously referenced letter from NMC on May 10, 2004. The NRC acknowledges that the corrective actions by D&Z in this matter addressed both the specific issue and the overall work environment for raising safety concerns and were accomplished without intervention by the NRC.

Therefore, to encourage prompt correction of violations, a safety conscious work environment, and resolution of employment discrimination issues without the intervention of the NRC, I have been authorized, after consulting with the Director, Office of Enforcement, and the Deputy Executive Director for Reactor Programs, to exercise discretion in accordance with Section VII.B.5 of the Enforcement Policy to not issue a Notice of Violation in this matter. Any future violation of 10 CFR 50.7 will be considered for full application of the Enforcement Policy.

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 letter from NMC dated May 10, 2004. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, please provide your response within 30 days of the date of this letter. Your response should be marked "Response to EA-04-105" and sent to the U.S. Nuclear Regulatory Commission, ATTN: Do[redacted]ent Control Desk, Washington, DC 20555 with a copy to the Regional Administrator and Enforcement Officer NRC R[redacted]III, and a copy to the Resident Inspector at the Point Beach Nuclear Power Plant. If you contest this enforcement ac[redacted] you should also provide a copy of your response, with the basis of your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you have any questions, please contact Julio Lara, Chief, Electrical Engineering Branch, at (630) 829-9731.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter 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, your response, if you choose to respond, 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/ Geoffrey E. Grant for

James L. Caldwell
Regional Administrator

Enclosure: Letter to NMC

cc[redacted]losure:
D. [redacted], Site Vice President, NMC
G. Van Middlesworth

.Vice President - Nuclear Assessment, NMC
R. Kuester, President and Chief
Executive Officer, We Generation
J. Cowan, Executive Vice President
Chief Nuclear Officer
D. [illegible]er, Senior Vice President, Group Operations
D. Weaver, Nuclear Asset Manager
Plant Manager
Regulatory Affairs Manager
Training Manager
J. Rogoff, Vice President, Counsel & Secretary
K. Duveneck, Town Chairman
Town of Two Creeks
Chairperson
Public Service Commission of Wisconsin
J. Kitsembel, Electric Division
Public Service Commission of Wisconsin
State Liaison Officer

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1. Documents are electronically available from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

2. Civil penalties are not normally considered for contractors of NRC licensees.

EA-04-109 - D.C. Cook (American Electric Power Company)

September 29, 2004

EA-04-109

Mr. M. Nazar
Senior Vice President and
Chief Nuclear Officer
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107

SUBJECT: NOTICE OF VIOLATION
[INSPECTION REPORT 0500315/2004007(DRS); 0500316/2004007(DRS)]

Dear Mr. Nazar:

Th[illegible]fers to information provided to the U.S. Nuclear Regulatory Commission (NRC) by the American Electric Power C[illegible]y (AEP) on March 24, 2004, concerning the permanent physical condition of a licensed Senior Reactor Operator (S[illegible]at the D.C. Cook Nuclear Plant. During a February 2004 review of medical information for licensed operators at the D.C. Cook Nuclear Plant, the new Medical Review Officer for Licensed Operators (MRO) determined that the NRC had not been informed of a cardiac condition experienced by an SRO during December 1996. The failure to provide the NRC with complete and accurate information concerning an SRO's permanent medical condition is an apparent violation of 10 CFR 50.9. A copy of the inspection report concerning this issue was provided to you on July 2, 2004.

In the letter transmitting the inspection report, we provided you the opportunity to address the apparent violation identified in the report by either attending a predecisional enforcement conference or providing a written response before we made our enforcement decision. You responded to the apparent violation in a letter dated August 2, 2004.

Based on the information developed during the inspection and the information you provided in your correspondence on March 24 and August 2, 2004, and during a telephone conversation on August 25, 2004, between Roger D. Lanksbury, Chief, Operator Licensing Branch, and Helen Etheridge of your staff, the NRC has determined that a violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding it are described in detail in the subject inspection report. In summary, the NRC issued an SRO license to the individual on February 1, 1994. On December 28, 1999, AEP submitted information to the NRC to renew the SRO license prior to its expiration on January 31, 2000. Included in the submission for renewal of the SRO license was a December 28, 1999, Form NRC - 396, "Certification of Medical Examination by Facility Licensee." Information on that Form NRC - 396 indicated that your prior MRO recommended only one condition be added to the SRO's license to require the SRO to wear corrective lenses when performing licensed duties. No other medical restriction was recommended by either AEP or the MRO in the December 28, 1999, renewal application. Based on the information submitted by AEP on December 28, 1999, the NRC renewed the SRO license on February 1, 2000, with the requirement that the SRO wear corrective lenses when performing licensed duties. The NRC placed no other medical restrictions on the SRO license based on the information submitted by AEP in the application for renewal.

Th[illegible] who provided the certification discussed above, retired during September 2001 and a new MRO was appointed. Dur[illegible]ebruary 2004, your new MRO reviewed medical records for licensed operators at the D.C. Cook Nuclear Plant.

Included in the new MRO's review were documents indicating that your prior MRO had been informed on January 15, 1997, that the SRO had experienced a myocardial infarction during December 1996. On February 23, 2004, the new MRO notified AEP that the SRO should no longer be allowed to continue to work as a solo operator and the NRC should be notified. That notification was provided to the NRC by AEP on March 24, 2004.

Licensed operators are entrusted with the safe operations of a nuclear reactor and must be capable of performing their assigned duties under normal, abnormal and emergency operating conditions of the plant. The physical condition and the general health of an operator is a significant concern of the NRC so that any sudden incapacitation of an operator due to an existing medical condition does not pose undo risk to the facility. Therefore, the NRC places restrictions for certain medical conditions on an operator's license to ensure that other licensed personnel are on duty and can immediately compensate for an operator who may be suddenly incapacitated because of an existing medical condition. By not informing the NRC of an operator's physical condition, such restrictions cannot be put in place and additional personnel may not be available to replace an operator who is suddenly incapacitated from an existing medical condition.

Furthermore, the information about the SRO's cardiac condition had been known to AEP's MRO since January 15, 1997, and the failure to provide accurate and complete information to the NRC regarding the pre-existing medical condition of a licensed SRO within 30 days, as required by 10 CFR 50.74(c), is a regulatory concern. Moreover, had the medical information been complete and accurate at the time the license renewal was sought by AEP on December 28, 1999, the NRC would have taken a different regulatory position by applying the appropriate restriction to the SRO license. Therefore, the information submitted to the NRC on December 28, 1999, was material to the licensing of an SRO on February 1, 2000, and the violation has been categorized in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, at Severity Level III.

In accordance with the Enforcement Policy, a civil penalty in the base amount of $55,000 would be considered for a Severity Level III violation at the time the violation occurred. Because your facility has not been the subject of escalated enforcement actions evaluated in accordance with the civil penalty assessment process described in Section VI.C.2 of the Enforcement Policy within the last two years, the NRC considered whether credit was warranted for Corrective Action in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. Credit was warranted for the Corrective Action factor. Corrective actions included: preventing activation of the SRO license until the medical status of the operator was resolved; discussing the requirements of ANSI 3.4-1983 with the current MRO and personnel in Operations Training and Regulatory Affairs; requiring all completed physical examination forms and recommendations from physicians be submitted to Regulatory Affairs for inclusion in license applications; and including an overview of the requirements for reporting changes in medical conditions in the operator requalification training program. Other corrective actions included: performing a 100% self-assessment review of licensed operator medical records; revising the procedure to require that all recent physical examination records be submitted to the NRC when requesting an initial or renewal reactor operator or SRO license; and planning by September 30, 2004, to revise the procedure for biennial self-assessment of medical records to discuss the requirements of ANSI 3.4-1983 with the designated MRO.

Therefore, to encourage prompt comprehensive correction of violations and in recognition of the absence of previous escalated enforcement action, I have been authorized, after consultation with the Director, Office of Enforcement, not to propose a civil penalty in this case. However, significant violations in the future could result in a civil penalty.

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 a letter from AEP dated August 2, 2004. Therefore, you are not required to respond to this letter unless the description therein does not accurately 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.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (should you choose to respond) 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, 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/

James L. Caldwell
Regional Administrator

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74

Enclosure: Notice of Violation

cc w/encl:
J. Jensen, Site Vice President
M. Finissi, Plant Manager
G. White, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

NOTICE OF VIOLATION

American Electric Power Company
D.C. Cook Nuclear Power Plant

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74
EA-04-109

During an NRC inspection that was completed on June 4, 2004, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

10 CFR 50.9 requires that information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects.

10 CFR 55.23 requires, in part, that to certify the medical fitness of the applicant, an authorized representative of the facility licensee shall complete and sign Form NRC - 396, "Certification of Medical Examination by Facility Licensee."

Form NRC - 396, when signed by an authorized representative of the facility licensee, certifies that a physician conducted a medical examination of the applicant (as required in 10 CFR 55.21), and that the guidance contained in American National Standards Institute/American Nuclear Society (ANSI/ANS) - 3.4 -1983, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants" was followed in conducting the examination and making the determination of medical qualification.

ANSI/ANS 3.4-1983, Section 5.3.2(1), provides, in part, that certain cardiovascular conditions, including myocardial infarction, preclude solo operation of a nuclear power plant.

10 CFR 55.25 requires, in part, that if, during the term of the license, the licensee develops a permanent physical condition that causes the licensee to fail to meet the requirements of 10 CFR 55.21, the facility licensee shall notify the Commission within 30 days of learning the diagnosis of the condition, in accordance with 10 CFR 50.74(c). 10 CFR 50.74(c) provides, in part, that each licensee shall notify the appropriate Regional Administrator within 30 days of the permanent disability or illness of a licensed operator or senior operator as described in 10 CFR 55.25.

Contrary to the above, on December 28, 1999, the licensee submitted to the NRC a Form NRC 396, an application for renewal of a Senior Reactor Operator (SRO) license, that was not complete and accurate in all material respects. Specifically, the Form NRC 396 certified that the applicant met the medical requirements of ANSI/ANS 3.4 -1983 and that the applicant's only restriction was to require corrective lenses be worn when performing licensed duties. During December 1996, the SRO developed a permanent physical condition which did not meet the minimum cardiovascular standards specified in ANSI/ANS -3.4 -1983, Section 5.3.2(1) and which precluded the SRO from " solo" operation of a nuclear power plant. This information was material to the NRC because the NRC relied on the information contained in the Form NRC 396 dated December 28, 1999, to determine whether the applicant met the requirements of 10 CFR Part 55 to operate the controls of a nuclear power plant and to renew the SRO's license on February 1, 2000. In addition, the facility licensee was provided on January 15, 1997, with information about the SRO's December 1996 myocardial infarction, but did not notify NRC of the SRO's physical condition until March 24, 2004, a period in excess of 30 days after learning of the SRO's physical condition.

This is a Severity Level III violation (Supplement VII).

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 will be achieved, is already adequately addressed on the docket in letter from the Licensee dated August 2, 2004. 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-04-109," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator and the Enforcement Officer, Region III, and a copy to the NRC Resident Inspector at the D.C. Cook Nuclear Plant, 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, United States 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), 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 29th day of September 2004.

Inside NRC
Volume 26 / Number 22/ November 1, 2004

ACRS criticizes industry PWR sump methodology, NRC evaluation

The industry's proposed PWR sump evaluation methodology and NRC's associated safety review are riddled with technical flaws, NRC's Advisory Committee on Reactor Safeguards (ACRS) said last month in an eight-page letter report.

NRC staff is finalizing its safety evaluation of the methodology developed by the Nuclear Energy Institute (NEI). NRC and industry anticipate that most PWR operators will use the methodology to respond to a Sept. 13 generic letter requesting that licensees evaluate and, if necessary, upgrade containment sumps to ensure their continued operability despite potential debris accumulation after a loss-of-coolant accident (INRC, 20 Sept., 5).

NRC staff released its draft safety evaluation (SE) in September and had hoped to issue the final version by the end of October. However, the ACRS expressed strong objections in an Oct. 18 letter." The SE should not be issued in its present form" because "both it and the NEI guidance contain too many technical faults and limitations to provide the basis for a defensible and robust longterm solution" to PWR sump safety issues, ACRS Chairman Mario Bonaca said in the letter to NRC Chairman Nils Diaz. "It is our judgment that too many gaps remain for a technically defensible resolution at this time," Bonaca said. "Some basic methods" in NEI's evaluation methodology "include equations that contain incorrect physical descriptions of the phenomena," Bonaca said. "There are also questions about the extent of the supporting test data and the data's interpretation, and the guidance on implementing the proposed methods is vague and contains inconsistencies," he concluded.

ACRS raised technical objections to NEI's proposed methods for calculating debris generation and accumulation on sump screens, modeling of debris transport, and estimation of the contribution of containment coatings to sump blockage. Also, "no definite guidance is provided for evaluating either chemical or downstream effects, both of which may become of major importance as more knowledge is acquired," Bonaca said.

In additional comments appended to the letter, ACRS members Graham Wallis and Peter Ford jointly suggested that "in the short term, there may be some practical actions that can be explored" to address PWR sump safety issues while NEI's methodology is improved. NRC should "encourage licensees to pursue, at an early stage, corrective actions that will be as independent as possible of known model uncertainties," Wallis and Ford said. They cited several examples, including removal from PWR containments of materials known to create "particularly detrimental sump blockage results in tests" and "testing alternative filtering devices, such as debris catchers and active screens."

In separate comments, Wallis suggested that NRC staff consider breaking up the process "into a set of phases, each of which is based on the best available methods, results that are clearly evident despite uncertainties, and decisions that can be implemented within a realistic schedule."

The ACRS letter "doesn't do any service to industry's methodology" or NRC's safety evaluation, Anthony Pietrangelo, senior director for risk regulation at NEI, said at NEI's annual licensing forum Oct. 20. ACRS's technical objections are "largely a lot of academic concerns" that fail to recognize that NRC and industry can "factor new information back in" to the evaluation process as it becomes available, Pietrangelo said.

NRC staff "is reviewing and will disposition each issue the ACRS raises," Brian Sheron, associate director for project licensing and technical analysis in NRC's Office of Nuclear Reactor Regulation (NRR), said at the forum. "The question is, are they showstoppers or can they be resolved in parallel" with efforts to evaluate and upgrade sumps, Sheron said, adding that "NRC makes decisions based on incomplete information all the time."

NRC staff is preparing a response to the ACRS letter and will reply before the SE is finalized, NRR's David Solorio told Inside NRC last week. As a result, the final SE "will be slightly delayed," he said. Staff may require "a few more days or a week or two" to complete its response to ACRS, which must then be approved by upper management, Solorio said.
—Steven Dolley, Washington

Exelon reviews uprate effects on Dresden and Quad Cities

Exelon Nuclear has determined that full-cycle operation at extended power uprate levels will not jeopardize plant components at Dresden and Quad Cities, the company said last week.

Exelon plans to install new steam dryers during refueling outages at Quad Cities-1 in April and Quad Cities-2 in May, James Meister, Exelon vice president for nuclear services, told NRC staff at an Oct. 25 meeting. Steam dryers in both units experienced cracking and failed in 2002 and 2003 after Quad Cities began operating at a 17.8% extended power uprate (EPU) level (INRC, 28 July '03, 6). Both Quad Cities units have since operated at their lower, previously authorized power ratings.

The new dryers are modifications of systems designed for General Electric's Advanced Boiling Water Reactor design. Scale model tests are nearly complete and will compare damage observed in the cracking incidents with various power output levels, Meister said. The dryer design will be finalized by mid-November and final analyses completed by the end of the year, he said. Dryer assemblies are now being fabricated and the main structures will be completed by January.

After the dryers are installed next spring, the Quad Cities reactors will be brought up "several power plateaus" to test the design and monitoring instrumentation, Meister said. Steam dryers at Dresden-3 will be inspected during the refueling outage that began Oct. 26, and Exelon will "preemptively implement" modifications to prevent the type of damage experienced at Quad Cities, Meister said.

Increased vibration from EPU operation is believed to have contributed to the steam dryer failures. Exelon subsequently conducted a vibration evaluation "to provide assurance that potentially affected components would perform acceptably at EPU full thermal power operation," Sharon Eldridge of Exelon's engineering division said at the meeting. Evaluation and testing are completed for all components except one safety/relief valve, Eldridge said.

The vibration evaluation concluded that "all components are acceptable as originally designed" for operation at full EPU power except for certain valves which required upgrades, Eldridge said. Vibration endurance testing of the new valves is complete and "provides assurance of full-cycle operation with only inconsequential wear of the affected components," she said.

"Implementation of actions is either planned or complete" to support return of Quad Cities to full uprated power levels, Eldridge said. She did not indicate when the units would increase power to EPU levels. The evaluations also support continuation of full EPU power operation at both Dresden units, she said.

Exelon also conducted a broader vulnerability review of Quad Cities and Dresden to "identify potential EPU-related failures" and take "actions to prevent failures induced by those vulnerabilities," said Mohammad Molaei, Dresden engineering programs manager. A total of 42 power systems and 10 safety systems were reviewed, Molaei said. The assessment analyzed among other issues "components susceptible to increased vibration due to increased feedwater flow" of about 17% at EPU power levels, he said.

The assessment concluded that "functions of safety systems remain uncompromised" at EPU power levels, Molaei said. Exelon "found no vulnerabilities" of power systems "that posed an immediate challenge to plant operation," but 101 actions for each unit were identified to "improve operating margin and prevent future failures," he said.

"Most of the actions address accelerated equipment aging or wear due to EPU, rather than something uniquely caused by" operation at EPU power levels, Molaei said. "Safe and reliable operation is achievable for Dresden and Quad Cities without any risk to plant safety or any challenge to balanceof-plant systems," he said. "Considerable knowledge was gained during the review on impact of EPU operation and was shared with the industry" through the BWR Owners Group and the Institute of Nuclear Power Operations, Meister said.

EPU impacts on license renewal

On Oct. 28, NRC approved 20-year extensions of operating licenses for Dresden and Quad Cities (see related story, p. 4). In September, the Advisory Committee on Reactor Safeguards (ACRS) had recommended that NRC staff and Exelon include steam dryers in the scope of the license renewal review for Dresden and Quad Cities (INRC, 4 Oct. 5).

In an Oct. 8 letter to NRC, Exelon agreed to include steam dryers in license renewal scope. Exelon will use the BWR Vessel and Internals Project steam dryer inspection and evaluation guideline or another NRC-approved approach as its aging management program, Keith Jury, Exelon nuclear director of licensing and regulatory affairs, said in the letter. Exelon will also "perform an evaluation of operating experience at EPU levels prior to the period of extended operation to ensure that operating experience at EPU levels is properly addressed by the aging management programs," Jury said. Steam dryer problems experienced at Dresden and Quad Cities are "very much in the forefront of the staff's mind for other units interested in EPUs," Brian Sheron, associate director for project licensing and technical analysis in NRC's Office of Nuclear Reactor Regulation, said at last month's meeting.

NRC staff agrees with ACRS that "power uprates could affect aging management programs" at plants seeking to extend their operating lives and is "putting together a plan on how to address that," Sheron told Inside NRC.

—Steven Dolley, Washington

Panelists differ on wisdom of folding component aging into PRAs

The question of how aging passive components should be treated in probabilistic risk assessments (PRAs)—and how valuable or appropriate it is to include aging in PRAs—drew a range of responses from a panel of experts at the NRC's Nuclear Safety Research Conference last week.

Karl Fleming of Technology Insights in San Diego said current PRAs need to be improved to support risk-informed decision making on aging-related issues. In the Oct. 25 session, Mohammad Modarres of the University of Maryland's Center for Technology Risk Studies agreed that PRAs should be expanded to take such issues into account.

But Kenneth Balkey of Westinghouse Electric Co. said that rather than incorporating the aging of passive components into PRAs, operators should keep failure rates low enough so "they don't get into PRAs." It would be "premature" to require modeling of materials aging in plant PRA models, although such models are needed for research, he said. Expressing a somewhat similar view, **William Shack of Argonne National Laboratory and NRC's Advisory Committee on Reactor Safeguards,** said that because predicting passive component failure is so complex, it is better addressed in the development of an aging management program rather than in plant PRAs.

NRC officials at the session wondered how they should interpret that variety of views in carrying out their jobs as regulators. Robert Tregoning of the Office of Nuclear Regulatory Research asked, "Isn't it almost incumbent on us to try to understand the effects of aging?"
Modarres said that since PRAs are "about predicting the future," it would be a "gross underestimation" not to include the aging of passive components. Fleming said the answer to Tregoning's question depends in part "on what the PRA is used for, what decision we are trying to make."
Balkey said that while he was not arguing that passive components should never be included, more work needs to be done on plant PRA models. Introducing the issue at this point would be "overwhelming," he said.
Shack said that for him the "first order of business" would be to try to minimize the problem through aging management programs. But if after such programs the aging is still significant enough to have an effect on PRAs, it should be taken into account, he said. It is a "question of emphasis," he said.—***Daniel Horner, Washington***

(2)

Risk-Informing 50.46
ECCS Acceptance Criteria

Briefing for ACRS

Brian Sheron, ADPT-NRR

November 4, 2004

Meeting Objective

- To receive letter from the ACRS in November endorsing release of the proposed rule for public comment

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Background

- July 04 SRM directed staff to risk inform LBLOCA requirements
- Proposed rule should be completed in six months
- ACRS briefed in July on conceptual approach
- Public meeting held in Aug to get inputs for regulatory analysis (costs/benefits)
- CRGR review deferred until final rule stage

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Rule Change Objectives

- Focus resources on more risk significant issues
- Expect licensees to reduce plant risk through optimization of safety systems operation
- Other proposed plant changes should not result in any significant risk increases

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Potential Safety Improvements

- Adjust containment spray timing and flow
  - Conserve RWST inventory
  - Reduce debris wash down and threat to sump NPSH
  - Extend time for manual switchover to recirculation
- Improve EDG reliability
  - Longer start times
  - Less demanding load sequencing
- Adjust accumulator setpoints
  - Better inventory control for more likely LOCAS

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Potential Safety Benefits (con't)

- Adjust LPSI setpoints to minimize time in mini flow operation
- Adjust system resistances to improve operation for more likely breaks
- Modify core design to reduce vessel fluence

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Today's Presentations

- Overview of Proposed Rule and Conforming Changes
- ECCS Analysis Requirements
- Process for Approval of Plant Changes based upon new DBA LOCA

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Schedule Forward

- Complete SOC in November
- Receive ACRS endorsement letter in November
- Proposed Rule Package to EDO - December
- Package to Commission by end of December
- Draft Regulatory Guide in June 2005

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③

Regulatory Structure of Proposed Rule

Risk-Informed 10 CFR 50.46

Richard Dudley, NRR Rulemaking Section
U. S. Nuclear Regulatory Commission
November 4, 2004

Draft Rule Structure

- Existing § 50.46 essentially unchanged
- Voluntary alternative rule added (§ 50.46a)
- Minor conforming changes to:
  - §50.34 – Contents of applications
  - §50.109 – Backfit rule
- Other conforming changes in some GDC

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Draft Rule Structure (§ 50.46a)

- LOCA break spectrum divided into 2 regions by "transition" break size (TBS)
  - based upon frequency and other considerations
- Breaks in smaller break region continue to be DBAs; must meet current § 50.46 requirements
- Breaks larger than TBS become beyond design-basis accidents, but mitigation capability must be demonstrated up to full DEGB
  - less stringent analysis assumptions/acceptance criteria
  - demonstrate for all at-power operating configurations
- TBS break conditions apply to certain other requirements based upon LOCA attributes

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Plant Changes Under § 50.46a

- After new ECCS analysis, some plant designs no longer limited by DEGB of largest pipe
- Changes proposed to plant operations or design must either be approved by NRC license amendment or meet "inconsequential risk" criteria
- License amendment submittals must be risk-informed
  - Meet criteria consistent with RG 1.174 (defense-in-depth, safety margins, monitoring program, **and acceptable risk**)
  - Meet PRA quality and scope requirements

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Changes to General Design Criteria

- Conforming changes to some GDC to avoid conflicting requirements
- Remove single failure requirement:
  - GDC 17 – Electric Power Sustems
  - GDC 35 – Emergency Core Cooling
  - GDC 38 – Containment Heat Removal
  - GDC 41 – Containment Atmosphere Cleanup
  - GDC 44 – Cooling Water
- No changes to GDC 4 and GDC 50

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Inconsequential Risk Plant Changes

- Licensees allowed to make "inconsequential" risk plant changes without specific NRC review
- Licensee submits PRA and review process
- PRA must meet acceptance criteria; licensee review process must ensure defense-in-depth and safety margins
- NRC approves and modifies license to authorize licensee to make future "inconsequential" changes

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Design Change Licensing Process

- Licensees submit design changes as risk-informed license amendments
- NRC review and approval to ensure compliance with acceptance criteria
- NRC evaluates possible security impacts during amendment review process

LOCA Frequency Reevaluations

- NRC periodically evaluates LOCA frequency information
- If significantly increases, NRC will change transition break size (rulemaking or order)
- Plant design changes must continue to meet acceptance criteria
- Licensees must restore design or make compensatory changes to meet acceptance criteria
- Backfit rule (10 CFR 50.109) does not apply

(4)

ECCS Analysis Requirements

Ralph Landry, Reactor Systems Branch
U. S. Nuclear Regulatory Commission
November 4, 2004

Transition Break Size

- TBS for PWRs – Falls above SBLOCA to LBLOCA transition.
  - SBLOCA dominated by two-phase level swell
  - LBLOCA dominated by dispersed flow film boiling
- TBS for BWRs
  - Automatic Depressurization System effect

Analysis Requirements 50.46a

- ≤ TBS
  - Approved methodology
  - Worst single-failure
  - LOOP
  - Safety systems only
  - Limiting tech. specs and operational characteristics
  - Limiting break size and location

No change

- > TBS
  - Approved methodology*
  - No single failure prescribed**
  - Non-safety equipment may be credited
  - Nominal tech specs and operational characteristics
  - Limiting break size and location

*review focused on only the most important phenomena for evaluation models in the > TBS region

**only analyzed operating configurations permitted

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Acceptance Criteria

- ≤ TBS
  - PCT ≤ 2200°F
  - MLO ≤ 17%
  - CWO ≤ 1%
  - Coolable Geometry
  - Long-term cooling
- > TBS
  - Coolable Geometry
  - Long-term cooling

NRC review

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Documentation Requirements

- ≤ TBS
  - Code documentation as currently required under 10 CFR Part 50, Appendix K, II, sufficient to demonstrate with **high probability** performance criteria would not be exceeded.
- > TBS
  - Code documentation sufficient to demonstrate that the performance criteria will not be exceeded.

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Reporting Requirements

- ≤ TBS
  - ΔPCT > 50°F; or
  - ΔMLO > 0.4% ⇒ MLO = *f* {T, time at T}
- > TBS
  - ΔPCT > 300°F

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Regulatory Review

- Review of evaluation models applicable in the beyond TBS region will focus on adequacy of evaluation model to represent the most important parameters.
  - Regulatory Guide

(5)

Risk-Informed Evaluation of the Acceptability of Proposed Plant Modifications

Glenn Kelly

Probabilistic Safety Assessment Branch, DSSA, NRR

November 4, 2004

1

Define The Proposed Change

- Define what will be affected in plant's design or licensing basis including licensing conditions and commitments
- Identify all SSCs, procedures, and activities to be changed or affected
- The totality of changes made under 50.46a are evaluated as a single change for purposes of tracking changes to risk

2

Define The Proposed Change (Cont)

- This conforms to RG 1.174 guidance for a combined change request, where contributors should be reviewed for overall risk effect if they impact the same plant functions.
- Because of the potential risk and regulatory significance of 50.46a changes, we are proposing in the draft rule that "in total" the changes meet the risk acceptance criteria, and are tracked as a group.
- This would allow tradeoffs of "safety benefits" versus "risk increases" that are related to 50.46a-allowed modifications.

3

Define The Proposed Change (Cont)

– This would serve as an incentive for industry to identify and implement safety benefits as part of this rule.

– However, the staff will consider other options that do not discourage implementation of unrelated changes that have a net safety benefit; To accomplish this the staff will explore additional criteria that could provide "bundling" flexibilities as part of the development of the RG.

4

Two Change Processes

- Draft Rule permits two plant change processes:

(1) "Normal" risk-informed Licensing Action Request review process available for any proposed changes

(2) Licensee may apply for approval to make inconsequential future changes without staff prior review and approval
  - Initial application required to demonstrate capability of evaluation processes and tools used to determine that acceptance criteria of rule remain satisfied
  - May limit initial staff review by limiting scope of future changes
  - Cumulative change in risk from all unreviewed changes must remain inconsequential

- Licensee's evaluation process is the same for all changes

5

Defense-in-depth Coolable Geometry

- To maintain defense-in-depth, the plant cannot enter or operate in a configuration unless it has been shown that in the event of a LOCA larger than the TBS a coolable geometry could be assured.

- This may place some limits on power uprates or operation configurations, because the analyses would need to account for major SSCs out of service for maintenance.

6

Defense-in-Depth Containment Performance

- Changes to containment systems will be allowed by the rule
- Some changes to containment systems will not affect CDF or LERF estimates, but could still change the likelihood of a large release
- Late containment failure and late release are qualitatively evaluated as part of defense-in-depth in risk-Informed licensing actions
- Late release frequency (LRF) was added to the CDF and LERF guidelines to provide a structured evaluation process and consistent acceptance criteria

7

Numerical Risk Criteria

- Rule requires that the total risk increase of all changes be estimated and be sufficiently small
  - It is expected that the effect of the changes proposed can be measured quantitatively and in a realistic manner
  - Estimates using risk assessments other than PRA are permitted (qualitative, bounding, screening, etc)
  - If proposed changes are not modeled, then they should be modeled, or it should be demonstrated that the change has no, or only a very small negative effect on CDF, LERF, and LRF.
- Numerical criteria for CDF and LERF based on principles and expectations of R.G. 1.174
- Guidance for LRF will be developed as part of the planned RG

8

PRA Technical Adequacy

- PRA will be assessed by NRC taking into account standards and peer review results (see trial use R.G. 1.200)
- PRA must be able to calculate mean CDF, LERF, and LRF
- Meeting NRC approved standards should reduce the NRC resources needed to review
- Phased approach to quality of PRA's endorsed by the Commission

9

Risk Assessment Technical Adequacy

- Plants using risk assessment methods other than PRA's would need to:
  - Justify methods produce realistically conservative numerical results and appropriate safety insights, or
  - Justify method is capable of accurately determining expected changes in CDF, LERF, and LRF or
  - Justify the absence of PRA modeling for this initiator would make no significant difference in numerical results and insights.

10

Implementation and Monitoring

- The updated PRA must retain sufficient technical adequacy to demonstrate that the acceptance criteria are met
- To provide confidence in the technical adequacy of the updated PRA, the licensee must report
  - If the baseline CDF increases by 20% or more after an update
  - If the baseline LERF increases by 20% or more after an update
  - If the change in CDF from 50.46a implementation increases by more than 1E-6/year
  - If the change in LERF from 50.46a implementation increases by more than 1E-7/year

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Proactive Materials Degradation Assessment

November 4, 2004
Rockville, MD

Dr. Joseph Muscara
Senior Technical Advisor for Materials Engineering Issues
US NRC Office of Nuclear Regulatory Research
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Proactive Materials Degradation Assessment

Background

- Materials degradation has been experienced in nuclear reactor components since almost the inception of power plant operations
  - Steam generator tube degradation
  - BWR pipe cracking
  - VC Summer hot leg cracking
  - Oconee vessel head penetration cracking
  - Davis-Besse vessel head degradation

- NRC and Industry have responded to occurrences as they have been discovered
  - Actions taken to maintain safety and reliability
  - Solutions developed have occasionally led to new problems

Proactive Materials Degradation Assessment

Motivation

- Reactive approaches to dealing with materials degradation problems have been inefficient
  - Increased financial and manpower burden
  - Compromise regulatory effectiveness and efficiency
  - Potential to erode public confidence

- NRC/RES decided to take a proactive approach to materials degradation assessment
  - Develop a foundation for appropriate actions to keep materials degradation from adversely impacting safety
  - Want to avoid surprises, so need to think in broader terms

Proactive Materials Degradation Assessment

Scope

- What is proactive with respect to materials degradation?
  - Predict and avoid
  - Predict, monitor, and repair
- Prediction is a critical aspect of PMDA
- Maintain component reliability, public confidence, and avoid surprises
  - Avoid release of radioactivity anywhere in the plant
  - Avoid failure of safety significant components
  - Hundreds/thousands of components need to be considered
- Consider risk importance of components susceptible to degradation
  - Prioritize research efforts
  - Develop regulatory guidance

Proactive Materials Degradation Assessment

Approach

- First step is to identify materials and locations where degradation can reasonably be expected in the future
- Next step is to develop and implement a research program for the components and degradation of interest that will review, evaluate, and develop as needed:
  - Inservice inspection and continuous monitoring techniques for the detection, characterization, and evaluation of degradation
  - Techniques to ameliorate stressors for mitigation or prevention of expected degradation
  - Materials for repair or replacement
  - Repair and replacement techniques
  - Post-repair and fabrication inspection techniques
- Research program will consider ongoing international research, and address gaining a better understanding of current and potentially new degradation mechanisms and dependencies

Proactive Materials Degradation Assessment

Identify Components of Interest

- Two activities to accomplish the first step
  - Existing information to identify components that have experienced degradation
    - Performed in short term with relatively quick results
  - Phenomena Identification and Ranking Table (PIRT) process to identify plant components susceptible to future degradation
    - Longer term, structured approach

Proactive Materials Degradation Assessment

Identify Components of Interest – Initial Studies

- Identified components that have already experienced degradation
  - Lead contractor is Pacific Northwest National Laboratory
  - Week-long workshop held at Argonne National Laboratory
  - Utilized various sources of operating experience
    - Aging studies such as Generic Aging Lessons Learned (GALL) reports
    - Licensee Event Reports (LERs)
    - INPO database: EPIX
- Evaluating current inspection and leak monitoring techniques and requirements for timely detection of degradation in the components of interest
  - Performance demonstration
  - Probability of detection
  - Inspection Methods
    - Periodic, Continuous monitoring
  - Risk Informed Inservice Inspection (RI-ISI)

Proactive Materials Degradation Assessment

Identify Components of Interest – Initial Studies (Cont.)

- Determine Conditional Core Damage Frequency (CCDF) for components whose inspection requirements need to be improved
- Probabilities of failure for future detailed Probabilistic Risk Assessments (PRAs)
  - Collect from existing information (FY05)
    - RI-ISI
    - LOCA frequency studies
  - Perform specific component analyses in the future (FY06)
    - Probabilistic fracture mechanics analyses
    - Piping failure/population databases

Proactive Materials Degradation Assessment

Identify Components of Interest – Longer Term Activity

- Expert elicitation only feasible approach to identifying components susceptible to future degradation
  - Analytically requires too much time/funding/data

- PIRT-like process identified as best method for expert elicitation
  - Key PIRT qualities
    - Structured expert elicitation
    - Phenomena identification and quantitative scoring of responses
    - Continuous documentation of results
  - 8-member international expert panel: Materials/corrosion, systems, operational experience
  - 8 week-long meetings over one year period
  - Provide background information to panel on materials, stressors, function of components, operating experience
  - Develop lists/reports of PWR and BWR components with associated degradation phenomena including the bases for the findings
  - International peer review of results

Proactive Materials Degradation Assessment

Identify Components of Interest – Longer Term Activity (Cont.)

- Important plant systems being addressed

- PWR systems
  - Reactor Coolant System
    - Reactor Pressure Vessel and internals
    - Steam Generators
    - Pressurizer
    - Reactor Coolant Pump
  - Emergency Core Cooling System
  - Auxiliary Feedwater System*
  - Steam Generator Blowdown
  - Chemical Volume and Control System
  - Component Cooling Water
  - Service Water System*
  - Feedwater System*
  - Residual Heat Removal
  - Main Steam*
  - Spent Fuel Storage/Cooling/Cleanup

*Safety significant portions only

- BWR systems
  - Reactor Coolant System
    - Reactor Pressure Vessel and internals
    - Recirculation Pumps
  - Low Pressure Core Spray Core Injection Systems (HPCI, RCIC)
  - Residual Heat Removal
  - Control Rod Drive System
  - Service Water System
  - Component Cooling Water
  - Reactor Water Cleanup
  - Suppression Pool Cleanup
  - Spent Fuel Storage/Cooling/Cleanup
  - Keep Fill System
  - Main Steam System
  - Feedwater System
  - Condensate System
  - Extraction Steam System

Proactive Materials Degradation Assessment

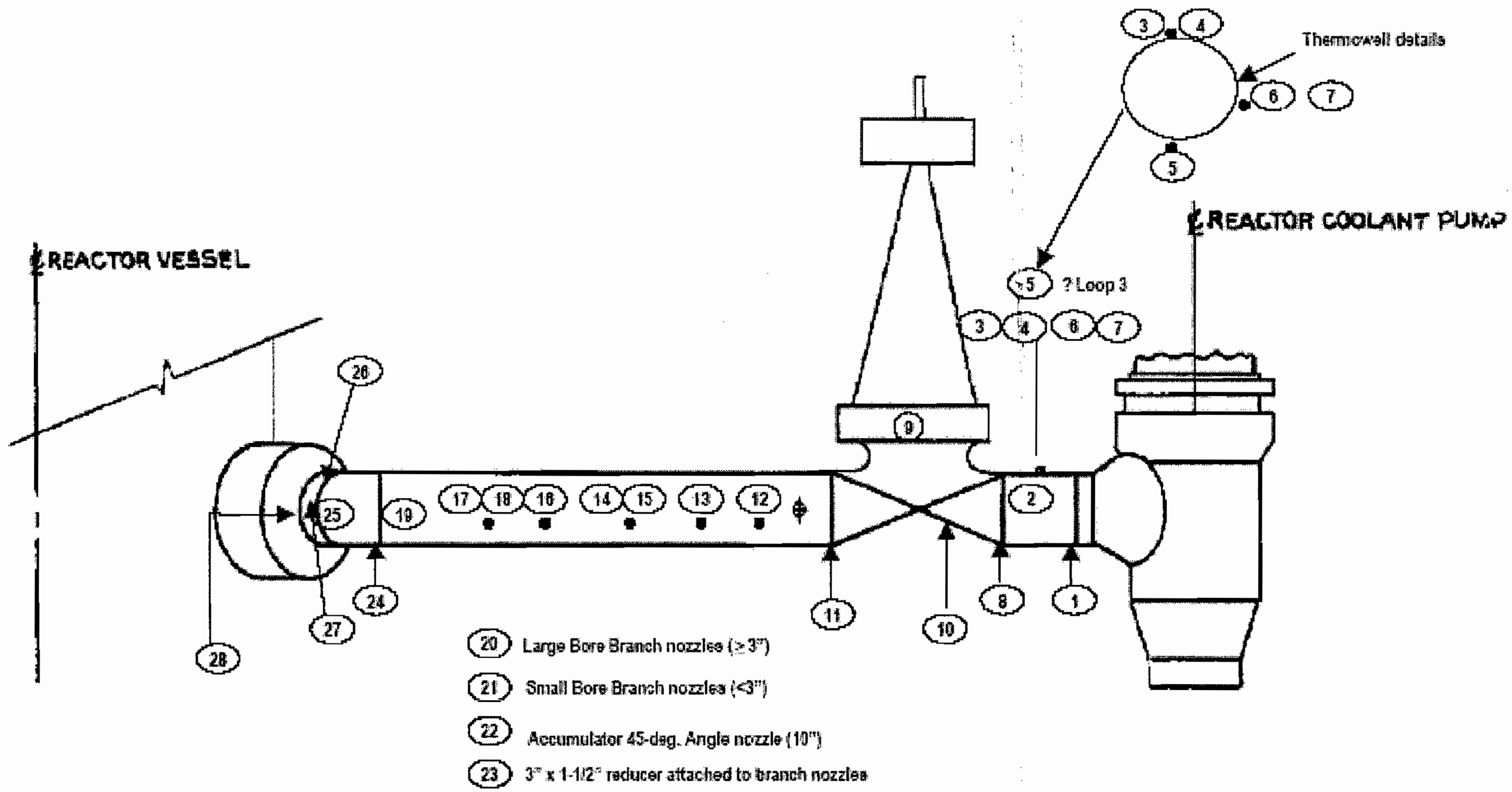
Identify Components of Interest – Longer Term Activity (Cont.)

- Background information collected by Brookhaven National Laboratory (BNL)
  - Components derived from systems of interest
    - "Component" is a continuous portion of the system that is of the same material and product form, and experiences similar "stressors"
      - Temperature, Pressure, Residual stress level, Fatigue cycles, Irradiation, Water chemistry, etc.
    - Multiple components of the same material that experience similar stressors are agglomerated
    - Partially developed from piping population database, PIPExp, licensed from Bengt Lydell and supplemented by plant drawings
  - Operational experience included with each component
    - Same sources used as for short term activity
      - GALL reports
      - LERs
      - EPIX
      - Presentations by NRC Technical Training Center staff
- Components and background information provided to Expert Panel

Proactive Materials Degradation Assessment

Identify Components of Interest – Longer Term Activity (Cont.)

GROUP 1: RCS Cold Leg Piping
(Covers worksheets RCS-CL – 1 thru 28)



Proactive Materials Degradation Assessment

Identify Components of Interest – Longer Term Activity (Cont.)

1	System Identification	Group Identification	Part ID	Part No.	Part Description	Part Size in inches	Part Thickness in inches	Material A	Material W	Material B	Weld Type	Operating Temp in °F
2	Reactor Coolant System (RCS)	Group 1 - RCS Cold Leg Piping (CL)	RCS-CL-	1	RCP DISCHARGE NOZZLE - 27.5" CL PIPE	27.5	2.21"MW	SA351 GR.CF8 (CASTING)	SS TP 308	SA376 GR.TP304N (SMLS PIPE)	Field	556 to 559
3	Reactor Coolant System (RCS)	Group 1 - RCS Cold Leg Piping (CL)	RCS-CL-	2	27.5" CL PIPE	27.5	2.21"MW	SA376 GR.TP304N (SMLS PIPE)		Not Applicable		556 to 559
4	Reactor Coolant System (RCS)	Group 1 - RCS Cold Leg Piping (CL)	RCS-CL-	3	27.5" CL PIPE - 2" SWEEPOLET	2	0.344"	SA376 GR.TP304N	SS TP 308	SA182 GR.F316N	Shop	556 to 559
5	Reactor Coolant System (RCS)	Group 1 - RCS Cold Leg Piping (CL)	RCS-CL-	4	BRANCH CONNECTION - THERMOWELL	2	0.375"	SA182 GR.F316N	SS TP 308	SA479 GR.TP316		556 to 559
6	Reactor Coolant System (RCS)	Group 1 - RCS Cold Leg Piping (CL)	RCS-CL-	5	27.5" CL PIPE - 2.5" OD THERMOWELL BOSS	2.5	0.375"	SA376 GR.TP304N (SMLS PIPE)	SS TP 308	SA182 GR.F316N		556 to 559
7	Reactor Coolant System (RCS)	Group 1 - RCS Cold Leg Piping (CL)	RCS-CL-	6	27.5" CL PIPE - 2.5" OD THERMOWELL BOSS	2.5	0.375"	SA376 GR.TP304N	SS TP 308	SA182 GR.F316N	Shop	556 to 559
8	Reactor Coolant System (RCS)	Group 1 - RCS Cold Leg Piping (CL)	RCS-CL-	7	BRANCH CONNECTION - THERMOWELL	2.5	0.375"	SA182 GR.F316N	SS TP 308	SA479 GR.TP316		556 to 559
9	Reactor Coolant System (RCS)	Group 1 - RCS Cold Leg Piping (CL)	RCS-CL-	8	27.5" CL PIPE - STOP VALVE 1RC8002A	27.5	2.21"MW	SA376 GR.TP304N	SS TP 308	SA351 GR.CF8M	Field	556 to 559
	Reactor Coolant	Group 1 - RCS Cold	RCS-CL-	9	STOP VALVE BODY	27.5		SA351		Not		556 to 559

Proactive Materials Degradation Assessment

Identify Components of Interest – Longer Term Activity (Cont.)

1	Oper. Press. in psi	Operating Flow	Design Temp in °F	Design Press in psi	Design Flow	Inside Environ	Outside Environ	Residual Stress in ksi	Normal Stress in ksi	Faulted Stress in ksi	CUF	Stress Comments	Operating Experience
2	2250	35 M#/HR	650	2485	35 M#/HR	Reactor Coolant	Containment Air		10.5	32.77		Stress= pressure+ deadweight + thermal. Note, stainless steel weld metals also susceptible to thermal aging, but will not age as badly as high ferrite number static casting.	
3	2250	35 M#/HR	650	2485	35 M#/HR	Reactor Coolant	Containment Air					SAME AS PART 19; Note in some Westinghouse plants this could be a cast stainless pipe. CF8A pipe is less susceptible to thermal aging than CF8M used in some other Westinghouse plants.	IN 86-108 BORIC ACID CORROSION IN A Carbon steel NOZZLE WELDED TO RCS PIPING. Also, EPIX-245: leak in the base metal of the outer radius of a 1 1/2 inch 60 degree elbow due to thermal fatigue
4	2250	35 M#/HR	650	2485	35 M#/HR	Reactor Coolant	Containment Air						
5	2250	35 M#/HR	650	2485	35 M#/HR	Reactor Coolant	Containment Air						
6	2250	35 M#/HR	650	2485	35 M#/HR	Reactor Coolant	Containment Air						
7	2250	35 M#/HR	650	2485	35 M#/HR	Reactor Coolant	Containment Air						
8	2250	35 M#/HR	650	2485	35 M#/HR	Reactor Coolant	Containment Air						
9	2250	35 M#/HR	650	2485	35 M#/HR	Reactor Coolant	Containment Air		10.3	30.51			
	2250	35 M#/HR	650	2485	35	Reactor	Containment						

Proactive Materials Degradation Assessment

Identify Components of Interest – Longer Term Activity (Cont.)

- First two Expert Panel meetings have been completed
  - PWR systems examined
    - Reactor Coolant System
    - Emergency Core Cooling System
- Panel Experts agglomerated components according to degradation expected (based on data, personal experience and knowledge)
  - RCS: 315 total components reduced to 88 sub-groups for identification of applicable degradation mechanisms, if any
- Experts assign numerical values to three parameters in the evaluation of potential degradation expected for a given component, and provide bases for their decisions
  - Susceptibility Factor - can significant material degradation develop given plausible conditions?
    - blank=not considered to be an issue
    - 1=conceptual basis for concern from data, or potential problems under unusual operating conditions, etc.
    - 2=strong basis for concern or known but limited plant problem
    - 3=demonstrated, compelling problem or multiple plant observations

Proactive Materials Degradation Assessment

Identify Components of Interest – Longer Term Activity (Cont.)

- Experts assign values (Continued)
  - Confidence Level - personal confidence in our judgment of susceptibility
    - 1=low confidence, little known about phenomenon;
    - 2=moderate confidence;
    - 3=high confidence, compelling evidence, existing problems
    - Note, "3" is assumed if Susceptibility Factor is "blank"
  - Knowledge Level - extent to which the relevant dependencies have been quantified
    - 1=poor understanding, little and/or low-confidence data;
    - 2=some reasonable basis to know dependencies qualitatively or semi-quantitatively from data or extrapolation in similar "systems";
    - 3=extensive, consistent data covering all dependencies relevant to the component, perhaps with models -- should provide clear insights into mitigation or management of problem
- Consensus on values not required

Proactive Materials Degradation Assessment

Identify Components of Interest – Longer Term Activity (Cont.)

Group 1 RCS Cold Leg Piping

Identification	Material & Environment combination at Full power temperature/pressure	Degradation mechanisms considered	Susceptibility	Confidence	Knowledge	Rationale for scoring	Critical factors controlling occurrence in plant	Components in this sub-group
			1=low, 2=med, 3=high					
1.1	All stainless steel components	TGSCC	1	3	3	Well known phenomenon. Cl from insulation and aerosols, the latter increasing with time	Concern only if wet. Tolerance level for Cl depends on buffer availability from insulation	All
	External surfaces when at <150C	PIT	1	3	3			
	Normally dry when at low temp							
1.2	Wrought austenitic stainless steel piping	CF	1	3	3	Good lab data base but uncertainty on accounting for magnitude of environmental effects	Very good field experience. Only likely to be a problem where present design rules give CUF>0.1approx.	2,19
	Types 304,316, PWR primary							
	556 to 559°F, 2250 psi							
1.3	Austenitic piping weld HAZs	IGSCC	1	3	3	Very good field experience - no known cracking due to SCC	Very good field experience and not anticipated to be a long term problem	2,9,10,19, 20,21,22,23,25,27
	Types 304,316, PWR primary	CF	1	2	2	Doubt lab data base is as good as for wrought materials	Very good field experience	2,9,10,19, 20,21,22,23,25,27
	556 to 559°F, 2250 psi							

Abbrev | Group 1 | Group 2 | Group 3 | Group 4 | Group 5 | Group 6 | Group 7 | Group 8

Proactive Materials Degradation Assessment

Identify Components of Interest – Longer Term Activity (Cont.)

- Interesting insights have already been developed by the Expert Panel on potential future degradation mechanisms
- Six more Expert Panel meetings remain to examine rest of PWR and BWR components
- PWR report, including peer review, prepared by June 2005
- BWR report, including peer review, prepared by December 2005

Proactive Materials Degradation Assessment

International Cooperative Research Group

- To accomplish the second step, an international group will be assembled
- Technical experts and sponsoring organizations
- Together develop a broad-based research program plan
  - Materials and degradation mechanisms
  - Mitigation
  - Repair and replacement
  - Nondestructive evaluation
- Through cooperative agreement, sponsor, implement, and share research results
- Meetings to develop program plan and cooperative agreement:
  - USA, Europe, Japan
- Initiate cooperation and any new research in 2006

Proactive Materials Degradation Assessment

Utilization of Results

- Results consist of lists of plant components susceptible to future degradation mechanisms, reasoning behind these calls, and knowledge base for these mechanisms

- Provide input into development of materials degradation International Research Cooperative Program to allow effective implementation of proactive approaches to materials degradation

- Provide basis for NRR to implement regulatory actions
  - ISI and leak monitoring

(7)

Update To ACRS On Industry Materials Initiative

November 4, 2004

Robin Jones EPRI
Robin Dyle SNC

Evaluating Knowledge Gaps and Vulnerabilities from a Strategic Perspective

Robin Dyle
SNC

Materials Initiative

- Approved by NSIAC in May 2003
- Initiative
  - Each licensee will endorse, support and meet the intent of NEI 03-08, "Guideline for the Management of Materials Issues"
    - Effective January 2, 2004. Actions required:
      - Commitment of executive leadership, technical personnel, funds and implementation of guidance documents
  - Purpose
    - Provide for
      - Consistent management processes
      - **Prioritization of materials issues**
      - **Proactive, integrated and coordinated approaches**
  - Assure the safe, reliable and efficient operation of U.S. nuclear power plants in the management of materials issues

Materials Initiative

- Policy Statement
  - Through the activities described in the following sections, the industry will ensure that its management of materials degradation and aging is ***forward-looking and coordinated*** to the maximum extent practical. Additionally, the industry will continue to rapidly identify, react and ***effectively respond to emerging issues***. The associated work will be managed to emphasize safety and operational risk significance as the first priority, appropriately balancing long-term aging management and cost as additional considerations. To that end, as issues are identified and as work is planned, the groups involved in funding, managing and providing program oversight will ensure that the ***safety and operational risk significance of each issue is fully established prior to final disposition***.

NEI 03-08 Defined Relationships

- Establishes two Standing Committees
  - Executive Oversight – 'MEOG'
    - Overall coordination/broad policy guidance
      - NSIAC members, Executive Leads of Issue Programs
  - Technical Advisory – 'MTAG'
    - Support MEOG and IPs, develop 'strategic' plan
      - Technical leads of the IPs
      - Serves as APWG for Materials Degradation/Aging in the EPRI NPC arena
- Establishes Policy
  - Defines roles, responsibilities, and expectations
- **IP oversight structures – continue to be responsible for technical program work**

Materials Initiative Issue Programs (IPs)

- Programs/Areas Governed by Materials Initiative
  - BWRVIP
  - MRP
  - SGMP
  - FRP
  - Non-Destructive Examination Program and Performance Demonstration Initiative (NDE, PDI)
  - Chemistry and Corrosion Research Programs
  - 3 NSSS Owners Groups Programs for Materials Management (WOG,B&WOG,BWROG)

How Much Are These IPs Spending on Resolving Materials Issues?

	2004 Budget ($K)	2005 Budget ($K)
BWRVIP	7000	9000
MRP	9000	9000
SGMP	6700	6700
FRP	7600	7600
NDE Center	7000	7000
Water Chemistry	2000	2000
Corrosion Research	1000	1000
WOG	3300	3200
BWOG	2100	2000
Total	**45700**	**47500**

Materials Initiative

- Expectations
  - The body of materials work conducted across the industry will be ***forward-looking and coordinated, resulting in fewer unanticipated issues*** that could consume an inordinate level of resources and divert focus from an orderly approach to managing materials
  - This initiative will enhance the issue programs' ability to rapidly identify, react and effectively respond to emerging issues
  - Every utility will fully participate in the implementation of the materials management activities applicable to its plants

Integrated Materials Issues Strategic Plan

- Provides Comprehensive View of all Materials Issues
  - Identifies highest priority challenges & activities
  - Identifies Issue Program's (IP) responsibilities for addressing challenges and issues (EPRI & NSSS OGs)
  - Coordinates IP industry efforts
- Provides for:
  - Proactively addressing existing and future materials problems before they become major operational or regulatory issues
    - Focuses the collective technical and financial resources to address problems
    - Identifies (and develops) future technological, personnel and resource needs to support the industry
    - Provides framework for industry and regulatory interaction and communications
    - Provides vehicle to coordinate industry's response to emerging issues

Integrated Materials Issues Strategic Plan

- Provides Systematic Approach to Managing Materials Issues
  - Identify vulnerabilities
  - Assess condition (inspect & evaluate)
  - Mitigate degradation initiation and propagation mechanism
  - Repair or replace as required
- Approach Used:
  - Degradation Matrix and Issue Management Tables
    - Degradation Matrix and Issues Management Tables to be maintained as living documents with annual updates

Integrated Materials Issues Strategic Plan

- **Degradation Matrix**
  - List all materials within Materials Initiative Scope
  - Obtain inputs from experts, laboratory R&D, industry OE
  - Identify potential degradation mechanisms
  - Determine material applicability for each degradation mechanisms
  - Issues identified that pose potential threats
    - Adequately addressed, programs managing issues
    - Work in progress that will develop tools to manage issues
    - No program to address, no work in progress to address vulnerability

IMT Process

- Identify component and component function
- Identify material(s) of construction
- Identify degradation mechanism(s)
  - May be a different mechanism for different location/material of a component
  - Likelihood or predominance of a mechanism should be considered and ranked (e.g. IGSCC may overwhelm fatigue)
- Identify locations that can fail
- Identify consequences of failure, including system responses to help prioritize location/component importance
- Identify inspection capabilities and history – what can be done and is it effective to deal with the degradation of concern

IMT Process (cont.)

- Identify evaluation capabilities – what is known about environmental effects on crack growth and initiation etc.
- Identify mitigation options/technologies. This would include things such as chemical (e.g. zinc, NMCA), mechanical (e.g. MSIP), or system operation changes (e.g. BWR feedwater flow controller)
- Identify repair or replacement options, capabilities and limitations.
- Based on the information above, identify knowledge gaps/needs
- Prioritize the work to resolve gaps and identify who will do what pieces of the work to eliminate the gap.

Example IMT, BWR

Equipment	Material	Failure Mechanism	Consequences of Failure	Mitigation	Repair / Replace	I & E Guidance	Gaps	Priority & Basis	Responsible Program(s)
BWR Recirculation piping	SS (lc and hc). Inconel welds	SCC, fatigue	Leakage, forced outage	Yes, chemical and stress improvement	Yes, replace pipe or weld overlay	Yes. BWRVIP-75		Low – solution available	BWRVIP, WCC
BWR Vessel	Cs/las. ss clad. welds	IGSCC. IASCC. TGSCC, FIV. Th & Env Fatigue. Emb. Th aging, Fluence	LOCA – loss of asset	Yes – HWC, NMCA	Yes – nozzle repair	Yes – covers embrittlement and weld degradation		Low – solution available	BWRVIP
BWR Internals	Ss. cass, cs. welds. Inc	IASCC. IGSCC, FIV, Wear. EF. Emb, Fluence – R&D needed	Core configuration	Yes – some, work needed	Yes – shroud and top-guide, costly – work needed	Yes (interim) – 13 BWRVIP I&E Guidelines – work needed		High – existing and potential unresolved issues	BWRVIP, WCC, FRP, Corrosion Research

Integrated Materials Issues Strategic Plan

- ***Current Strategic Issues Identified by DM/IMT Process***
  - Nickel Based Alloy Stress Corrosion Cracking (SCC)
  - NDE Technology
  - High Fluence in BWRs and PWRs
  - Steam Generator Tubing
  - Fuel Integrity
  - Water Chemistry
- Detailed in 'Industry Materials Management Annual Work Plan'
  - Managed as a Supplement to the 'Strategic Plan'
  - DM/IMT are living documents and will be updated at least annually

8

Defining Materials Degradation Vulnerabilities

Robin Jones
EPRI

ACRS Meeting
November 4, 2004

EPRI

Summary

- In support of the "Industry Initiative on Management of Materials Issues", an expert elicitation process has been used to obtain input on the degradation vulnerabilities of all of the classes of materials used in the major passive components in BWR and PWR reactor coolant systems.
- The experts' input has been used to create a first-generation electronic tool called the Degradation Matrix which can be used, in conjunction with other information, to assess the relative priority of current and potential materials degradation issues and associated R&D needs.
- The Degradation Matrix development process will be described and, time permitting, a brief demonstration will be conducted

EPRI

Integrated Materials Issues Strategic Plan

- Defines a Systematic Approach to Managing Materials Issues
  - Identify vulnerabilities
  - Assess condition (inspect & evaluate)
  - Mitigate degradation initiation and propagation mechanisms
  - Repair or replace as required
- The Degradation Matrix and Issue Management Tables are Tools to Support the Systematic Approach
  - Degradation Matrix and Issue Management Tables will be maintained as living documents with annual updates

EPRI

Integrated Materials Issues Strategic Plan

- Degradation Matrix
  - Identify materials used for major passive components/systems within Materials Initiative Scope
  - Obtain inputs from experts, laboratory R&D, industry OE
    - Identify potential degradation mechanisms
    - Determine material applicability for each degradation mechanism
    - Define areas of uncertainty
  - Identify and characterize issues that pose potential threats
    - Adequately addressed, programs managing issues
    - Work in progress that will develop tools to manage issues
    - No program to address, insufficient work in progress to address vulnerability

EPRI

Materials Degradation Matrix

Level 1

PWR						BWR		
PWR Reactor Pressure Vessel	PWR Pressurizer	PWR SG Shell	PWR Reactor Internals	PWR Piping	PWR SG Tubes & Internals	BWR Pressure Vessel	BWR Reactor Internals	BWR Piping

EPRI

Materials Degradation Matrix

Level 1

PWR						BWR		
PWR Reactor Pressure Vessel	PWR Pressurizer	PWR SG Shell	PWR Reactor Internals	PWR Piping	PWR SG Tubes & Internals	BWR Pressure Vessel	BWR Reactor Internals	BWR Piping

Level 2

PWR Component	Material	SCC SCC					Corrosion/Wear C & W				Fatigue Fat.			Reduction in Toughness RiT					
														Aging	*Irradiation*				
	Subdivision→	IG	IA	TG	LTCP	PW	Wstg	Pit	Wear	FAC	HC	LC/Th	Env	Th	Emb	VS	SR	Th_n	Fl
PWR Pressurizer (Including Shell, Surge and Spray Nozzles, Heater Sleeves and Sheaths, Instrument Penetrations)	C&LAS																		
	C&LAS Welds																		
	Wrought SS																		
	SS Welds & Clad																		
	Wrought Ni Alloys																		
	Ni-base Welds & Clad																		

EPRI

Materials Degradation Matrix

Level 1

PWR						BWR		
PWR Reactor Pressure Vessel	PWR Pressurizer	PWR SG Shell	PWR Reactor Internals	PWR Piping	PWR SG Tubes & Internals	BWR Pressure Vessel	BWR Reactor Internals	BWR Piping

Level 2

PWR Component	Material	SCC SCC					Corrosion/Wear C & W				Fatigue Fat.			Reduction in Toughness RiT					
														Aging	Irradiation				
	[1]Subdivision→	IG	IA	TG	LTCP	PW	Wstg	Pit	Wear	FAC	HC	LC/Th	Env	Th	Emb	VS	SR	Th_n	Fl
PWR Pressurizer (Including Shell, Surge and Spray Nozzles, Heater Sleeves and Sheaths, Instrument Penetrations)	C&LAS	? e002	N	? e002	N	? e003	Y e004	N	N	Y e005	N	Y e006	Y e007	Y e008	N/A	N/A	N/A	N/A	N/A
	C&LAS Welds	? e002	N	? e002	N	? e003	Y e004	N	N	Y e005	N	Y e006	Y e007	Y e008	N/A	N/A	N/A	N/A	N/A
	Wrought SS	? e012	N	? e012	? e013	? e012	N	N	N	N	N	Y e014	Y e015	N	N/A	N/A	N/A	N/A	N/A
	SS Welds & Clad	Y e016	? e017	Y e018	? e013	? e019	N	N	? e020	N	N	? e014	Y e015	Y e022	N/A	N/A	N/A	N/A	N/A
	Wrought Ni Alloys	N	N	N	? e023	Y e023	N	N	N	N	Y e014	Y e014	Y e015	N	N/A	N/A	N/A	N/A	N/A
	Ni-base Welds & Clad	N	? e024	N	Y e023	Y e025	N	N	N	N	N	Y e014	Y e015	N	N/A	N/A	N/A	N/A	N/A

Level 3

e030	Corrosion-assisted fatigue is a known phenomenon on secondary side (e.g., in the vicinity of girth welds in steam generator shells and in the region of feedwater nozzles) and is not like environmental fatigue described in other areas of this DM. Environmental fatigue research relevant to this specific phenomenon is not ongoing within MRP Fatigue ITG, and is a potential gap.

Materials Degradation Matrix

Level 1

PWR						BWR		
PWR Reactor Pressure Vessel	PWR Pressurizer	PWR SG Shell	PWR Reactor Internals	PWR Piping	PWR SG Tubes & Internals	BWR Pressure Vessel	BWR Reactor Internals	BWR Piping

Level 2

PWR Component	Material	SCC SCC					Corrosion/Wear C & W				Fatigue Fat.			Reduction in Toughness RiT					
														Aging	Irradiation				
	[1]Subdivision→	IG	IA	TG	LTCP	PW	Wstg	Pit	Wear	FAC	HC	LC/Th	Env	Th	Emb	VS	SR	Th_n	Fl
PWR Pressurizer (Including Shell, Surge and Spray Nozzles, Heater Sleeves and Sheaths, Instrument Penetrations)	C&LAS	[illegible]	N	[illegible]	N	[illegible]	Y e004	N	N	Y e005	N	Y e006	Y e007	Y e008	N/A	N/A	N/A	N/A	N/A
	C&LAS Welds	[illegible]	N	[illegible]	N	[illegible]	Y e004	N	N	Y e005	N	Y e006	Y e007	Y e008	N/A	N/A	N/A	N/A	N/A
	Wrought SS	[illegible]	N	[illegible]	[illegible]	[illegible]	N	N	N	N	N	Y e014	Y e015	N	N/A	N/A	N/A	N/A	N/A
	SS Welds & Clad	Y e016	[illegible]	Y e018	[illegible]	[illegible]	N	N	[illegible]	N	N	[illegible]	Y e015	Y e022	N/A	N/A	N/A	N/A	N/A
	Wrought Ni Alloys	N	N	N	[illegible]	[redacted]	N	N	N	N	Y e014	Y e014	Y e015	N	N/A	N/A	N/A	N/A	N/A
	Ni-base Welds & Clad	N	[illegible]	N	[redacted]	[redacted]	N	N	N	N	N	Y e014	Y e015	N	N/A	N/A	N/A	N/A	N/A

Level 3

e030	Corrosion-assisted fatigue is a known phenomenon on secondary side (e.g., in the vicinity of girth welds in steam generator shells and in the region of feedwater nozzles) and is not like environmental fatigue described in other areas of this DM. Environmental fatigue research relevant to this specific phenomenon is not ongoing within MRP Fatigue ITG, and is a potential gap.

Materials Degradation Matrix

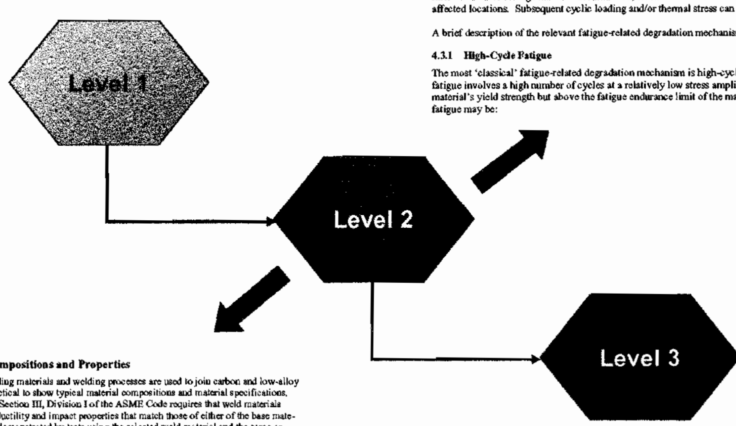
4.3 Fatigue Degradation Mechanisms and Mitigation Options

Fatigue is the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads or temperatures. After repeated cyclic loading, if sufficient localized micro-structural damage has been accumulated, crack initiation can occur at the most highly affected locations. Subsequent cyclic loading and/or thermal stress can cause crack growth.

A brief description of the relevant fatigue-related degradation mechanisms is provided below.

4.3.1 High-Cycle Fatigue

The most 'classical' fatigue-related degradation mechanism is high-cycle (HC) fatigue. HC fatigue involves a high number of cycles at a relatively low stress amplitude (typically below the material's yield strength but above the fatigue endurance limit of the material). High cycle fatigue may be:



3.2 Material Compositions and Properties

A large variety of welding materials and welding processes are used to join carbon and low-alloy steels, and it is not practical to show typical material compositions and material specifications. Section NB-2431.1 of Section III, Division I of the ASME Code requires that weld materials have tensile strength, ductility and impact properties that match those of either of the base materials being welded, as demonstrated by tests using the selected weld material and the same or similar base materials. Section NB-2432.2 of Section III, Division I of the ASME Code requires that the chemical composition of the welding material be in accordance with an appropriate ASME Code welding specification (in Section II.C of the Code), but leaves the choice of the specific material up to the manufacturer.

The most common weld processes used to join carbon steel and LAS parts include submerged arc welding, shielded metal arc welding (SMAW), and gas tungsten arc welding (GTAW). Post-weld heat treatment is generally required per ASME Code rules after welding of the carbon and low-alloy steels used for reactor coolant system service.

Plans for 2005

- The Degradation Matrix will be updated/revised in 2005:
  - Update current tables and comments via another expert elicitation workshop
  - Add a table to address degradation of materials used only in active components
  - Complete the development of the Core Materials Degradation Matrix
  - Switch from Microsoft Word to a web-enabled approach to facilitate implementation of future updates and to provide an easy means of linking to key reference documents

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Post-Fire Operator Manual Actions Rulemaking

David Diec
Richard Rasmussen
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November 4, 2004
ACRS Brief

NRR
Office of Nuclear Reactor Regulation

9

Post-Fire Operator Manual Actions Rulemaking

NRR
Office of Nuclear Reactor Regulation

- Background
- Key topics
  - Security interface
  - Compliance
  - Risk informing proposed rule
  - Acceptance criteria
  - Detection and suppression
  - Time Margin Concept
- Proposed rule status

Background

- SECY 03-0100 Rulemaking Plan on Post-Fire Operator Manual Actions [ML023180599]
  - Revise 10 CFR Part 50, Appendix R, Section III.G.2
  - Codify operator manual actions option in section III.G.2 (redundant trains located in the same fire area)
  - Consider enforcement discretion or other alternatives to provide regulatory stability
- Staff Requirements Memorandum (SRM)
  - Commission approved staff rulemaking plan on September 17, 2003

Background

- Rule objectives
  - Effectiveness
    - Clarify use of operator manual actions as a regulatory option
    - Reduce need for individual review of plant specific OMA
  - Ensure safety
    - Provide a framework to establishing feasible and reliable operator manual actions (OMA) and detection and automatic suppression

Background

- Stakeholder interactions
  - September 9, 2003 ACRS Fire Protection Subcommittee on rulemaking plan
  - October 17, 2003 public meeting
  - November 12, 2003 public meeting and FRN publication
  - April 24, 2004 ACRS Fire Protection Subcommittee
  - June 23, 2004 public meeting
  - Proposed rule text publicly available on October 25, 2004

Security and the Rule

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- Security is not currently addressed in 10 CFR 50, Appendix R
  - Security concerns must be considered in a broader context than fire
  - Safety-security interface is being evaluated for future rulemaking
  - Industry communication is being developed

Compliance

NRR Office of Nuclear Reactor Regulation

- Non-compliance is not condoned
- NRC confirmed that unapproved operator manual actions under III.G.2 are a non-compliance
- ROP continues today

TIMELINE	NRC ACTIVITIES
1980 ↓	Conduct Appendix R Fire Protection (FP) inspections
1990 ↓	Continue FP inspections
	Discredit Thermo-Lag
2000 ↓	Conduct FP Triennial & ROP inspections
	Determine non-exempted III.G.2 OPMAX are non-compliant
	Revise IP 71111.05 (March 2003)
	Initiate III.G.2 OPMAX rulemaking

Compliance (cont.)

NRR
Office of Nuclear Reactor Regulation

- SECY 03-0100 Rulemaking Plan on Post-Fire Operator Manual Actions [ML023180599]
  - Revise 10 CFR Part 50, Appendix R, Section III.G.2
  - Codify operator manual actions option in section III.G.2
  - Consider enforcement discretion or other alternatives to provide regulatory stability

Risk Informing Proposed Rule

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- Risk is plant and situation specific
- Risk informing possible only by establishing acceptance criteria relating to CDF, DID, and SM
- Risk informing option is available
  - 10CFR50.48(c)
  - RG-1.174 exemptions
- Existing Appendix R rule is deterministic
- Risk-informing only III.G.2(c-1) affects other sections of the rule
- Maintain consistency with III.G.2(a)-(c)

Acceptance Criteria

NRR
Office of Nuclear Reactor Regulation

- Feasible (it can be done) and reliable (ensures low probability of failure)
- Permit both the licensees and NRC to establish consistency as to what operator manual actions will be allowed
- Provide the parameters which both licensees and NRC will use to conduct evaluations and inspections in a thorough manner.

Acceptance Criteria

NRR
Office of Nuclear Reactor Regulation

- Criteria were developed considering
  - Fires present unique hazards in efforts to mitigate their effects
  - Fires result in unique environmental conditions for operators
  - Similar requirements exist in accepted standards and regulatory guidance (e.g., III.G.3, NRC IP, ROP/SDP, NUREGs, and NUREG/CRs)

OPERATOR MANUAL ACTIONS FIRE DETECTORS AND AUTOMATIC SUPPRESSION

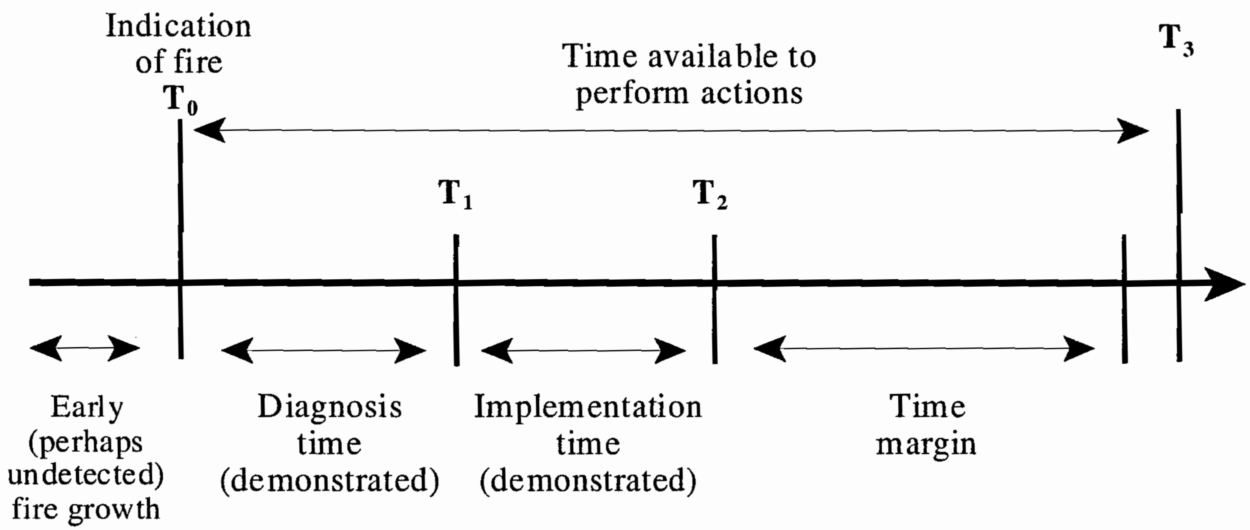
NRR
Office of Nuclear Reactor Regulation

GRAPHICAL REPRESENTATION OF REQUIREMENT FOR FIRE DETECTORS AND AUTOMATIC FIRE SUPPRESSION SYSTEM FOR OPERATOR MANUAL ACTIONS OPTION III.G.2(C-1)

<table>
<tr><td colspan="4">COMPLIANCE ACHIEVED (IMPLIED EQUIVALENCIES)</td></tr>
<tr><td rowspan="3">3-HR FIRE BARRIER</td><td colspan="3">FIRE DETECTORS</td></tr>
<tr><td colspan="3">AUTOMATIC FIRE SUPPRESSION SYSTEM</td></tr>
<tr><td>20-FT SEPARATION W/O INTERVENING COMBUSTIBLES</td><td>1-HR FIRE BARRIER</td><td>OPERATOR MANUAL ACTIONS WITH ACCEPTANCE CRITERIA</td></tr>
<tr><td>III.G.2(a)</td><td>III.G.2(b)</td><td>III.G.2(c)</td><td>III.G.2(c-1)</td></tr>
</table>

Time Margin Concept

NRR
Office of Nuclear Reactor Regulation



Time Margin Concept (cont.)

NRR
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- Expert panel recommended 100% of total demonstrated time (i.e., double the demonstrated time (T_2) and show still within time available (T_3))

Proposed Questions in FRN

NRR
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- Time margin
  - Single vs. a range of multiplicative factors
    - Minimum additive time (for very short times, where multiplicative factor is insufficient)
  - Other means of demonstrating margin
- Suppression
  - Fixed vs. automatic
- Applicability of criteria
  - III.G.1 and III.G.3

Post-Fire Operator Manual Actions Rulemaking

NRR
Office of Nuclear Reactor Regulation

- Schedule
  - Proposed rule to the Commission by early December 2004

(10)

Industry Views
Manual Actions Rulemaking: An Update

ACRS Meeting

November 4, 2004

Fred Emerson, NEI

NEI

Industry Recommendations for Manual Actions Rulemaking

- Provide simple rule change to effect rulemaking goals
  - Provide acceptance criteria for manual actions in a Regulatory Guide
- Address security events in 10 CFR 73 rulemaking rather than in manual actions rulemaking
- Eliminate requirement for automatic suppression in the area of the fire
- Eliminate requirement for time margin factor and treat manual actions consistently with other operator actions
- Improve stakeholder participation in
  - The process of developing reasonable acceptance criteria
  - Addressing other concerns about the rulemaking

NEI

Simple Rule

- Modify NRC-proposed III.G.2 paragraph c-1 as follows:

  "Operator manual actions that, in concert with other fire protection features, maintain one train of safe shutdown equipment free of fire damage."

- Place appropriate acceptance criteria in a Regulatory Guide
  - Criteria in Inspection Manual on Fire Protection 71111.05, 3/6/03, are generally appropriate

Security Events

- Remove reference to security events from manual actions rulemaking and regulatory guidance

- Address security events within 10 CFR 73 rulemaking

Automatic Suppression

- Remove requirement for new automatic suppression capability in the area of the fire
  - Adequate suppression is already provided in fire areas based on fire hazards analysis results
  - Requirement adds nothing to licensee ability to carry out manual actions in areas separate from the fire area
    - Would not enhance either feasibility or reliability of these actions
  - Likelihood of requesting exemptions to this provision negates the intent of rulemaking
  - High cost for exemptions or modifications with little or no safety gain

NEI

Time Margin

- Remove this requirement; treat manual actions consistently with other operator actions in plant operations and event response
  - These operator actions are not penalized with arbitrary time margin factors to guarantee reliability
  - A performance-based approach would
    - Provides more credit for demonstrated performance
    - Allow alternate methods for demonstrating reliability
    - Reduce or eliminate need for high-cost changes to existing T-H analyses
    - Avoid duplicate or burdensome conservatism
  - Could use public interactions or workshops to develop performance goals and explore methods for satisfying them
  - Likelihood of requesting exemptions to this provision negates the intent of rulemaking

NEI

Net Result of Industry Recommendations

- Provides simple, flexible rule
- Maintains a safety focus with appropriate acceptance criteria
- Treats manual actions consistently with operator actions used in plant operations and event response
- Provides performance goals for reliability and recognizes alternate methods to meet performance goals
- Provides more opportunity for stakeholder input
- Reduces or eliminates need for expensive revisions to T-H analysis, modifications, or exemptions with little or no safety benefit

NEI

Electrical Grid Reliability

John G. Lamb
Division of Engineering Applications
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

SUMMARY

- Generic communication may be needed in order to ensure future licensee readiness to cope with an event similar to the August 14, 2003, power outage.

BACKGROUND

- August 14, 2003, Blackout
- Chairman directed EDO to review the issues raised in the "State of U.S. Power Plants from a Nuclear Power Plant Perspective."

STAFF ACTIONS

- Grid-Related Issues - 48 Issues
- Group 1 - Short-Term (Summer 2004) - 10 Issues
- Group 2 - Federal Energy Regulatory Commission (FERC)/ North American Electric Reliability Council (NERC) -21 Issues
- Group 3 - Long-Term - 17 Issues

KEY INFORMATION

- NRC staff believes effective actions are being taken to enhance the availability of offsite power for safe nuclear power plant operation.
- Nuclear power plant operators need to be aware of the offsite power needs.
- Found considerable variability and uncertainty among licensees regarding the responses to the three key questions of the Temporary Instruction.

KEY INFORMATION

- Cooperation of the transmission system operator may have to be enlisted through an appropriate communication interface to ensure that offsite power will be available.
- Generic communication may be needed in order to ensure future licensee readiness to cope with an event similar to the August 14, 2003, power outage.

MILESTONES

- In the Offsite Power System Availability and the Station Blackout Review topical areas, the staff is considering a generic communication.
- The staff will determine if regulatory action is warranted based on RES risk analyses in the Risk Insights topical area.
- The staff will set up a process for NRC to receive NERC operational data and to interact with NERC during grid emergencies.

STATUS OF THE ASSESSMENT OF GRID OPERATING DATA FOR CHANGE AND POTENTIAL VULNERABILITIES

By William Raughley
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

Overview

- **Purpose** - Review grid data for signs of change or potential vulnerabilities that may be masked by investigating NPP data alone.

- **Use** – Evaluate effectiveness of electric power regulatory documents and protective features, and revisit the assumptions about the grid in risk analyses

- **Objectives –** Use grid data to identify and assess:
  - Grid reliability
  - Percent of the time grid is degraded near an NPP
  - Insights from consideration of the offsite power supply as a complex system
  - Vulnerabilities that are potentially risk significant issues for the NPPs

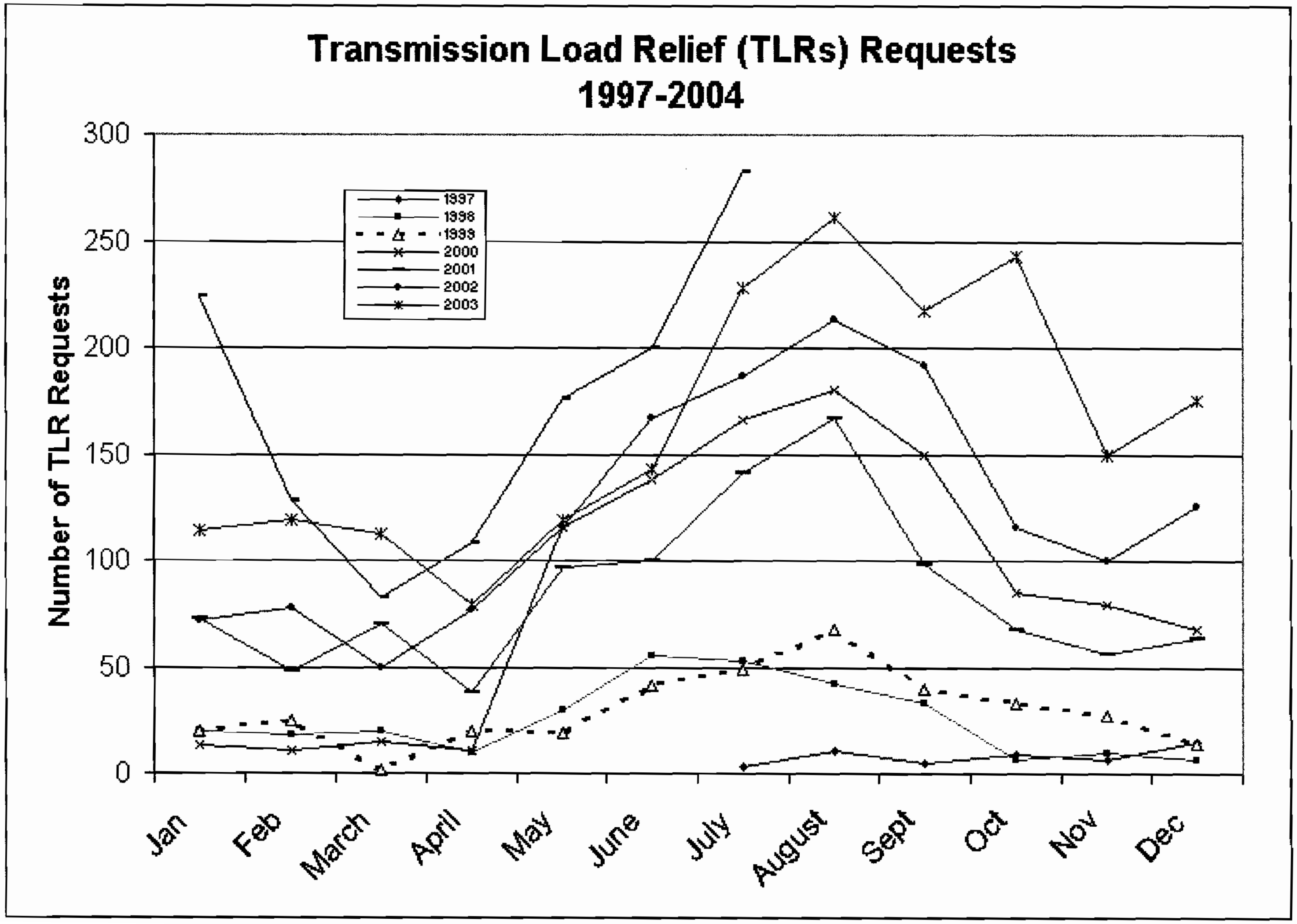
- **Summary** - Developed indices and insights to gauge the impact of changes in transmission system loading and grid reliability based 600 grid events from 1984-2003 and 7000 transmission line records from 1997-2004. Since 1999:
  - Transmission system congestion has increased
  - Grid reliability has changed. The number of larger and longer lasting blackouts have increased
  - Both the grid and the NPP's offsite power supply tend to be complex systems

- **Next Steps**

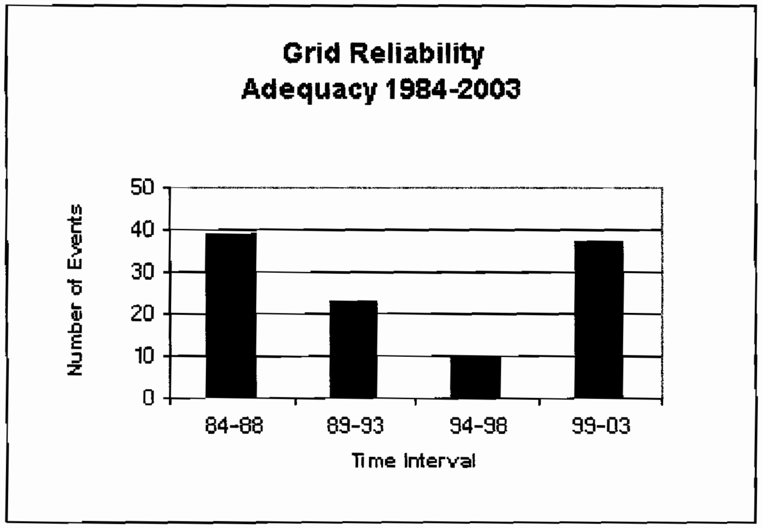
Background

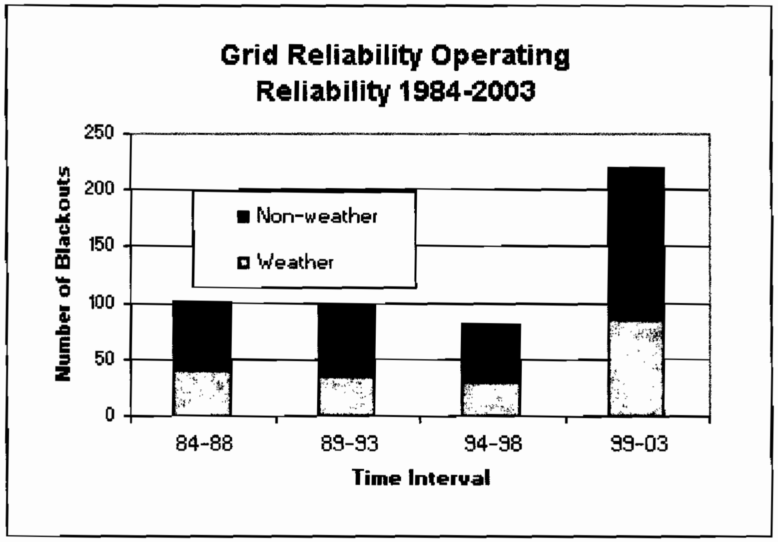
- NERC definitions of reliability
  - Adequacy of generation supply
  - Operating reliability of the power system to withstand a sudden disturbance
- Grid event above predetermined thresholds reported to DOE and NERC.
  - Bin as adequacy, operating reliability, or unusual event
  - Similarities and differences to NPP data
- Potential for increased transmission line loading or congestion
  - Open access of generators to the transmission system from deregulation.
  - Increased utilization to meet increased demand (Blackout Task Force)
  - NERC transmission load relief (TLR) request procedure to manage congestion
  - Experience shows reactor trip with congestion can degrade NPP voltages
- Experts in chaos theory view grid as a complex system

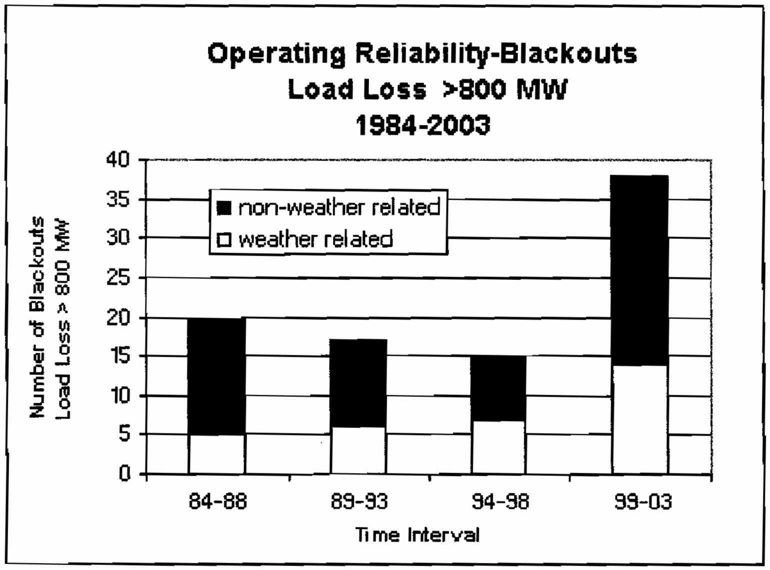
Observation-Increased transmission line loading since 1999

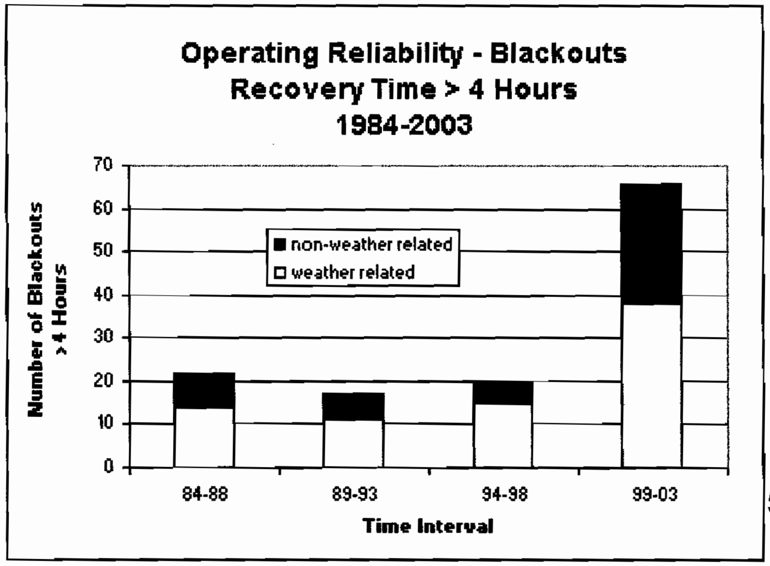


Observations – Grid reliability has changed since 1999. The data since 1999 may reflect true grid performance and challenge the NRC assumptions that use grid data before 1999.





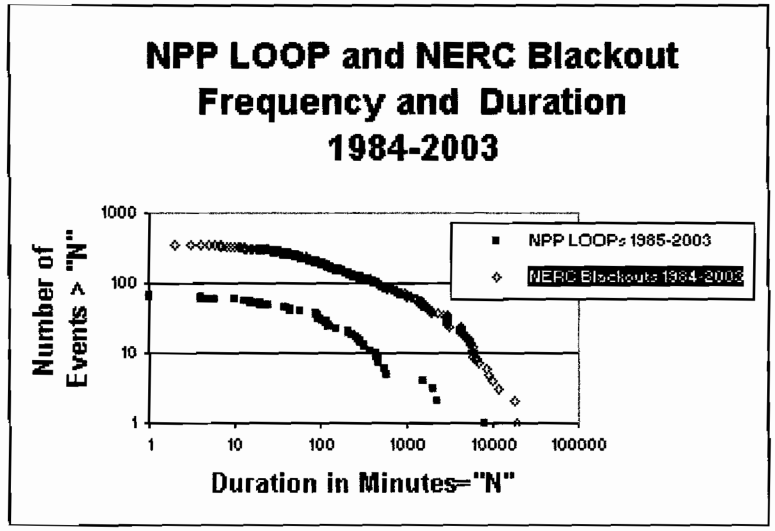


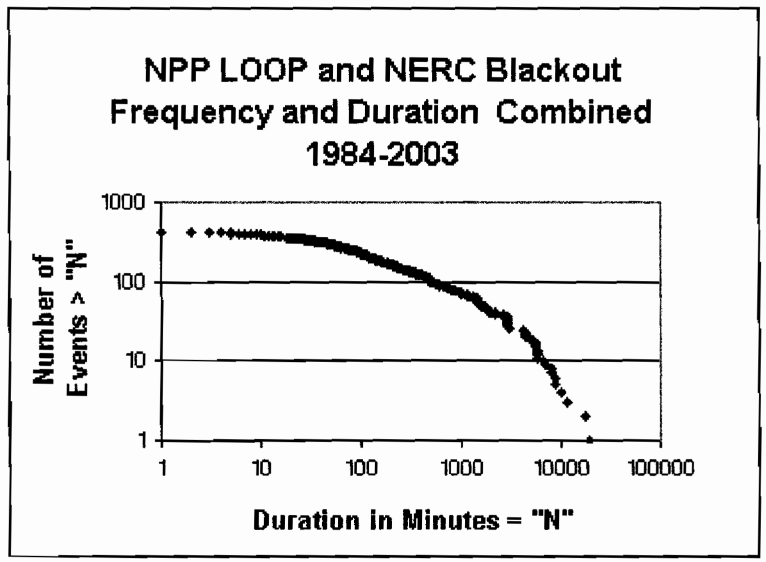


Observations - Grid and NPP offsite power tend to be complex systems

- Possess complex system attributes
  - Described by power laws
  - Small disturbance has widespread effects
  - August 14, 2003 blackout was predictable

- Methods used to describe complex systems differ from those that the NRC currently uses and may provide different results and grid risk insights :
  - LOOP and blackout size rather than NRC cause classification (plant, weather, grid) may be more informative characterization of LOOPs for PRA.





Overview of LOOP Frequency and Duration Update

Dale M. Rasmuson
Operating Experience Risk Analysis Branch
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

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Definitions

- Loss of offsite power (LOOP) is defined as loss of offsite power to all safety buses

- Station blackout (SBO) is the loss of all offsite and onsite AC power to the safety buses

Grid Tasks

- Provide preliminary accident sequence precursor analyses of each affected plant to provide insights for near term agency actions
- Evaluate SBO implications. Using data from recent LOOP events, update the SBO LOOP frequency and duration
- Evaluate SBO risk. Calculate SBO risk (core damage frequency) with updated Standardized Plant Analysis Risk models for a spectrum of plants

Previous LOOP Studies

- LOOP frequencies and durations have been evaluated in four NRC studies
  - NUREG-1032 (1968-1985)
  - NUREG/CR-5496 (1980-1996)
  - NUREG/CR-5750 (1987-1995)
    - Frequency only
  - NUREG-1784 (1986-2002)

LOOP Event Categories

<table>
<tr><th>Study</th><th colspan="4">LOOP Event Category</th></tr>
<tr><td>NUREG-1032</td><td colspan="2">Plant</td><td>Grid</td><td>Weather</td></tr>
<tr><td>NUREG/CR-5496</td><td colspan="2">Plant</td><td>Grid</td><td>Weather</td></tr>
<tr><td>NUREG-1784</td><td>Plant</td><td colspan="2">Grid</td><td>Weather</td></tr>
<tr><td>Current Study</td><td>Plant</td><td>Switchyard</td><td>Grid</td><td>Weather</td></tr>
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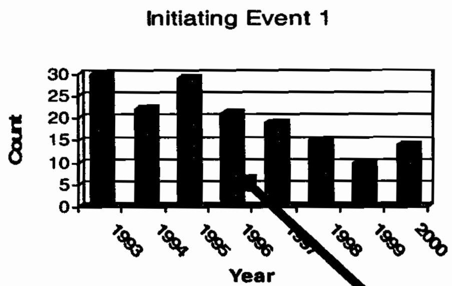
LOOP and SBO Core Damage Frequency

LOOP Frequencies

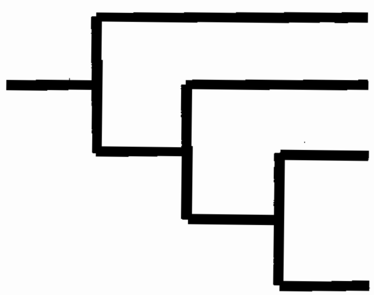


LOOP Durations

EDG Reliability

SPAR Models

Event Tree 1



LOOP and SBO Core Damage Frequency

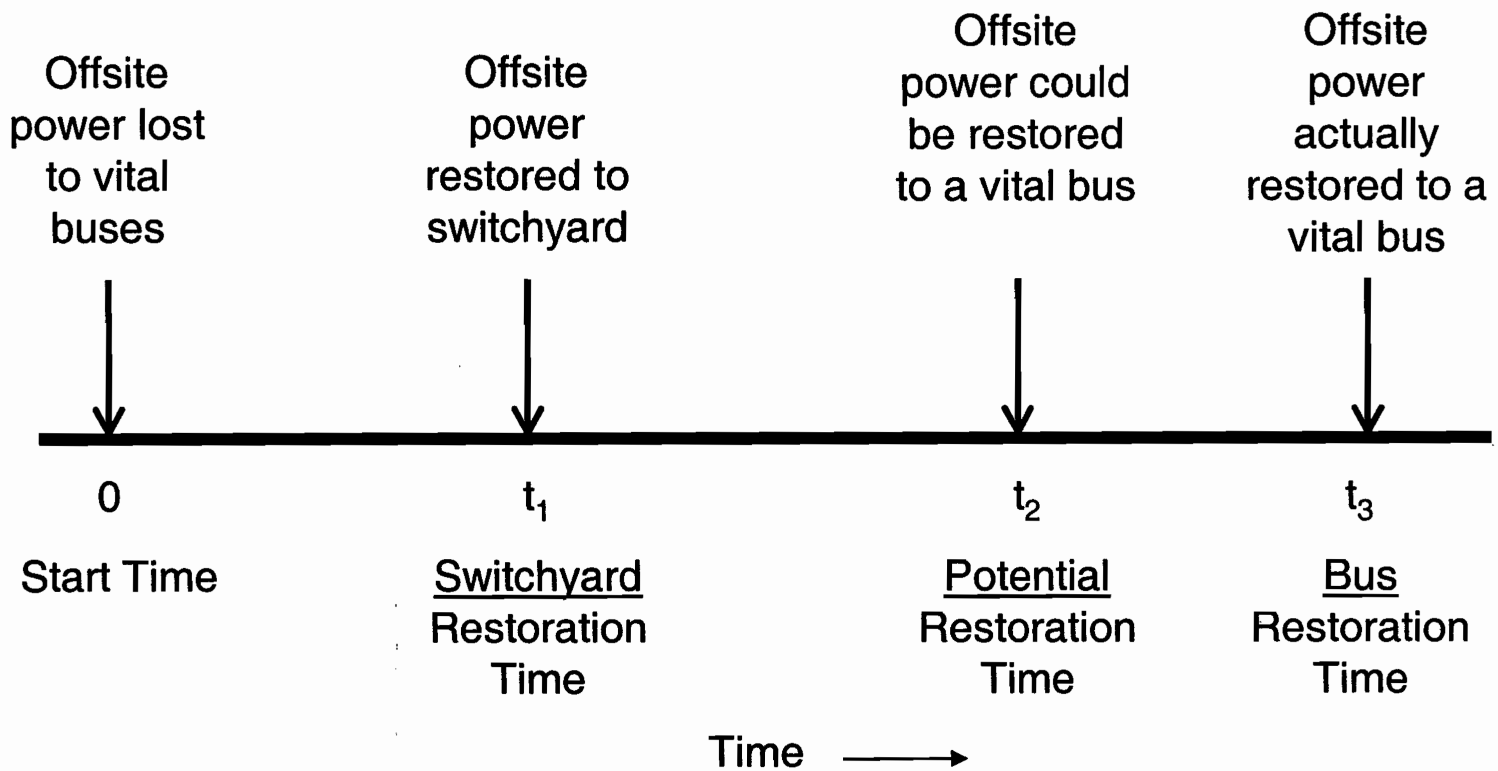
Plant-Specific Coping Features

- Battery depletion time
- Turbine-driven pumps
- Alternate AC power sources
- RCP seal design

SPAR LOOP Model

- The SPAR model LOOP event tree has been updated with:
  - The new Westinghouse and CE pump seal LOCA models
  - Basic event parameter estimates based on EPIX information
- EDG performance will also be evaluated using the latest information available to the NRC

Restoration Time



Potential Restoration Time

- No other power sources are available (i.e., station blackout exists),
- Power is to be restored through the switchyard,
- Power restored to the switchyard is of usable quality,
- Urgency to restore power exists because of potential accident conditions,
- No extensive diagnostics or repair are required,
- Faults have been cleared,
- Operator actions needed involve alignment with relatively routine verification and switching,
- Recovery time is based on a best estimate of the time operators would need to execute necessary power recovery tasks in a pending accident situation , and
- The reasonableness of the estimated recovery time would be based on consideration of HRA factors (e.g. stress, time available, difficulty in recovery tasks, adequacy of training and procedures).

Plant-Specific LOOP Frequencies

- Approaches for estimating plant-specific LOOP frequencies include:
  - Use industry values
  - Use regional estimates
  - Use Bayesian estimates obtained by updating industry distributions with plant-specific information

Status

- ASP analyses are being finalized and will be issued soon
- Frequency and duration analyses are nearing completion. Draft report to be issued for stakeholder review
- CDF evaluations are starting. Draft report to be issued for stakeholder review in early 2005

General Insights

- LOOP frequency has decreased. Basically constant 1997 – 2002
- LOOP durations have slowing increased 1986 – 1996. Relatively constant from 1997 – 2003
- Since 1997 LOOP events have occurred more during summer (May – October)
- The probability of a LOOP event due to a reactor trip has increased during the summer months

Loss of Offsite Power Events

Presentation to ACRS

Thomas Koshy, Section B/EEIB
Division of Engineering
Office of Nuclear Reactor Regulation

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Agenda

- Recent Events
- Overview
- Vermont Yankee Main Transformer Failure
- Limerick Unit # 2 Trip
- River Bend Unit Trip
- Dresden Unit #3 Trip
- Palo Verde – Three Unit Trip
- NRR Actions

OVERVIEW

- Plant trips have resulted from a variety of switchyard and grid related problems
  - Design deficiencies
  - Lack of adequate maintenance
  - Operational oversight – lack of preparation or inappropriate remedial actions in grid management

Vermont Yankee
Main Transformer Failure

Loop Events - ACRS Presentation
Nov. 4, 2004

Vermont Yankee Trip

- On June 18, 2004, a ground fault from the dislodged peace of the isophase bus and the failure of two surge arresters ignited transformer oil on the main transformer
- The fire lasted more than 10 min. Unusual event was declared
- The offsite power remained available
- The licensee's root cause indicates that fire could have been avoided with periodic inspection of Isophase bus and testing of surge suppressors.

Limerick Unit #2 Trip

- On June 22, 2004 when 500KV breaker was opened for maintenance, an internal fault of this breaker and a concurrent failure on another breaker resulted in the isolation of several breakers
- Both of main output breakers tripped
- Unit #2 safety buses auto transferred to the alternate offsite power
- Emergency Diesel Generators were not required

River Bend

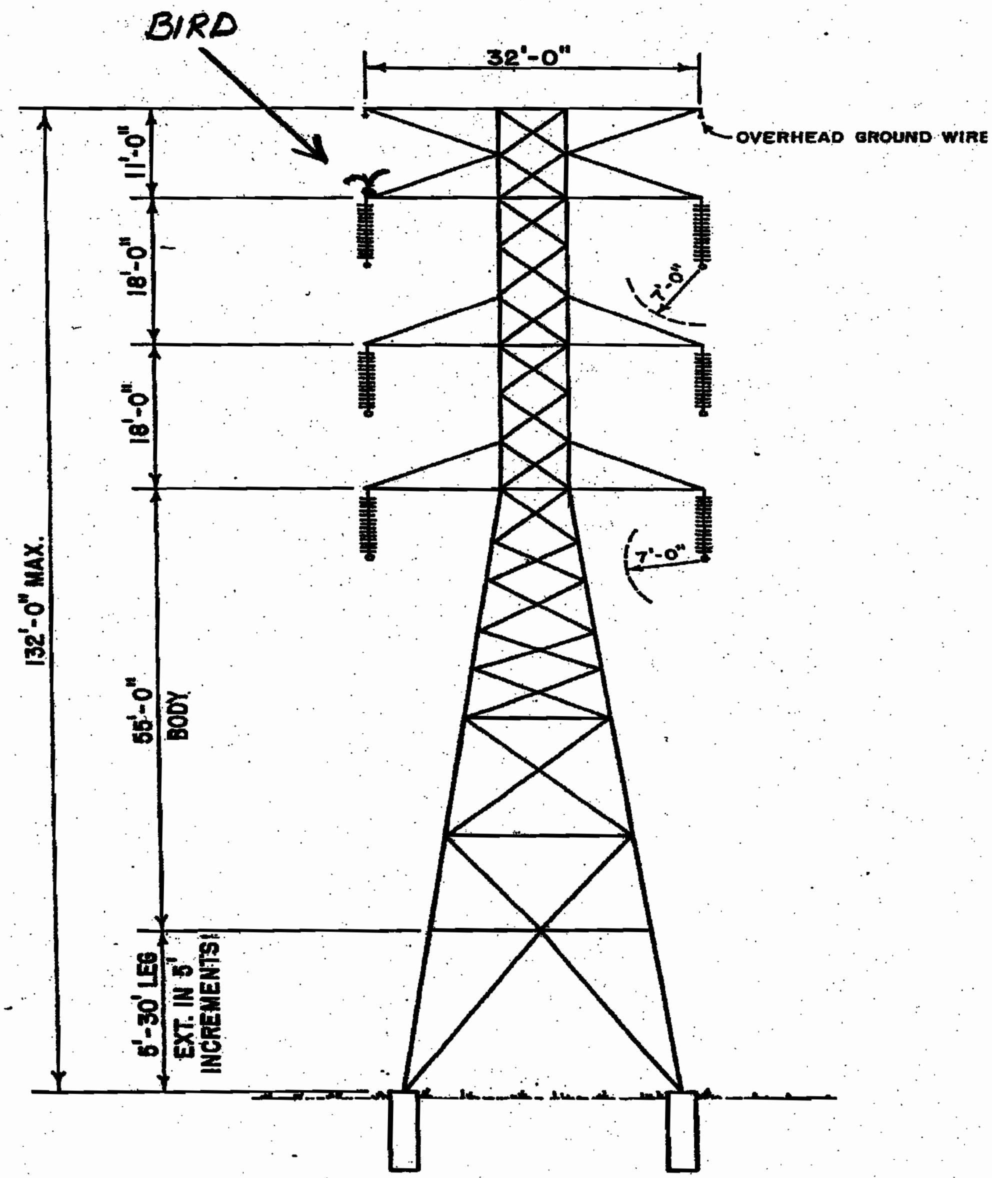
- On August 15, 2004, a remote transmission tower guy wire failure required an automatic trip of certain breakers at the River Bend switchyard
- Since the first set of breakers were slow in clearing the fault, the back up protection actuated another set of breakers including one of the main generator to contain the fault
- This delay in tripping also actuated the ground fault on main step up transformer. Initially one of the main output breakers and then both breakers were tripped.
- Division 2 safety bus was powered by the emergency diesel generator for about 8 hrs.
- Slow operation of the breaker was attributed to inadequate maintenance

Palo Verde- 3 Unit Trip

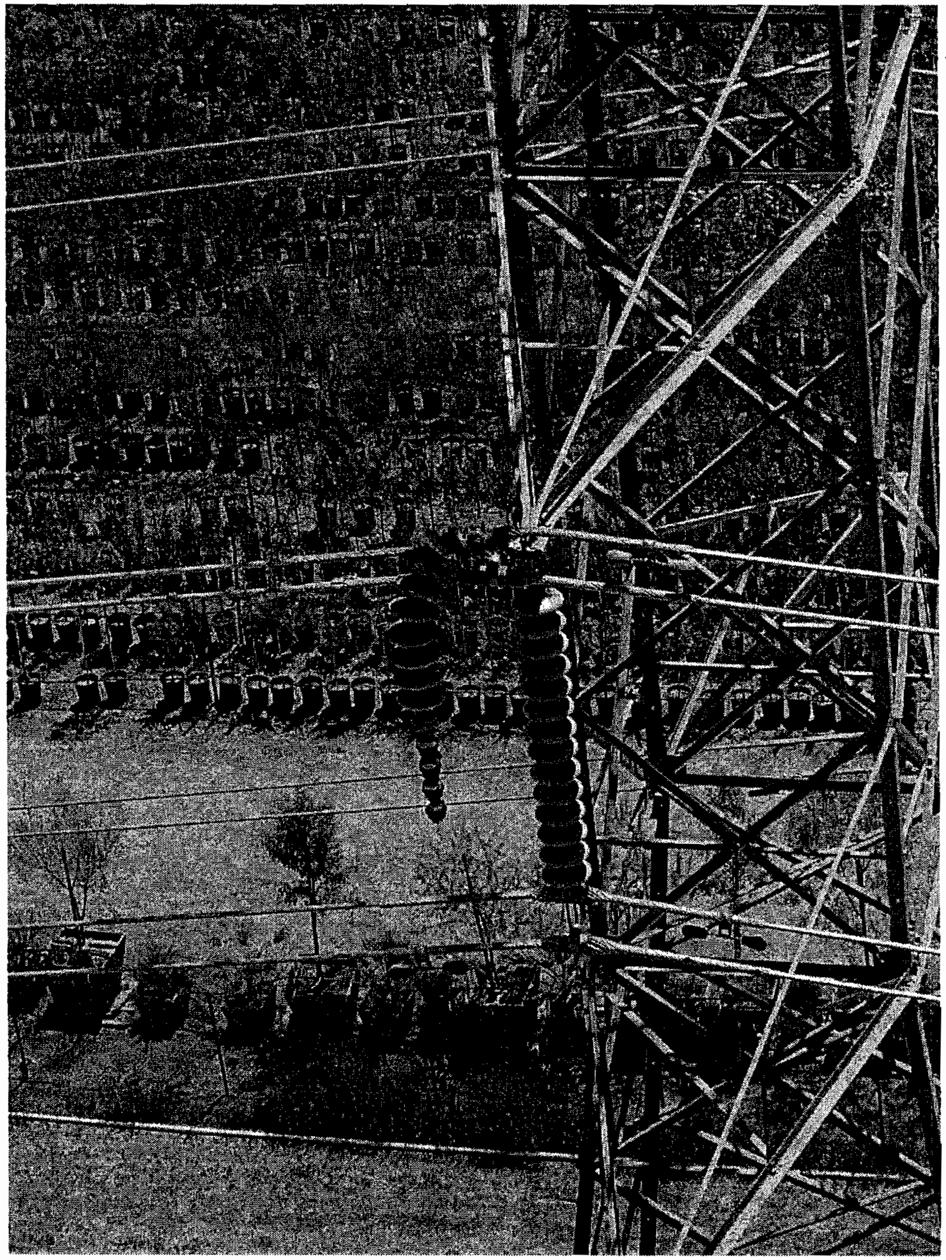
- On June 14, 2004 at 7:44 a.m. all three Palo Verde Units tripped. One EDG for unit #2 failed to run
  - Unit #2 went into an Alert because of one EDG and all offsite power unavailability
  - All three units tripped from loss of load
  - Electrical fault remained for 39sec.,
  - Offsite power recovered by 8:18 a.m.
  - 9:51 a.m. safety buses energized from offsite power for unit#2

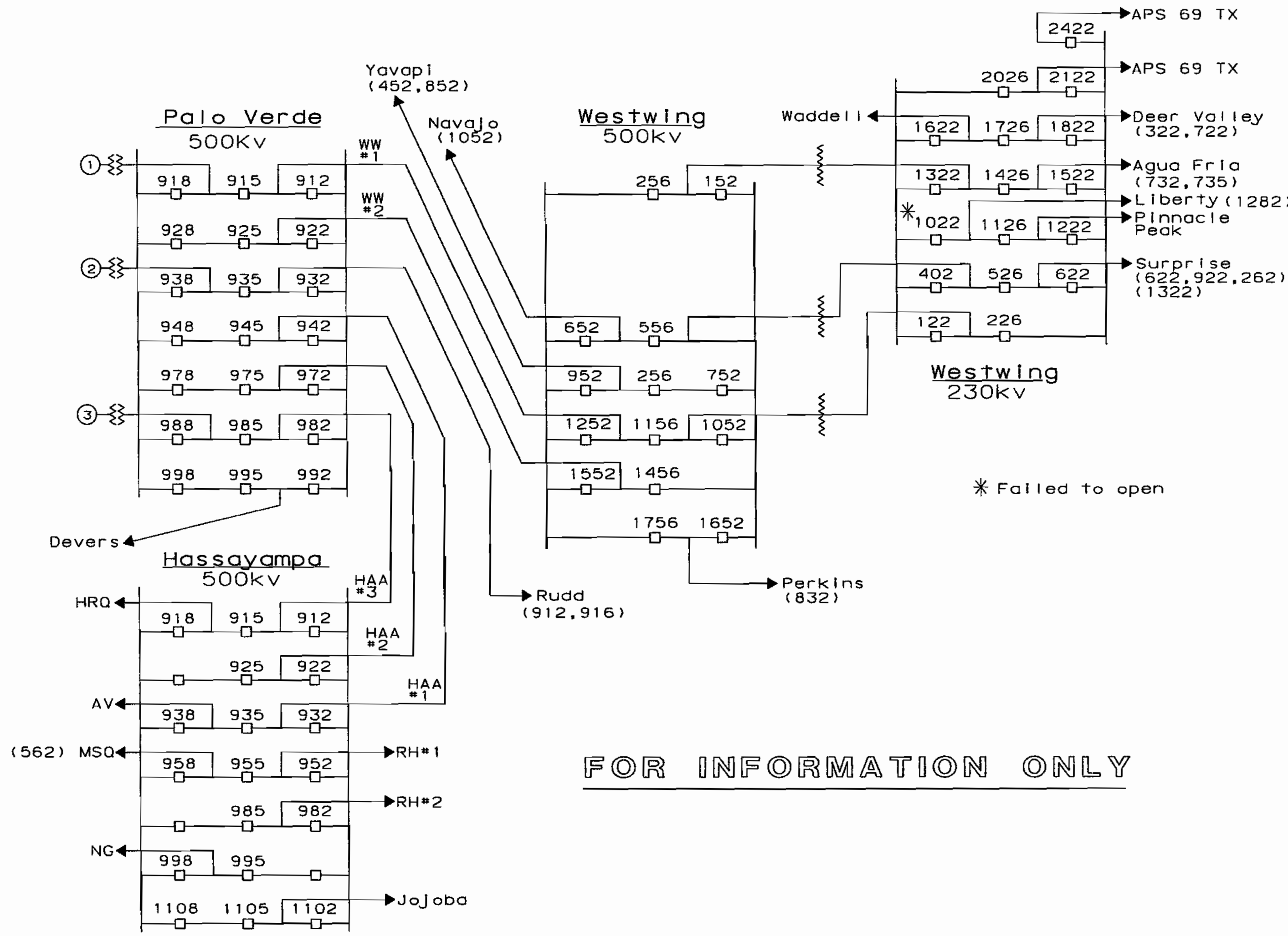
Root cause

- Bird excrement on 230 KV line insulator approx. 40 miles away from the plant
- Phase flashed over to the tower, dropped one phase above the other
- Fault began as phase to ground, then changed to phase to phase fault and then 3 phases to ground when the neutral wire dropped



BIRD
32'-0"
OVERHEAD GROUND WIRE
7'-0"
7'-0"
11'-0"
18'-0"
18'-0"
55'-0"
BODY
5'-30' LEG
EXT. IN 5'
INCREMENTS!
132'-0" MAX.





FOR INFORMATION ONLY

Design Deficiency

- One auxiliary relay failed to operate
- Redundant protective relay signals were wired to one aux. Relay even for breakers with two trip coils.
- Fault current from 230KV switch yard contributed enough current to trip 500KV breakers at West Wing Substation & Palo Verde 500KV substation

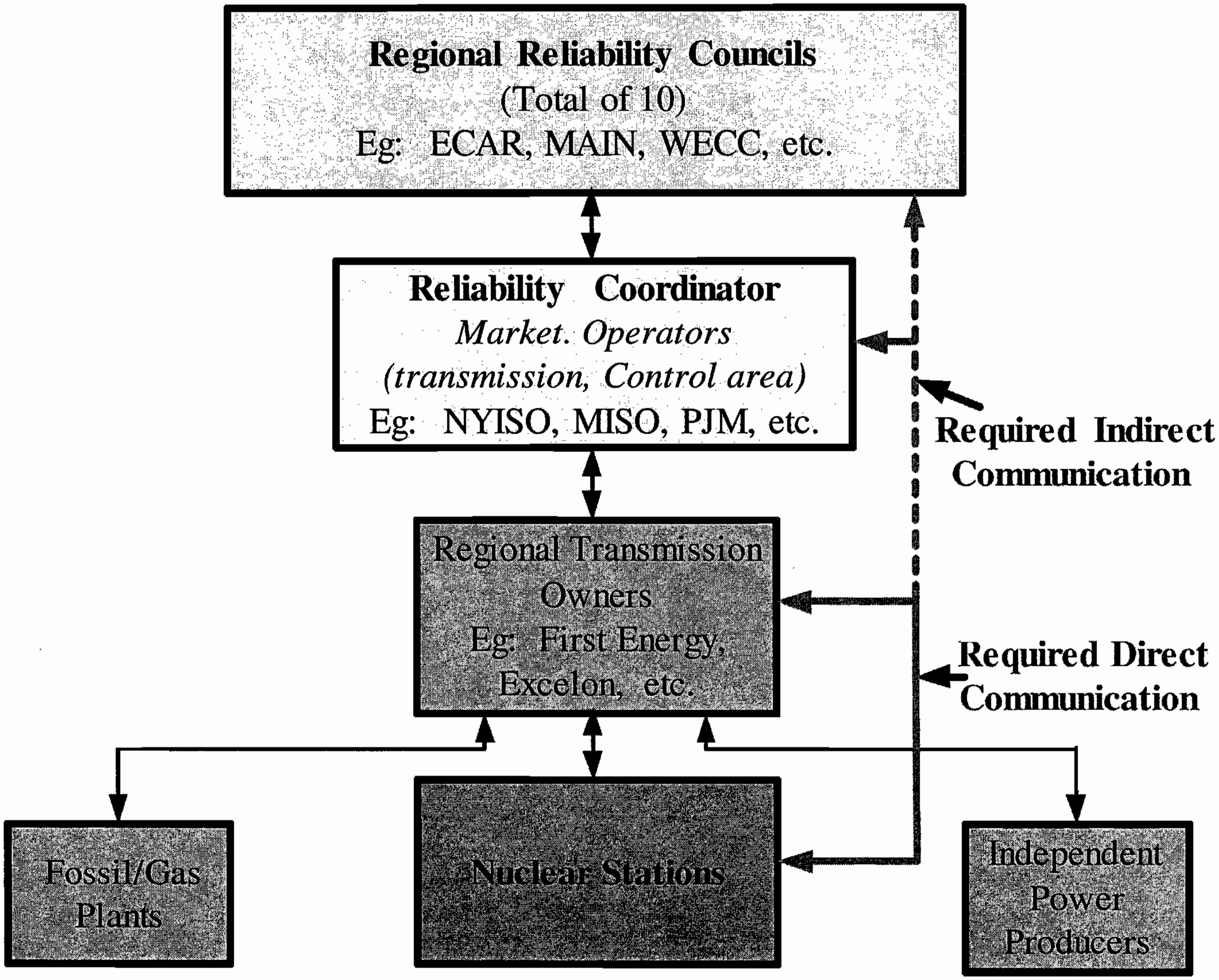
Corrective Actions

- Installed double relays for all breakers with 2 trip coils at 230 KV level for substations in the immediate neighborhood, others being done
- On breakers with only one single trip coil, APS planning to install a second set of trip coils
- Removed the second layer of protection from zero sequence relays on Hassayampa lines
- After conferring with the staff, APS agreed to add another set of zone 2 /ground current to protect the transition lines between 230 & 500 KV
- An automated response to 3 Unit trip is being developed at control center

NRR Actions

- Review licensee's study on power system stability (planned to be complete by March 2005)
- Non-public generic communication being processed to share the grid problems

Backup Slide



Presentation to the Advisory Committee on Reactor Safeguards

EARLY SITE PERMIT (ESP) REVIEW STATUS

Presented by:

Michael Scott

Nanette Gilles

Raj Anand

New, Research and Test Reactors Program

November 5, 2004

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Purpose

- Brief the Committee on status of ESP application reviews
- Discuss issues that have emerged during the reviews
- Discuss future milestones for ESP reviews, including Committee involvement
- Address Committee questions or comments

Agenda

• ESP review status	15 min
• ESP review issues	20 min
• Upcoming milestones	15 min
• Discussion / Committee questions	40 min

ESP Review Status

- Three applications received
  - Clinton 9/25/03
  - North Anna 9/25/03
  - Grand Gulf 10/17/03
- Staff informed applicants it would stagger review of the applications
  - North Anna first
  - Clinton review products two months later
  - Grand Gulf review products two months after Clinton
- Nearing completion of draft safety evaluation report (DSER) for North Anna ESP application (December 20, 2004) – on schedule

ESP Issues

- Applicants and staff have exercised Subpart A to 10 CFR Part 52, Subpart B to 10 CFR Part 100, and Review Standard RS-002, "Processing Applications for Early Site Permits" for the first time
- Issues have arisen regarding:
  - Tornado wind speed
  - Seismic analysis
  - Emergency planning

Tornado Wind Speed

- To define site design basis tornado wind speed, RS-002 allows use of:
  - Regulatory Guide (RG) 1.76 (360 mph wind speed east of Rockies)
  - 1988 interim staff position (300 or 330 mph east of Rockies depending on location)
  - Any site-specific wind speed justified by applicant
- Advanced reactor vendors have relied on SECY-93-087, which accepted use of 300 mph for design of advanced reactors
- All design certifications to date have assumed wind speed of 300 mph

Tornado Wind Speed (Continued)

- In the SRM for SECY-03-0227, Commission approved RS-002 but directed staff to re-evaluate this issue
- Staff response (SECY-04-0200) reports that:
  - Staff is re-evaluating maximum tornado wind speed based on new data
  - Staff recommends development of risk-informed approach
- SECY currently in Commission review
- When available, re-evaluation results will inform ESP reviews

Seismic Analysis

- Two of three applicants (North Anna and Clinton) advanced a "performance-based" approach for determining safe-shutdown earthquake (SSE)
  - Goal is the mean annual frequency of 10^{-5} of unacceptable performance of nuclear structures, systems, and components as a result of seismically initiated events
  - Methodology described in draft ASCE standard

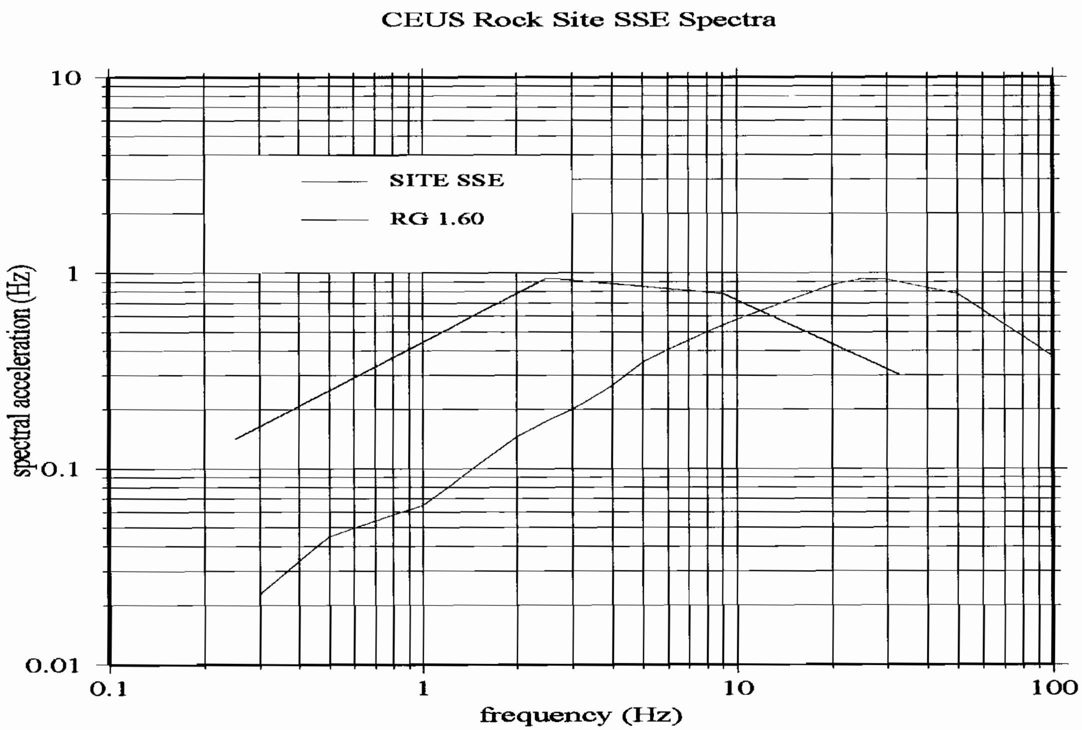
Seismic Analysis (Continued)

- Staff has not reviewed acceptability of new approach
- Staff informed applicants that additional review time would be needed
- North Anna application subsequently revised to use staff-approved method in RG 1.165
- Impact on Clinton review schedule still under discussion

Seismic Analysis (Continued)

- SSE at rock sites may exceed certified plant design SSE at high frequencies
  - Rock sites effectively transmit high-frequency ground motion
- Applicants for combined license (COL) or construction permit will need to address this issue

Typical Rock Site SSE vs Design



Emergency Planning

- All three applicants seek acceptance of "major features" of emergency plans
- Term defined only in draft guidance for review of emergency planning information at ESP stage (Supplement 2 to NUREG-0654)
- Industry has concerns regarding:
  - Finality associated with acceptance of major features at ESP stage
  - Level of detail in staff review relating to previously filed information
  - Review of non-applicant-controlled (i.e., State and local) emergency plans

Emergency Planning (Continued)

- NRC and FEMA have established Supplement 2 to NUREG-0654 as review standard applicable for "major features" of an emergency plan
- ESP applicant can obtain finality on description of major feature
- Details of implementation of major features will be evaluated at COL stage

Emergency Planning (Continued)

- Consistent with Commission policy, previously filed information will generally not be reviewed in detail
- State and local plans will be reviewed when applicant seeks approval of major features related to offsite emergency planning

Coming Milestones

- All safety-side reviews are on schedule
- Expect to seek Committee review of DSERs as follows:
  - North Anna
    - DSER to Committee early January 05
    - Subcommittee and full Committee meetings February/March 05
    - ACRS interim letter March 05
  - Clinton
    - DSER to Committee February 05
    - Subcommittee and full Committee meetings March/April 05
    - ACRS interim letter April 05
  - Grand Gulf
    - DSER to Committee April 05
    - Subcommittee and full Committee meetings May/June 05
    - ACRS interim letter June 05

Coming Milestones (Continued)

- Committee reviews of final SERs
  - North Anna July 05
  - Clinton September 05
  - Grand Gulf November 05

Conclusions

- ESP safety reviews on track
- Some challenging “first-of-a-kind” issues have arisen
- Compiling lessons learned for future reviews
- These lessons learned will be useful to staff and COL applicants even for cases in which COL applicant does not reference an ESP

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RSK Statement

04.03.2004

Incident at the Davis Besse nuclear power plant (USA) rated at INES Level 3 of 6 March 2002, "Boric Acid Corrosion on the Reactor Vessel Head" and lessons learned for German plants

1 Advisory request

The Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) issued an advisory request to the RSK Committee on REACTOR OPERATION regarding the incident at the Davis Besse NPP with the aim to investigate its applicability to German plants. The RSK Committee on PRESSURE-RETAINING COMPONENTS AND MATERIALS discussed material- and component-related issues, indicators for boric acid corrosion and non-destructive examinations (NDEs).

2 Background

On 6 March 2002, a corrosion-induced cavity was discovered at control rod drive mechanism (CRDM) nozzle No 3 of the reactor pressure vessel (RPV) during refuelling outage at the American Davis Besse nuclear power plant with pressurised water reactor. This corrosion cavity extended through the ferritic steel down to the cladding (size: 180 x 100 – 125 mm). At nozzle No 2, a crack was detected with a maximum width of 9.5 mm and a maximum depth of 90 – 100 mm.

During the 2002 outage, the control rod nozzles were inspected for cracking (NRC Bulletin 2001-01). The inspection of a total of 69 control rod nozzles, made of Alloy 600, revealed cracks in five nozzles; ten of these were through-wall cracks at nozzles No 1, 2 and 3. The repair measures initiated at nozzle No 3 had to be interrupted because the nozzle was displaced in the downhill direction (i.e. away from the top of the RPV head). After removal of the nozzle and the boric acid crystals deposited on the top of the RPV head, the above-mentioned corrosion cavity was discovered.

3 Course of the discussion

Discussions took place at the meetings of the RSK Committee on REACTOR OPERATION on 24.04.2002 and 22.01.2003. At meeting No 141, GRS reported to the Committee.

On 15 May 2002, the topic was dealt with by the RSK Committee on PRESSURE-RETAINING COMPONENTS AND MATERIALS at meeting No 25, where GRS and the plant manufacturer made their reports. At following meetings, the Committee was informed about the progress of the investigations on the American side. At meeting No 29 on 04.09.2002, the Committee was informed about the current status of the RPV head (construction, manufacture, quality, geometry, operation, material condition, NDEs, examination intervals, examination methods, monitoring of the RPV head area, especially with regard to leakages) of the German Obrigheim nuclear power plant (KWO) by oral reports from the licensee and the expert. At meeting No 30 on 09.10.2002, the Committee issued its statement.

At the 367th meeting on 13.11.2003, the RSK discussed the facts of the case. On this item, a written report on the previous meetings of the Committee was available.

141st meeting of the RSK Committee on REACTOR OPERATION on 24.04.2002 (consultancy document [1])

GRS reported on damages induced by boric acid corrosion on the RPV head of the American Davis Besse nuclear power plant [1]. During the 2002 outage, a corrosion cavity was detected at nozzle No 3 with corrosion wastage of the ferritic steel down to the cladding. In addition, a crack was detected at nozzle No 3 with a maximum width of 6 mm. Boric acid corrosion was assumed to be the damage mechanism.

At the Davis Besse nuclear power plant, flange leakage at the nozzles of the control rod drive mechanisms already occurred before. Regarding the detectability of leakages, GRS pointed out that at the Davis Besse nuclear power plant, an integral leakage rate of 23 l/h was considered as "normal" operational leakage in the primary circuit.

The cause of the corrosion was attributed to leakages in the area of the flanges and stress corrosion cracking in the RPV head nozzles made of Alloy 600. In the past ten years, leakages at flange connections occurred repeatedly. These had not always been removed during the outages. The "integral" leakage rate measured increased to twice the values measured in the years before.

Since the resulting boric acid deposits were not removed consequently, the deposits on the RPV head accumulated. The reactor vessel head insulation and the extensive deposits had obstructed a visual examination of the nozzle areas. In the year 2000, it was decided to remove these deposits. This, however, could not be fully performed due to hardening of the deposits for radiation protection reasons. Due to the residual deposits it was not possible to perform a complete visual examination of the nozzle area at the RPV head and so it was decided not to perform the examination, which is actually required to be performed within the frame of the monitoring programme on boric acid corrosion (NRC Generic Letter 88-05) completely (only at the accessible locations).

GRS explained the examination requirements regarding boric acid corrosion that exist at the plant. It was laid down in examination requirement "ASME XI (2001)" that a leak test and a visual examination for leakage were to performed after each refuelling. NRC Generic Letter 88-05 stipulates an identification of potential small leak locations with regard to corrosion to prevent large leaks in the pressure-retaining boundary. This provision would also include descriptions of different methods for the localisation of small leaks and for estimation and prevention of corrosion damages. Further, it referred to the maximum corrosion velocity, reached in tests, of 120 mm/a for ferritic steels.

In another NRC Bulletin (NRC 2001-01), the possibility of circumferential cracking of nozzles made of Alloy 600 due to primary water stress corrosion cracking (PWSCC) was pointed out. According to a plant-specific assessment of NRC under consideration of the RPV head operating temperature, the operating time and the material, the control rod nozzles at the Davis Besse nuclear power plant are susceptible to PWSCC. As a consequence, the plant was to be subjected to a 100 % visual examination of the nozzles from the top or a volumetric examination from the bottom by the end of 2001. In agreement between NRC and the licensee, the volumetric examination was then performed in February/March 2002 within the frame of a refuelling outage.

During examination by means of ultrasonic testing methods from the bottom it was detected that five nozzles had through-wall cracks (four axial cracks and one circumferential crack at nozzle No 3). While repairing the RPV head, nozzle No 3 loosened. After removal of the reactor vessel head insulation, severe damages were detected in the area of this control rod nozzle.

There were boric acid corrosion indicators as early as 1999 in form of brownish crusts and deposits on the RPV head and the containment air coolers which was a clear sign of iron borate formation. The licensee attributed the increase and brownish colouring of the deposits on the containment air coolers to corrosion processes on the coolers. Due to increasing deposits of similar type on the filters, the filters were changed every day instead of every month.
From this, NRC had drawn the conclusion that leakages in combination with deposits were the cause of this incident. Further, several indicators for boric acid corrosion processes at the RPV head had not been realised by the licensee. The corrosion cavity was mainly due to a leakage at nozzle No 3. These processes started at least four years before their detection. The role of deposits regarding the corrosion growth, the specific chemical processes of corrosion, the impact of the RPV head operating temperature and the velocity of crack growth on the nozzles of the Davis Besse nuclear power plant were unclear points.

Further, GRS explained the following:

- In response to the incident, NRC had ordered a more stringent performance of the corrosion management programme at the American plants. The licensees had to submit corresponding reports by the end of April.

- In Germany, comparable integral leakage rates in the containment alone do not lead to a reportable event. However, they are observed by the respective regulatory authority.

With the concentrations normally present in the primary circuit, the behaviour of the boric acid would not be aggressive. Due to the evaporation process, the concentration in the area of leakages increases and pH-values of 3 could be reached at temperature of 100 - 150 °C. Under these conditions, boric acid behaved quite aggressive towards ferritic steels. In each case, however, the precondition for corrosion processes was humidity. Ferritic steels would practically be dissolved by boric acid corrosion, stainless steels (chromium content > 13 – 15 %) were virtually immune to this corrosion. For this reason, the cladding at Davis Besse had not suffered any damage. In addition, there are also indications of a considerable corrosive effect of "dry" boric acid which eliminated water at temperatures above 185 °C, thus being converted into a viscous liquid.

- Due to the deposits, the nozzle area had not been subjected to a 100 % examination, at least in the year 2000. How the NRC arrives at four years for the incident in its assessment was not traceable by means of the available documents. Obviously, the period for the corrosion processes was derived from the maximum corrosion velocity, determined by experiments, and the thickness of the RPV head.

- The existing degree of boric acid corrosion on the nozzle was not clearly identifiable by means of a visual inspection of the RPV from the top due to the deposits.

- In some cases, the leakages also resulted in the distribution of boric acid in gaseous form and deposition on other locations (e. g. on the control rod drive mechanism).

- Compared to American plants, there are no reportable events due to damage caused by boric acid corrosion.

The Committee added that the boric acid corrosion problems in the past had been extensively discussed at the RSK by the Committee on PRESSURE-RETAINING COMPONENTS. As a result of these discussions, examinations of RPV nozzles were performed at all German plants. Except for the KWO plant, the nozzles at all German PWR plants were provided with double seals which were subjected to leakage tests after installation. At KWO, a special reactor vessel head leak monitoring system was installed, being more sensitive than the integral leak monitoring system at other plants.

25th meeting of the RSK Committee on PRESSUER RETAINING-COMONENTS AND MATERIALS on 15.05.2002 (consultancy documents [2 to 7])

In accordance with the information published on the Internet and the reports given at meeting No 141 of the RSK Committee on REACTOR OPERATION on 24.04.2002, GRS added the following:

Retrospectively, different indicators of significant corrosion/leaks on nozzles have been occurred repeatedly since 1998. The NRC assumed that further leakages were hidden by the deposits resulting from leakage at flanges. The signs of corrosion at the RPV head had not been detected. The cavity at nozzle No 3 had mainly been caused by the leakage. The wastage had at least begun four years before detection. There were still unclear points regarding the role of the deposits, the chemical processes of corrosion, the impact of the RPV head operating temperature and the crack growth in the nozzle.

The plant manufacturer Framatome ANP reported on construction-, manufacture- and material-related aspects of nickel-based alloys, especially on RPV head penetrations made of Alloy 600. The influence of the nickel content regarding susceptibility to intergranular stress corrosion cracking (IGSCC) in high-temperature water, the boric acid corrosion on the RPV head of the Davis Besse nuclear power plant, the boric acid leakage at the inspection holes of the plant, and the indications of boric acid corrosion detected there, the time history of the damage occurred, and the corrosion velocities were discussed. Globally seen, there was the following situation regarding the reactor vessel head penetrations and its applicability to Siemens/KWU plants:

Foreign plants	Siemens/KWU plants
Alloy 600 nozzles	compound tube (St52/1.4550 cladded)
contraction joint	threaded joint (trapezoidal thread bearing the loads)
susceptible to IGSCC	not susceptible to corrosion

In particular, the comparison is as follows:

	Davis Besse	Obrigheim NPP (KWO)	Siemens/KWU PWR plants
Nozzle Material	Alloy 600	Alloy 600	Compound tube (St52/1.4550 cladded)
Solution annealing	870-930° C	1,000-1,050° C	
$Rp_{0,2}$	340 N/mm^2	257 N/mm^2	
Mounting	Shrinking/ weld-in	Shrinking/ weld-in	Shrinking/ weld-in
Weld metal	Alloy 182	Alloy 82/182	Austenite24/12 (seal weld)
Stress-relief heat treatment	none	600° C, > 10h	none
IGSCC susceptibility	high high tensile stress from manufacture (tube) and shrinking	low low tensile stress due to stress-relief heat treatment	not given

The ferritic RPV head nozzles are welded with nickel-base weld metal (Alloy 182, unbuffered) with the austenitic nozzle flange: The root is normally austenitic (material 1.4551). At the pressure tubes, the circumferential welds RN 1 and RN 2 are unbuffered dissimilar welds with nickel-base weld metal (Alloy 82/182, normally medium-swept); the circumferential welds RN3 and RN 4 are austenitic (material 1.4551).

At Siemens/KWU PWR plants, the design of the flange seal between control rod nozzle and pressure tube was considerably more elaborate with regard to safety compared to the seal design of the US-American manufacturer Babcock und Wilcox due to an internal and external conoseal lining made of the material No

1.4541 and an instrument line (leakage monitoring). At Siemens/KWU PWR plants, the load-bearing function (trapezoidal thread) and the sealing function (seal weld) had consequently been separated (exception: KWO plant).

The report of the company IntelligeNDT and the plant manufacturer Framatome ANP on the inspection of RPV heads at Siemens/KWU PWR plants, first gave an overall survey of the inspection areas of the RPV heads and the pressure tubes. The basic principle, the manipulator of the ligaments between the RPV head nozzles, the data evaluation and the areas for the ligament examination were presented. The examination areas reached with a creeping wave phased array probe at the external surface of the RPV head were explained. They led to full examination coverage at the external surface. Regarding the examination of the ligaments between the RPV head nozzles (volume and internal surface), the phased array technique covered a large part of the volume and all ligaments at the internal surface. The detectability of discontinuities in the surfaces corresponded to a groove of 3 x 20 mm minus 6 dB and CRR 3 mm in the volume. In the area of the ligaments between the RPV head nozzles, there were no recordable indications at Siemens/KWU PWR plants. Since 1989, examinations were performed by means of the phased array technique. Since 2002, ultrasonic tests and integral visual examinations have been performed simultaneously.

Regarding the pressure tube inspection, the EMUS (electromagnetic ultrasonic) technique and the eddy-current test were mentioned, and regarding the in-core instrumentation nozzles the inspection with eddy current rotating probes. The inspection could be performed from the external surface by means of the EMUS technique, and the internal surface by means of the eddy current technique. Welds No 2, 3 and 4, the control rod nozzle welds and the in-core instrumentation nozzle welds were examined by means of the eddy current technique (internal surface examination). The detectability of discontinuities in the surfaces corresponded to a groove of 0.5 x 10 mm with eddy current and with EMUS CRR 3 mm in the volume. There were no recordable indications at Siemens/KWU PWR plants. If required, a visual examination was performed with regard to surface indications. Circumferential weld No1 of the pressure tubes can only be examined after dismantling of the pressure tube due to the latch unit installed in the interior.

The weld examination (circumferential weld) of the RPV head was performed by means of a phased array tandem system as well as 45° ET and 70° SEL, so that longitudinal and transverse discontinuities in the circumferential welds and the high-stress areas in the transition area to the head flange can be detected. Both circumferential welds were subjected to full examination. There were no indications to operational damages at Siemens/KWU PWR plants. The detectability of discontinuities in the surfaces corresponded to a groove of 3 x 20 mm minus 6 dB and CCR 3 mm in the volume.

According to the explanations given by the reports presented and the representative of the utility EnBW, the German operators of PWR plants have intensively dealt with the examination of the RPV head, among others, also with regard to the requirements of the competent *Land* authority. Regarding materials, construction and manufacture, there were no indications that the incident at the American Davis Besse nuclear power plant can be applied to German plants. In addition, it was pointed out that in the next months, an intensive exchange of experiences with US-American experts will take place.

The results of stress corrosion cracking analyses for Alloy 500 in PWR coolant, presented to the RSK Committee on PRESSURE-RETAINING COMPONENTS AND MATERIALS at the meeting, show, according to the rapporteur, a clear dependence of the crack growth data on the temperature. The area determined here – as in many other cases – in which the velocity of the crack growth by stress corrosion cracking is almost independent on the stress intensity factor, is about 10^{-9} m · s^{-1} (equivalent to about 30 mm per year) for 350° C, about 3 · 10^{-10} m · s^{-1} for 320° C, and about . 3 · 10^{-11} m · s^{-1} for 290° C, for basic material and weld metal each. Similar measurements of the crack growth velocity were also achieved for the stress-relief heat-treated, the cold worked and not cold worked condition.

In the subsequent discussion, the Committee dealt with the following aspects:

- Causes (materials, stress corrosion cracking, cracks, material Alloy 600, etc.),
- detectability (indicator quality),
- non-destructive examinations (NDEs):
  - accessibility,
  - walk-down, walkability,
  - visual examinations (Preferably after shut-down? Or also during start-up during leakage test?),
  - examination intervals,
  - examination methods,
- time behaviour (e. g. corrosion velocity and the like) regarding the question whether a plant affected by boric acid corrosion can be operated for a cycle without any safety concerns,
- procedures/testing manual. Are revisions required? (from the point of view of both committees)

29th meeting of the RSK Committee on PRESSURE-RETAINING COMPONENTS AND MATERIALS on 04.09.2002 (consultancy documents [8-14])

The operator of the KWO plant presented a safety assessment of the RPV head nozzles. The analyses and methods performed by the licensee from the first event at the French NPP Bugey 3 in the year 1991 until the outage in 2002, inclusively, were described. The main statements of the documents presented to the RSK Committee on PRESSURE-RETAINING COMPONENTS at meeting No 190 on 06.12.1991 and meeting No 193 on 06.03.1992 for the assessment of the RPV head nozzles of the KWO nuclear power plant were presented.

The measures continued by the licensee since 1992, such as calculations, installation of leakage detection systems and non-destructive examinations, were mentioned.

The licensee explained the initial situation as well as the analyses and assessments for the period from 1992 to 2002. The latter concerns construction/manufacture/quality, loads, stress and fatigue analysis, visual examinations and non-destructive examinations. The arrangement of the RPV head nozzles at the KWO nuclear power plant was presented. According to the licensee, the spare nozzles located in the peripheral area were subjected to a full examination over the length. The licensee addressed welding of the nozzles into the RPV head and the respective weld method, and compared it, regarding construction/manufacture/quality, with the Davis Besse nuclear power plant. With regard to former indications in nozzles at French plants, the operator of the KWO plant addressed the comparison of nozzle loads during operation (comparison between KWO and Bugey, Unit 3), factors promoting crack initiation at French plants and factors preventing crack initiation at the KWO plant.

In connection with the analyses for keeping up the operating licence, the operator of the KWO plant declared that finite element calculations were performed for the weld-in area of the RPV head nozzle; according to the plant operator, the maximum cumulative usage factor from the fatigue analysis was D = 0.24 for the weld nozzle/head and D = 0.02 for the ferrite near the weld. Analyses on crack growth induced by stress corrosion cracking für the material Alloy 600 in PWR coolant were performed.

The operator of the KWO plant compared the flanges of the RPV head nozzles KWO/Davis Besse and presented the leakage rates at the US-American Davis Besse plant from January 2000 until January 2002. This was compared to the flange leakage rates (seals) at the KWO plant in the year 1994. In addition to global methods, the two operational local leakage detection systems for the RPV head BLISS (Bartec Leakage Indication Sensor System: self-monitoring and located in the area of the RPV head nozzle flange for monitoring of the flange connection YA01 M001 (sensor tube)) and FLÜS (Feuchte-Überwachungssystem – humidity monitoring system), located in the area of the RPV head nozzle YA01 M010 (sensor tube), are used for leakage detection; upon inquiry of the Committee, information was given on the technical data and the capacity. According to the plant operator is BLISS a leakage detection system with yes/no indication and a response threshold of about 10 l/day, used since the beginning of the nineties; the FLÜS system had been used since 1995. According to the competent *Land* authority, the Ministry for the Environment and Transport - Baden-Württemberg (UVM B-W), tests for the determination of the response thresholds with FLÜS resulted in about 1 l/hour.

Finally, the operator of the KWO plant addressed the visual inspections in the area of the RPV head and the further NDEs for nozzles and RPV head, as well as the special tests on nozzles in the year 1992 and the eddy current tests on nozzles (1992, 1994/2000). On the basis of the examinations and tests performed in the last ten years and leakage monitoring during operation, the plant operator stated that there were no indications of incipient cracks or leakages on the nozzle tubes of the RPV head.

Due to its high sensibility in case of operational leakages, the very sensitive global and local leak detection, the accessibility and the regular visual inspections, as well as the comprehensive additional NDEs (recurrent inspections), the plant operator is of the opinion that a damage mechanism comparable to the incident at the Davis Besse plant can safely be excluded.

With regard to the leak rate of 30 l/day (from 06.01.2000 to 02.01.2002) at the American Davis Besse plant with peak values of up to 80 l/day (from 10.02.2001 to 12.02.2001), the typical basic leak rate for the KWO plant with about 10 to 15 l/day and the sensibility of the plant management in case of clear trends towards higher values which may lead to plant shut-down was addressed in the discussion. On the part of the UVM B-W, reference was made in this respect to a notification of the plant operator prior to plant shut-down due to leakages at the flange seals of the nozzles in position H12 and E05 in the year 1994. Regarding the construction-, manufacture- and material-related aspects and the seal design of the control rod nozzle at PWR plants in Germany, reference was made to meeting No 25 of the Committee on PRESSURE-RETAINING COMPONENTS AND MATERIALS on 15.05.2002.

At present, a visual inspection of the RPV head (outside) in German nuclear power plants is not mandatory.

It was reported to the Committee that in response to the incident at the American Davis Besse plant examinations were performed at several German plants. With respiratory protective equipment used by the inspector, the reliability of the results of the inspection, especially of the nozzle field, was to be regarded as impaired. It was recommended to perform a mechanised visual inspection with video recording at intervals of four years..

With the available special test probe it was possible to examine the control rod nozzles from the inner side with eddy current if the width of the gap between control rod guide tube and the nozzle is larger than 1.5 mm. This was not given for all PWR plants. Regarding the incident at the Swedish Ringhals NPP, Unit 4 (reporting of GRS at meeting No 23 of the Committee on PRESSURE-RETAINING COMPONENTS AND MATERIALS on 17.01.2001), the Committee is of the opinion that the non-destructive examination (NDE) of the medium-swept welds, which currently is not feasible in some cases, should be improved.

The representative of the expert organisation TÜV Energie- und Systemtechnik GmbH Baden-Württemberg presented a safety assessment of the RPV head nozzle penetrations of the KWO plant; from his point of view, the preceding comprehensive report of the plant operator was appropriate. The rapporteur briefly described the activities related to expert opinions on the KWO plant; in the years 1991 to 1994 detailed investigations were performed which had been confirmed by the results of non-destructive examinations (NDEs). With regard to the statement of the TÜV Südwest of 17.01.1994, the expert addressed – for comparison of the Alloy 600 head nozzles at the plants Bugey, Unit 3/Davis Besse/KWO regarding their susceptibility to stress corrosion cracking – the wall thickness of the RPV head, the spherical radius, the nozzle design, the pressure test, the material and the type of manufacture. Further, he compared installation, weld-in, weld volume and the performed stress-relief heat treatment and gave a survey of the NDE of the RPV head of the KWO plant with participation of the TÜV which had not revealed any indications of incipient cracks or leakages. The expert summarised his assessment that from his point of view there are two main factors for the resistance of the Alloy 600 nozzles against stress corrosion cracking at the KWO plant:

- Due to construction and manufacture, the operational stress maxima for the inner side of the nozzles in the weld-in area are considerably lower compared to the nozzles at comparable French and US-American plants.

- Due to the type of manufacture (usage of hot worked tubes), the material condition is more homogeneous and less susceptible.

This assessment was confirmed by the results of the comprehensive NDEs.

The representative of the UVM B-W gave the additional information that the reported facts of the case, at that time had been dealt with in co-operation with the RSK. On the part of the competent *Land* authority there were no supplements to be made to the reports of the licensee and the expert organisation.

148th meeting of the RSK Committee on REACTOR OPERATION on 21.01.2003 (consultancy documents [15-17])

In its assessment, the RSK Committee on REACTOR OPERATION agreed with the recommendations in the GRS Information Notice on the incident [17].

367th RSK meeting on 13.11.2003 (consultancy document [18])

The BMU stated that the RSK should assess the incident sequence at the Davis Besse plant. In its statement, the RSK should address the causes and assess each of them individually. In this respect, the relevant individual aspects, as e. g. the behaviour of the licensee, in comparison with Germany, the damage sequence, etc., should be addressed.

4 Assessment criteria

The general safety requirements are based on the safety criteria of the Federal Ministry of the Interior (BMI), the RSK guidelines for pressurised water reactors and the respective KTA standards.

For the assessment, the RSK considered the following aspects:

Boron deposits which may lead to corrosion attacks on safety-relevant components/systems must be prevented and existing deposits have to be detected and measures against further deposits be implemented in time.

For the assessment of the fulfilment of this requirement, the following principles have to be referred to:

- Prevention of leakages by corresponding precautions in construction, material selection, mode of operation and maintenance management,
- early detection of leakages by continuous monitoring and recurrent inspections,
- functioning safety management in such a way that in case of indicators given with regard to primary system leakages, deposits, indications of corrosion etc., efficient measures for the prevention of inadmissible consequences are immediately initiated.

The RSK examined whether these requirements were considered in the assessment according to the state of the art in science and technology and whether the analyses are plausible with regard to the facts of the case presented and explained.

5 Safety-related assessment

The RSK draws the conclusion that due to the fast progressing corrosion, the incident is of high safety significance because all of the four safety objectives are concerned. Under the incident circumstances, a failure of the wall in this area would have led to a loss of coolant accident (LOCA) with leakage at the RPV where a simultaneous ejection of the control rod concerned cannot be excluded. The special significance is also due to the fact that the operator of this plant underestimated the corrosion type and rate although the mechanism was known. In addition, there were numerous indications at the plant, as e. g. massive accumulation of corrosion products in coolers and filters which would have had to lead to a clear determination of the causes and initiation of countermeasures. On principle, the corrosion mechanism can also be applied to German plants. Regarding the cause and the maintenance deficits, a direct applicability is not given. This issue will be addressed more detailed in the following. The incident was primarily due to a massive maintenance deficiency and not only to the material mechanics.

The RSK points out that the authorities and experts working in the frame of licensing and supervision in Germany accompany plan operation more closely. In the past, the problems associated with corrosion processes have extensively been discussed at the RSK by the Committee on PRESSURE-RETAINING COMPONENTS. As a result of these discussions, the RPV head nozzles at all German plants were inspected. Except for the KWO plant, the nozzles at all German PWR plants of the manufacturer Siemens/KWU were provided with double seals and instrument lines (leakage detection) which were subjected to leakage tests after installation.

At KWO, a special reactor vessel head leak monitoring system being more sensitive than the integral leak monitoring system at other plants was installed. Further, it has to be pointed out that the construction (control rod nozzles/RPV head) and the material of the control rod nozzles of Siemens/KWU PWR plants are not comparable to the Davis Besse plant.

In comparison to the American plants, there were only very few events with boron corrosion at German plants. On this issue, a *Länder* survey was prepared on behalf of the BMU (corrosion indications at RPV head at a US-American plant, letter of the BMU to GRS of 3 April 2002 with enclosures (statements of the *Länder* on an inquiry of the BMU – RS I 3 – 14 200/1)), in which it was explicitly confirmed that at German PWR plants – for BWR plants, boron corrosion is not possible during normal operation due to the technical design – no leakages occurred at head nozzles.

Prevention of leakages

According to the present state of knowledge, the damage observed at the US-American Davis Besse nuclear power plant (stress corrosion cracking of the Alloy nozzles resulting in boron-induced surface corrosion) is not to be expected for German plants, except for the Obrigheim nuclear power plant (KWO), due to the construction – screw-in nozzles with welded joint – and due to the material-related boundary conditions. The limitation regarding the KWO plant explicitly refers to the material. After the detection of cracks in the nozzles of the pressure tubes of the control rod drives at the French plants Bugey (Units 1 and 3) and Fessenheim (Unit1) in the year 1991, the German plants were inspected without indications. At the KWO plant, the nozzles were subjected to a full examination. The RPV head penetrations installed at KWO are more resistant against stress corrosion cracking compared to comparable French and US-American plants due to conditions given by construction and manufacture type (e. g. low stress level)..

In general, through-wall cracks in components or, above all, leakages at seals cannot be excluded. In case of leakage of boric coolant (at BWRs not given in safety-significant areas with regard to integrity), boron deposits may occur. At German PWR plants, boron deposits occurred, among others, in safety valves of the primary system and in some cases also in case of small leakages. There are no cases known to the RSK where deposits led to significant corrosion. This is an indication of a well-functioning maintenance management system at German plants in case of detected leakages.

Detection of leakages

The external areas of the RPV head are annually subjected to a visual inspection within the frame of the refuelling outage. For this purpose, the external insulation is removed. The accessibility of the RPV head area at German plants is better compared to US-American plants because there is no weld-on "jacket" or "collar" as it is the case for American plants. At German plants, the insulation (insulation cover) is completely removable.

In Germany, the intervals for non-destructive examinations (NDEs) of the RPV head are, in accordance with KTA Safety Standard 3201.4, four or five years. The circumferential welds and the basic material areas of the ligaments between the RPV head nozzles are subjected, in accordance with the KTA safety standards, to ultrasonic tests and in the other areas to visual examinations (integral visual examination). The ultrasonic tests are performed at the internal and external surfaces and volumetrically. According to a declaration of intent by the operators of PWR plants, the test equipment is based on the phased-array technique which, at present, is the most advanced and progressive technique. Further, integrals visual examinations are performed by which deposits and discolorations would be detected.

Due to the monitoring system at German plants at the flanges of the pressure tubes, leakages are detectable. this has also be confirmed by experiences made with leakages in the area of the RPV head. In case of leakages, their causes immediately have to be clarified and corrective measures performed.

At the KWO plant, there are the two very sensitive leakage detection systems BLISS and FLÜSS in the area of the RPV head. The reports submitted showed that with these systems, a sensitivity for operational leakages and their potential impacts on normal operation was developed to the required degree.

Compared to the US-American Davis Besse plant, the conditions at KWO are also considerably better due to the non-destructive examinations (manufacturing and initial tests and recurrent inspections).

As a general rule, the screwed joints at the other components of the primary system are visually inspected after each loosening (direct visual examination) in accordance with the KTA safety standards. In addition an integral visual examination is performed within the frame of a plant walk-down. By means of these examinations, leakage indications and deposits as well as discolorations at not isolated surfaces of the components can be identified.

Safety management

An essential part of a functioning safety management is the implementation of an efficient safety-directed maintenance management. This begins with preventive maintenance and also includes efficient leakage detection (see above) and the implementation of measures for the prevention of inadmissible consequences. At the Davis Besse plant, there were sufficient knowledge and indicators for unexpected corrosion processes which absolutely would have given reason for taking action.

In case of leakages in other areas, especially in the area of detachable connections, the maintenance management has to ensure that no boron-induced corrosion processes occur.

Also due to the fact that the supervisory authorities and their authorised experts closely accompany plant operation and due to the respective rules and regulations it is not to be expected that the cause of such clear indicators, such as the large accumulation of corrosion products in coolers and filters, will not be investigated adequately.

6 Recommendations

Altogether, it is to be stated that the corrosion problem generally is of increasing safety significance during the operation of nuclear power plants for the further lifetime. This is shown by the incident at the American Davis Besse plant. Due to the fast progressing corrosion in the RPV wall, the incident is of high safety significance because all of the four protection objectives are concerned.

On the basis of the conditions at German nuclear power plants, the RSK recommends the following measures:

(1) During each refuelling outage, an integral visual examination (direct visual examination) is to be performed after removal of the external insulation of the RPV head; the result of this examination has to be documented.

(2) During each recurrent inspection of the ligaments between the RPV head nozzles, a visual examination of the nozzle area is to be performed (visual examination according to DIN 25435-4).

(3) Within the frame of the recurrent inspections, the areas of the other primary system components additionally have to be subjected to a visual examination (direct visual examination) after removal of the insulation

(4) For component areas with medium-swept welds made of nickel-based alloys where removal of the insulation is not scheduled (e. g. heating rod nozzle at the pressuriser, instrument nozzles, small-bore pipes), integral visual examinations have to be established.

(5) In case of anomalies during operation, such as increased accumulation of corrosion products in the room air (filters), the cause has to be clearly identified and further measures have to be determined.

(6) In case of indications of leakages and boric acid deposits in the primary system, the area concerned has to be inspected with regard to the cause and potential corrosion attacks.

(7) In case of indications of potential boron leakages in the immediate area of the RPV, especially in the area of the reactor cavity, due to leak water accumulation with boric acid or deposits of corrosion products, corresponding measures have to be initiated to clarify the cause.

(8) The scopes of the examinations required above have to be laid down in the testing manual. The measures to be taken in response to (5), (6) and (7) have to be specified in the instruction manual (Betriebshandbuch - BHB).

For the rest, the RSK agrees with the recommendations of the GRS Information Notice WLN 02/2003 an.

7 Documents, information and expertise

The following documents, information and expertise were referred to:

[1] Borsäurekorrosion am Reaktordruckbehälterdeckel der Anlage Davis Besse
Vortragsfolien zum Bericht der GRS
Tischvorlage zur 140. Sitzung des Ausschusses REAKTORBETRIEB am 24.04.2002

[2] Kopien von Folien, die von der GRS in der 141. Sitzung des RSK-Ausschusses REAKTORBETRIEB am 24.04.2002 gezeigt wurden

[3] RSK-Information DKW25/8
Korrosionsmulde im Deckel des Reaktordruckbehälters im US-amerikanischen Kernkraftwerk mit Druckwasserreaktoren Davis Besse, Block 1, entdeckt im Brennelementwechsel Februar/März 2002

[4] A. Roth, G. König:
Benetzungen von Bauteilen aus un- und niedriglegierten Stählen in DWR-Anlagen mit borsäurehaltigen Medien und deren Folgen – Betriebserfahrungen und aktueller Stand von Wissenschaft und Technik zur Korrosion von un- und niedriglegierten Stählen in wässrigen Borsäurelösungen
25. MPA-Seminar, Stuttgart, 07./08.10.1999
Vortrag Nr. 47

[5] Kopien von Folien, die von der GRS in der 25. Sitzung des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN UND WERKSTOFFE am 15.05.2002 gezeigt wurden

- Borsäurekorrosion am Reaktordruckbehälterdeckel der Anlage Davis Besse
- Große Korrosionsmulde im RDB-Deckel
- Befunde
- Reactor Pressure Vessel Head Degradation Location
- Photo of Degraded Area Adjacent to Nozzle 3
- Cross-Sectional Sketch of Degraded Area Adjacent to Nozzle 3
- Nozzle 2 Metal Loss
- Allgemeine Prüfanforderungen der WKP nach ASME XI (2001)
- Generic Letter 88-05: Programm gegen Borsäurekorrosion, Ziel: Vermeidung großer Lecks in der DFU
- RPV Head Configuration
- Access Openings

- Flanschleckagen und Leckraten
- Ergebnisse der visuellen Inspektionen bei Revisionen
- NRC Bulletin 2001-01: Umfangsrissbildung an Deckelstutzen
- Inspection Results
- Ergebnisse der US-Prüfung nach NRC Bulletin 2001-01
- Findings
- Indikatoren für signifikante Korrosion/Lecks an Stutzen
- Schlussfolgerungen der US NRC

[6] Deckelprüfung in KWU-Druckwasserreaktoren, Vorlage zur 25. Sitzung des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN UND WERKSTOFFE am 15.05.2002, Kopien von Folien, IntelligeNDT Framatome ANP

[7] Kopien von Folien von Framatome ANP:
- Nickelbasislegierungen, Alloy 600 RDB-Deckeldurchführungen
- Einfluss des Nickelgehalts auf die IGSCC-Anfälligkeit in Hochtemperaturwasser
- RDB-Deckel Borsäurekorrosion, Konstruktion Aufbau RDB Davis Besse
- RDB-Deckel Borsäurekorrosion, Fotos Davis Besse
- RDB-Deckel Borsäurekorrosion, Borsäureaustritt Inspektionsöffnungen, Fotos
- RDB-Deckel Borsäurekorrosion, Hinweise auf Borsäurekorrosion
- Probable Timeline
- Davis Besse Estimated Reactor Vessel Closure Head Corrosion Rates
- RDB-Deckeldurchführungen
- Übertragbarkeit auf S/KWU-DWR-Anlagen
- Dichtungsausführung Steuerstab-Stutzen

[8] Auszug TOP 4 des Ergebnisprotokolls der 193. Sitzung des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN am 06.03.1992

[9] Auszug TOP 6 des Ergebnisprotokolls der 195. Sitzung des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN am 04.05.1992

[10] Auszug TOP 3 des Ergebnisprotokolls der 207. Sitzung des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN am 11.05.1993

[11] Auszug TOP 3 des Ergebnisprotokolls der 219. Sitzung des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN am 18.05.1994

[12] Auszug TOP 8 des Ergebnisprotokolls der 287. RSK-Sitzung am 14.09.1994

[13] Sicherheitstechnische Bewertung der Deckelstutzen des Reaktordruckbehälters, Vorlage des Kernkraftwerks Obrigheim zur 29. Sitzung des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN UND WERKSTOFFE am 04.09.2002 in Bonn, 15.08.2002

[14] Tischvorlage des TÜV Süddeutschland zur 29. Sitzung des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN UND WERKSTOFFE am 04.09.2002 in Bonn, 03.09.2002

[15] Anlage 1 zum Ergebnisprotokoll der 30. Sitzung des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN UND WERKSTOFFE am 09.10.2002
STELLUNGNAHME des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN UND WERKSTOFFE
„Korrosionsmulde im Deckel des Reaktordruckbehälters im amerikanischen Kernkraftwerk Davis Besse Block 1"

[16] Ergebnisprotokoll der 29. Sitzung des RSK-Ausschusses DRUCKFÜHRENDE KOMPONENTEN UND WERKSTOFFE am 04.09.2002 (TOP 6)

[17] Gesellschaft für Anlagen und Reaktorsicherheit (GRS)
Weiterleitungsnachricht 2003/02:
Große Korrosionsmulde im Reaktordruckbehälter-Deckel des Kernkraftwerkes Davis Besse (USA)

[18] RSK-Information RSK365/9
Vorkommnis der INES-Kategorie 3 im amerikanischen Kernkraftwerk "Davis Besse" vom 06.03.2002, "Borsäurekorrosion am Reaktordruckbehälterdeckel"
SCHRIFTLICHER BERICHT der RSK-Ausschüsse REAKTORBETRIEB und DRUCKFÜHRENDE KOMPONENTEN UND WERKSTOFFE

16

G:PPHandout

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
517	11	11.1

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

Authors

JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

11

Instructions to Preparer

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

From Staff Person
JOHN T. LARKINS

November 5, 2004
G:PlanPro(ACRS):ppmins.517

INTERNAL USE ONLY

SUMMARY MINUTES OF THE ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING November 3, 2004

The ACRS Subcommittee on Planning and Procedures held a meeting on November 3, 2004, in Room T-2B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 11:50 a.m. and adjourned at 12:50 p.m.

ATTENDEES
M. Bonaca
G. Wallis
S. Rosen

ACRS Staff
J. T. Larkins
S. Duraiswamy
J. Gallo
M. Snodderly
H. Nourbakhsh
M. Sykes
M. El-Zeftawy
C. Santos
J. Flack
S. Meador
M. Afshar-Tous
M. Weston
R. Caruso

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

Member assignments and priorities for ACRS reports and letters for the November ACRS meeting are attached (pp. 5-9). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the November ACRS meeting be as shown in the attachment (pp. 5-9).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through February 2005 is attached (pp. 5-9). The objectives are to:

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

During this session, the Subcommittee also discussed and developed recommendations on items included in Section IV of the Future Activities list (pp. 10-11).

RECOMMENDATION

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

3) Expanded Meeting of the Planning and Procedures Subcommittee

During its October 2004 ACRS meeting, the Committee decided to hold an expanded meeting of the Planning and Procedures Subcommittee to discuss certain process and regulatory issues. The Committee agreed to hold this meeting at the ACRS Conference Room on January 27-28, 2005. A summary of all the topics proposed by several members are included in the attachment (pp 12-13). A list of topics proposed by the Subcommittee and the member assignments are provided below. Additional assignments for the members and the staff will be made by the ACRS Chairman and the ACRS Executive Director.

Process Issues

- Reviewing fewer issues and spend more time on each (TENTATIVE) (GEA)
- Documenting some ACRS members' concerns regarding the quality of science and engineering that goes into regulations more importantly into the regulatory process (DAP)
- Role of the cognizant Subcommittee Chairman during full Committee meetings (MVB)
- Effectiveness of preparing ACRS reports during full Committee meetings (WJS)

Regulatory Issues

- Extended power uprate issues (MVB/GBW/TSK)
  - Use of containment overpressure credit for NPSH calculations - Impact on defense in depth/safety margins
  - Uncertainty associated with determining the adequacy of the NPSH
  - Should there be a limit on containment overpressure credit to be allowed?
  - Adequacy of the risk methodology to allow a risk-informed decision on the percentage or amount of overpressure credit to be allowed.
  - Generic Implications of extended power uprates
  - Issues proposed by Dr. Kress (pp. 14-15)

Issues raised by Dr. Bonaca and a list of questions and technical issues, associated with power uprates, prepared by the ACRS staff are also attached (pp. 16-20).

- Role of design-basis accident concept in future reactors (DAP)

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the list of topics proposed above as well as on the assignments.

4) Election of Officers for CY 2005

During the December 2004 ACRS meeting the Committee will elect Chairman and Vice Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee. In accordance with the ACRS ByLaws, those members who do not wish to be considered for all or any of the Offices should inform the ACRS Executive Director in writing two weeks (November 19, 2004) prior to the election.

RECOMMENDATION

The Subcommittee recommends that those members who do not wish to be considered for all or any of the Offices (Chairman, Vice Chairman, and Member-at-Large) inform the ACRS Executive Director by November 19, 2004.

5) Christmas Party

Each year, the members sponsor a Christmas party to the ACRS/ACNW Office staff during the December meeting. The members need to decide whether they want to continue with this tradition and sponsor a Christmas party to the ACRS/ACNW Office staff this year. If decided to hold a party, the ACNW/ACNW Executive Director suggests that the party be held at the Greenfields restaurant in Rockville.

RECOMMENDATION

The Subcommittee recommends that the members sponsor a Christmas Party to the ACRS/ACNW Office staff during the December 2004 ACRS meeting. The ACRS staff should provide cost estimates for holding the party in a restaurant, and catering the food and holding it in the ACRS Office (as in the past) for use by the members in deciding where to hold the party.

6) Anonymous Letter Concerning the TRACE Computer Code Development and Review Practices

During the October 2004 ACRS meeting, the Committee considered the anonymous letters sent to Drs. Wallis and Ransom and directed the ACRS Executive Director to send these letters to the EDO for disposition. Accordingly, the ACRS Executive Director sent these letters to the EDO on October 14, 2004. In a memorandum dated October 19, 2004, (pp. 21-27) Mr. Paperiello, RES Director, transmitting the anonymous letter to the Inspector General (IG), states that the issues raised in the letter appear to be technical in nature and he plans to handle them as such. He also states that the letter is being sent to IG for any action he may wish to take.

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director keep the Committee informed of the EDO's and/or the IG's disposition of this issue.

ANTICIPATED WORKLOAD
NOVEMBER 4-6, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	Rosen/Kress	Nourbakhsh/ Duraiswamy	Assessment of the Quality of the NRC Research Projects on Sump Blockage and on MACCS Code	A	To support the staff schedule	**Draft**
Bonaca	—	Savio/Major	Safeguards and Security Matters	—	—	—
		Santos	**Subcommittees Report** — Interim Review of the Farley License Renewal Application — Subcommittee Mtg. November 3, 2004	—	—	—
Kress	—	El-Zeftawy	AP1000 Lessons Learned Report	A	To identify issues stemming from the ACRS review of AP1000	**Draft**
		El-Zeftawy	Status of Early Site Permit Reviews - **INFORMATION BRIEFING**	—	—	—
Rosen	—	Sykes	Proposed Rule on Post-Fire Operator Manual Actions	A	To support the staff schedule	—
Shack	—	Snodderly	Proposed Rule Language for Risk-Informing 10 CFR 50.46	A	To support the staff schedule	—

ANTICIPATED WORKLOAD
NOVEMBER 4-6, 2004 (Cont'd)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Sieber		Weston	Proactive Materials Degradation Assessment Program - **INFORMATION BRIEFING**	—	—	—
		Weston	Significant Operating Events and Grid Reliability	—	—	—
Wallis	Ford	Santos	Response to the August 25, 2004 EDO Response to the May 21, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG 1740, "Voltage-Based Alternative Repair Criteria"	B	To respond to the EDO	**Draft**

ANTICIPATED WORKLOAD
DECEMBER 2-4, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	Shack	Snodderly	Draft Final NUREG Documenting the Expert Elicitation on LBLOCA Frequencies	A	To support staff schedule	—
	—	Snodderly	**Subcommittee Report** - Status of the Development of Draft NUREG on Treatment of Uncertainties - Subc. Mtg 11/16/04	—	—	—
Bonaca	–	Savio/Major	Safeguards and Security Matters	A	To provide the Committee's views	—
		Santos	**Subcommittee Report** - Interim Review of the Arkansas Unit 2 License Renewal Application - Subc. Mtg. 12/1/04	—	— —	—
Denning	–	Nourbakhsh	Proposed Revisions to Management Directive 6.4, Generic Issue Process	Possible Larkinsgram	–	–
Kress	—	El-Zeftawy	Commission Paper on "Regulatory Structure for New Plant Licensing, Part 1: Technology Neutral Framework"	A	To support staff schedule	—

ANTICIPATED WORKLOAD
DECEMBER 2-4, 2004 (Cont'd)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Shack	Apostolakis/ Wallis	Nourbakhsh/Santos	PTS Technical Basis Reevaluation Project	A	To support staff schedule	—
	—	Snodderly	Proposed Rule for Risk-Informing 10 CFR 50.46	A	To support staff schedule	—

ANTICIPATED WORKLOAD
FEBRUARY 10-12, 2005

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	—	Savio/Major	Safeguard and Security Matters	A	To provide the Committee's views	—
		Santos	**Subcommittee Report** - D.C. Cook License Renewal Application - Subc. Mtg. Feb. 9, 2005	—	—	—
Ford	—	Santos	Industry Response to Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections"	Report As needed	To provide the Committee's views	—
Kress	—	El-Zeftawty	Draft Final 10 CFR Part 52 **[TENTATIVE]**	A	To support staff schedule	—
	—	El-Zeftawy	Proposed Rule for AP1000 Design Certification	Possible Larkinsgram	—	—
Powers	Wallis	Caruso/Nourbakhsh	Status Report - Assessment of the Quality of the NRC Research Project on Thermal-Hydraulic Experiments	—	—	—
		Weston	MOX Fuel Fabrication Facility	A	To support staff schedule	—
Sieber	—	Snodderly	Digital I&C Research Plan	B	To provide the Committee's views	—
Wallis	—	Caruso	Waterford Power Uprate	A	To support staff schedule	—
		Caruso	Integrated Chemical Effects Test Results (GSI-191) **[TENTATIVE]**	A	To support staff schedule	—

ACRS Items Requiring Committee Action

1 **Proposed Rule: Fitness For Duty (FFD) Programs, 10 CFR Part 26** (Open)

Member: Mario Bonaca **Engineer:** Med El-Zeftawy

Estimated Time:

Purpose: Determine a Course of Action

Priority: Medium

Requested by: NRR R. Karas

During the 508th meeting of the ACRS, December 3-5, 2003, the Committee considered the proposed rule on drug testing. A Larkinsgram to the EDO was issued on December 10, 2003, stating that the Committee has no objection to the staff's proposal to issue this rule for public comment, and the Committee would like the opportunity to review the draft final rule after reconciliation of public comments.

On May 25, 2004, the Commission directed the staff to combine Part 26 draft proposed rule on drug testing with Part 26 proposed rule on worker fatigue. The additional language that has been added for worker fatigue, pursuant to the combination of the two rule makings, would establish work hour controls for various personnel at nuclear power plants, including operations, maintenance, health physics, chemistry, fire brigade and security . The NRC staff would like the Committee to defer its review of the combined proposed rulemaking after the public comments have been received and analyzed by the staff.

The Planning and Procedures Subcommittee recommends that the Committee review the draft final revision to 10 CFR Part 26 after reconciliation of public comments.

2 **Proposed Rule for AP1000 Design Certification** (Open)

Member: Thomas Kress **Engineer:** Med El-Zeftawy

Estimated Time: 2 hours

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR J. Wilson

The ACRS issued its report on the Safety Aspects of the AP1000 design on July 20, 2004. The NRC staff issued the final safety evaluation report (FSER) and the final design approval (FDA) on September 13, 2004. The current schedule for completing the design certification rulemaking is December 2005. The NRR staff would like the views of the ACRS on the proposed rule for the AP1000 design certification during the February 10-12, 2005 ACRS meeting. The NRR staff expects to provide the proposed rule to the ACRS by December , 2004. The Planning and Procedures Subcommittee recommends that Dr. Kress propose a course of action.

3 <u>DG-1127,"Combining Modal Responses and Spatial Components in Seismic Response Analysis"</u> (Open)

Member: Mario Bonaca **Engineer:** Michael Snodderly

Estimated Time:

Purpose:

Priority: Medium

Requested by: RES T.Y. Chang, RES

DG-1127 will update Revision 1 of RG 1.92 which provides guidance concerning the seismic analysis and design of nuclear power plant structures, systems, components. DG-1127 incorporates improved guidance on the use of Gupta's method for combining modal responses. The NRC staff would like the Committee to defer its review of DG-1127 until after the public comments have beeen received and analyzed by the staff. The Planning and Procedures Subcommittee recommends that the Committee review the draft final version of this Guide after reconciliation of public comments.

11

EXPANDED MEETING OF THE PLANNING AND PROCEDURES SUBCOMMITTEE
JANUARY 27-28, 2005

Summary of Topics Proposed by the Members

Process Issues

- Incoming Chairman's agenda (DAP)
- Alternative approach to review license renewal applications in view of Mr. Leitch's retirement (DAP)
- Should the ACRS take on fewer issues and spend more time on each? (DAP)
- Assessment of the Subcommittee structure and member assignments (DAP)
- Should the ACRS continue to defer voicing and/or documenting its concerns regarding the quality of science and engineering that goes into regulations, more importantly into the regulatory process? (DAP)
- Does the ACRS have a proactive role in nuclear safety issues? If so, how does it originate? (VHR)
- Can the ACRS better influence the NRC safety research (i.e., topics and quality) (VHR)
- Could the report writing be conducted so that a draft is available before the review starts? (This is done sometimes now). Perhaps, the review could begin by going around the table and hearing the main point of view of each member. (VHR)
- What will it take for the ACRS to recommend disapproval of a license renewal application? (TSK)
- Interactions with the NRC staff outside the formal Subcommittee and full Committee meetings (GBW)
- What are the limits to "working with the staff" (suggested by the Commission on certain issues) (GBW):
  - Acting like members of the staff, or its consultants, in developing solutions to the problems
  - Stepping into management gaps and trying to organize the staff and set objectives
  - Planning activities or strategies
  - Doing joint brainstorming or analysis of alternative approach to issues
  - Being peer reviewers
- Should we hold discussions with the staff only at formal meetings or through formal communications, so that any criticism or disagreement is entirely out in the open (GBW)
- Should we workout anything that appears at all potentially disagreeable in private and have more apparent consensus in public? (GBW)
- Agenda for the Quadripartite meeting (should be a focused meeting - perhaps on risk-informed regulation or on divergence) (TSK)

Technical Issues

- Is a design-basis accident a useful concept for future reactors? Should the concept be abandoned in favor of an examination of risk? Would not abandoning the design-basis concept lead to technology neutral regulations? (DAP)
- What should be the role (if any) of design-basis accidents in future regulations? (TSK)
- Why do we think redefining the LBLOCA size is O.K.? (TSK)
- What is the ACRS position on a technology neutral regulatory framework? (TSK):
  - risk acceptance criteria
  - appropriate F-C curves
  - defense in depth - in particular the question of whether a containment is always required
- Any preliminary concerns about design certification of ACR-700 (TSK)
- What should ACRS recommend on sump blockage? (TSK)
- What more should the ACRS do about security issues - particularly with respect to spent fuel? (TSK):
  - pool
  - dry cask storage
  - transportation
  - interim storage facility
- Shouldn't the ACRS do some more complaining about air oxidation? (TSK)
- Power uprates - Is there any limit other than component capacities? Is there not a risk limit? (TSK)
- Organizational factors (Safety Culture) (TSK/SLR)

From: <TSKress@aol.com>
To: <apostola@mit.edu>, <dapower@sandia.gov>, <ransom@ecn.purdue.edu>, <HistoryArt2004@aol.com>, <graham.b.wallis@dartmouth.edu>, <mvbonaca@snet.net>, <JDSIEBER@aol.com>, <wjshack@anl.gov>, <FPCTFord@aol.com>, <denning@battelle.gov>, <jtl@nrc.gov>, <sxd1@nrc.gov>, <hjl@nrc.gov>, <rps1@nrc.gov>
Date: 10/31/04 2:12PM
Subject: Retreat discussion on Vermont Yankee

Gentlemen: Apparently, as requested by the state of Vermont, we are going to be the "independent" reviewers of the Vermont Yankee (VY) power uprate request and I understand this will be the subject of our upcoming retreat. I have given this a little preliminary thought which I would like to share with you as a possible approach for the ACRS review to be discussed at the retreat.

The main contentious issue on the VY power uprate is whether containment overpressure should be credited to maintain NPSH on the sump recirculation pumps. Keep in mind that VY is a BWR with a Mark I containment for which the sump screen blockage issue has supposedly been resolved. In addition, the ACRS has been on record as agreeing to the use of containment overpressure in some specific cases.

I believe this particular review will be highly visible and will be a testament to how fair and independent a review ACRS can give. Therefore, I think we should discount any previous positions of ACRS and resolutions of the sump screen blockage issue and give a fresh new look at the issue. Whatever position we develop should be highly technically defendable. I suggest that there may be two technically defendable approaches:

I. Deal with this issue in "reality space" as opposed to "design basis space".

1, For a complete spectrum of LOCA sizes (a la PRA), require three sets of uncertainty analyses:

A. Uncertainty distribution on the quantity and type of debris that reaches the screen and uncertainty in the consequent head loss versus flow for the various LOCAs.

B. Uncertainty distribution on the calculated containment pressure caused by the various LOCAs.

C. Uncertainty distribution on the required NPSH for the particular pumps at VY.

2. Convoluting A, B, and C will result in a probability of loss of NPSH for any LOCA size.

3. Develop a criterion on what is an acceptable probability for (2) above.

II. The second possible technically defendable approach would be to stay strictly in "design basis space;" Calculate containment overpressure, screen blockage, loss of NPSH, and required NPSH for the DBA LOCAs in a demonstrably conservative way to see if there results an acceptable NPSH for the various DBA LOCAs.

Note, either Approach I or II gives results that can be time dependent.

The staff and the applicant will probably use Approach II. I much prefer Approach I but, for it, we will have to insist on enough data and analyses to make the requisite uncertainy analyses. For the acceptance criterion on the resulting probability, we should view this as a late containment failure and use the LOCA frequencies developed by the expert elicitation and see if the late containment large release frequency meets a new safety goal that we will have to come up with that would be consistent with the current safety goals.

Cheers, Tom Kress

From: "Mario Bonaca" <mvbonaca@snet.net>
To: "John Larkins" <JTL@nrc.gov>, "Sam Duraiswamy" <sxd1@nrc.gov>
Date: 11/1/04 3:39PM
Subject: FW: ACRS PLANNING AND PROCEDURES (P&P)

-----Original Message-----
From: Mario Bonaca [mailto:mvbonaca@snet.net]
Sent: Monday, November 01, 2004 11:06 AM
To: 'Noble Green'
Cc: 'John Larkins'; 'Sam Duraiswamy'
Subject: RE: ACRS PLANNING AND PROCEDURES (P&P)

I have some comments for John and Sam:

Re Item 3, the retreat.

You are proposing that we discuss 4 process issues.

The first is a recurrent subject that I am not sure it is worth to discuss at the retreat. The P&P can always propose to eliminate certain reviews, so that we are more selective in our reviews. I am not sure what we can accomplish at the retreat. Do you think that we can work out a plan on how to reduce and focus our review? And how?

The third will end up with the same agreement that will be routinely ignored the following year. But we can talk about it.

The forth may be a good issue, but I am afraid it is going to be treated just like issue 3.

Re: regulatory issues, I agree with what you are (and Tom is) proposing. However, when I proposed this issue I meant also to discuss its broader implications. The letter from Vermont contends that the Staff (and maybe the ACRS) has essentially effected a policy change without proper explicit justification and evaluation of the supporting bases, Commission approval and public participation and review. I have a sense that this may be true, and it may be happening in other issues where we see a changed NRC attitude due to the industry pressure and to a recent trend of applying engineering judgment with some seat-of-the-pants risk judgment not necessarily supported by rigorous engineering evaluations. So, I propose that we do discuss the specifics of the Vt. power uprate and the approach to use, as proposed by Tom, but also dedicate some time to discuss the broader issue.

Re the Christmas party. The main concern I have about an outside dinner is that it will take too much time out of the office, further squeezing the time we have for letter writing. Furthermore, the traditional party we have offers an opportunity to invite the Commissioners and staff management, and to meet with them informally. But let's talk about it at the P&P.

Mario

-----Original Message-----

16

ACRS Retreat
Issue for consideration
Containment Overpressure Credit

Background

Power uprates have the possibility to alter the relationship between containment and ECCS performance, by making ECCS performance more dependent on successful containment functioning. E.g., some uprates require additional containment overpressure credit to ensure ECCS pump NPSH, and may require operators to maintain containment pressure in a fashion that they have not previously been trained to do.

In addition, the staff position on granting containment overpressure has changed over the past 35 years. In RG 1.1, containment overpressure credit was not allowed. The latest revision to RG 1.82, Rev 3, which was reviewed by the ACRS, states that credit may be granted for plants "...for which the design cannot be practicably altered..." This guidance is not precise, because it does not define "practicably altered".

Over the past 5 years, the staff has granted more and more overpressure credit, especially for power uprates in BWRs, because licensees are reluctant to change their ECCS pumps, and therefore "the design cannot be practicably altered."

The ACRS is on the record as having changed its position on containment overpressure, having stated at one point that

> "allowing some level of containment overpressure is not an acceptable corrective action because adequate overpressure may not be present when needed. In particular, it may not be available during shutdown and containment bypass accidents." (June 17, 1997)

Six months later, the ACRS reconsidered, and concluded:

> "As a result of further review of this issue, we now concur with the NRC staff position that selectively granting credit for small amounts of overpressure for a few cases may be justified. We recommend that instead of using qualitative arguments and restricting attention to a limited range of accident sequences, the decisionmaking process should consider the time variation of NPSH for a broad range of accident sequences such as typically found in a probabilistic risk assessment." And
>
> "The margins in NPSH afforded by the DBA approach constitute a level of defense in depth. Allowing more credit for containment overpressure reduces defense in depth. The staff's justification for this was that the consequences of losing NPSH would not be catastrophic, i.e., the particular pumps at issue would not suffer damage and the discharge flow rates would remain sufficiently high.

We believe that the evidence to support these assertions needs to be identified as a part of the decisionmaking process."(December 1997)

The ACRS did not explicitly consider the portion of the latest revision to RG 1.82, Rev 3, which allowed containment overpressure credit.

In applying for approval of containment overpressure credit, licensees frequently use PRA arguments to demonstrate that risk from the uprate is minor, and that granting overpressure credit is reasonable. Licensees and the staff have presented only limited portions of the PRA to support these positions, and there are some concerns that they have not appropriately modeled all of the consequences of the uprates, including especially new interactions and dependencies between containment performance and ECCS performance, which may not have been modeled or considered before the uprate.

Policy Questions

1. Has the change to the staff position on overpressure credit been properly considered and accepted by the ACRS?

2. When is containment overpressure credit "necessary"?

3. When can a plant not be "practicably altered"?

4. Does granting containment overpressure credit "link" the performance of the containment and the ECCS to create a common-mode failure situation which did not exist before the application for the uprate and the request for overpressure credit?

5. Does this sort of linkage violate the "defense-in-depth" principle?"

6. Should PRAs performed in support of uprates be complete, and include potential new dependencies and interactions created by the uprate?

7. What should be the relationship between "defense-in-depth" and the PRA in supporting the uprate request? Should the definition of "defense-in-depth" be reconsidered or redefined to accommodate new dependencies and interactions created by the uprate?

8. What is the role of maintaining "sufficient" safety margins in the context of a risk-informed power uprate? What are the appropriate metrics for evaluating the safety margins, and how much margin is "sufficient"?

9. When a licensee requests a change such as containment overpressure credit, should it be considered to be such a significant change that its approval needs to reopen consideration of other technical aspects of containment performance, such as seismic design, qualification of the penetrations, and other upgrades to current licensing standards?

10. How much should the plant be upgraded to current licensing standards when it applies for a power uprate? How much "cherry-picking" of new requirements or burden reduction opportunities be allowed?

11. Should there be a limit on the amount of containment overpressure that is granted?

12. Does granting containment overpressure credit, which places operators under some pressure to maintain containment pressure within a varying band for a period of time, violate the TMI Lessons Learned that operators should not be required to perform actions with conflicting goals?

Technical Issues

1. Should uprate plants that request containment overpressure credit be required to assume that containment pressure-retention function fails, as part of their new design basis?

2. Should such plants as described above also be required to assume some other single failure besides containment failure? If so, why? Also, if so, then why not assume containment failure for other sequences?

3. How should the staff determine whether these licensees should be required to change their ECCS pumps to provide the needed NPSH margin?

4. Is there sufficient uncertainty in various aspects of calculating NPSH that the margin provided by containment overpressure should not be surrendered when there is no need to do so, and when the design can be practicably altered to avoid it?

5. In the calculations of success paths for the analyses supporting PRAs related to power uprates, should nominal or bounding input values be used?

6. Is the risk evaluation methodology that is used to support uprate requests sufficiently developed to account for uncertainties?

7. Should there be limits on the amount of overpressure credited? If so, how much, or for how long during the scenario?

8. Should containment overpressure credit be allowed for non-LOCA scenarios, such as SBO or ATWS?

9. What sort of operator training is needed to allow overpressure credit?

10. What sort of containment/penetration testing should be required to ensure that the containment is able to maintain the requested overpressure value for the requested amount of time?

Noble: Put on P&P Agenda Package

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 19, 2004

MEMORANDUM TO: Hubert T. Bell
Inspector General

FROM: Carl J. Paperiello, Director
Office of Nuclear Regulatory Research

Carl J Paperiello

SUBJECT: ANONYMOUS LETTER CONCERNING THE TRACE COMPUTER CODE DEVELOPMENT AND REVIEW PRACTICES

Attached is an anonymous letter concerning allegations relating to the TRACE reactor systems thermo-hydraulic computer code. They appear similar to an earlier anonymous e-mail. With respect to the Office of Nuclear Regulatory Research, the issues appear to be technical in nature and I plan to handle them as such.

This letter is being for forwarded to you for any action you wish to take.

Attachment: October 14, 2004 memo from John T. Larkins to Luis A. Reyes,
Subject as above, w/September 20, 2004 Anonymous Letter

cc: M. Virgilio, DEDMRS
J. Larkins, ACRS

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

October 14, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: ANONYMOUS LETTER CONCERNING THE TRACE COMPUTER CODE DEVELOPMENT AND REVIEW PRACTICES

On September 20, 2004, Dr. Graham Wallis and Dr. Victor Ransom, both members of the Advisory Committee on Reactor Safeguards (ACRS), received an anonymous letter (attached), which describes several issues related to the development, validation, and verification of the TRACE reactor systems analysis code. On March 8, 2004, we provided you with a copy of an anonymous e-mail on the same subject that had been sent to Dr. Wallis. The ACRS intends to consider the comments in the letter as it continues its technical reviews of the development of the TRACE code. We are forwarding the letter to you for your information, and for any action that you see fit to take regarding the development, verification, and technical adequacy of the TRACE code.

Attachment: As stated

cc: ACRS members
C. Paperiello, RES
J. Dyer, NRR
J. Rosenthal, RES
W. Burton, RES

Subject: The TRACE Code

Professor Wallis:

There are several issues that need to be addressed relative to the TRACE code. Generally, these issues are overlooked when NRC codes are under review. They are not however overlooked when the NRC reviews codes submitted by other organizations for review and approval. The issues are discussed following the next paragraph. The next paragraph is an aside that however needs to be addressed by the NRC and the ACRS T/H Subcommittee.

It is a very unfortunate situation in that the extreme adversarial environment that has been present at the NRC for the past thirty years or so makes this form of communication necessary. Free and open discussion of the technical issues that are truly important has not been possible for all these years. The ACRS T/H Subcommittee and its Consultants have been filled with persons who have individual agendas and who do not listen to the very people who know the most about the subject matter that is presented. Those organizations that submit codes for review have hundreds of thousands of very specialized and focused man-hours invested in their products. These people know exactly what is important for each and every application of their codes and experimental data. The ACRS and it consultants, on the other hand, generally do not have the time, or more importantly the inclination, to digest the material presented to the depth necessary to understand the important items that really matter in any application. The personal agendas of the T/H Subcommittee and its Consultants, as reflected in the ACRS transcripts, almost never are important to the practical issues of an application. Quite frankly, the T/H Subcommittee and its Consultants and the material on which they focus and the manner on which they discuss the material are the subject of many jokes and not-so-kind comments all over the industry.

The issues that are being overlooked relative to the TRACE code at this stage in its development include:

1. Independent verification of the coding.

2. A fundamental issue associated with the numerical methods used in the code.

Independent Verification

It is accepted procedure that software in computer codes must be verified before the models and methods are validated. Verification is the process of ensuring that the equations used in the code have been correctly coded. Validation is the process of ensuring that the correct equations have been chosen and coded. Verification must always precede validation, and that is the methodology applied to codes submitted to the NRC by all commercial organizations.

Generally, the computer codes developed under NRC funding have never undergone verification. Additionally, the validation procedure applied to these codes has not measured up to the standards required by the NRC for commercial organizations. Almost all the so-called "validation" or "assessment" calculations done with the NRC codes have not been done under an approved and qualified procedure with "frozen" versions of the software.

I have not seen that verification of the coding in the TRACE code is to be performed. To proceed to validation without verification invalidates the validation process. Additionally, it is not clear that the NRC has a qualified and approved Q/A plan in place for TRACE. Such plans are required of commercial organizations by the NRC.

Issues with the Numerical Methods

If the documentation for the numerical solution methods used in TRACE, both the code manuals and papers in the literature, are studied in detail the results will show that the basic SETS solution method is based on less-than-exact methodologies. Many solution orders for the equations were simply experimented with until one that "works" was discovered. While this approach is less than satisfactory from a theoretical view, it might be called an "engineering solution".

The following discussion is based on the documentation given in the (1) TRAC-P manual of NUREG/CR-5673, LA-12031-M, 1993, and (2) the draft of the manual for the first F-90 version of TRAC/M given in the LA-UR-00-910, 2000. The latter document is basically the former only re-arranged. The latter document is also, I think, a rough first cut at the TRAC part of the TRACE manual. The latter document, (2), will be cited in the following discussion, although the exact same material can be found in the former.

Generally, no multi-step method actually satisfies the original FDEs, and the many approximations used in the TRAC SETS method are somewhat out the the ordinary relative to almost all other numerical solution methods for transient compressible fluid flow. The lack of satisfying the original FDEs and the many approximations can only be appreciated after digging through the manuals in a very, very detailed study. But these are not the main issues here, however.

There is a basic problem that has never been addressed and it is kind of hard to dig out of the documentation. This basic problem is as follows. The numerical method does not solve for the void fraction in a way that can be theoretically justified. A reference to section 2.1.8.2.4 at the bottom of page 2-31 of (2) is the only clue in the manual for how the void fraction at the new-time level is obtained. Section 2.1.8.2.4 is on page 2-61 of (2), 30 pages from where it is needed. The material on page 2-61 states that a system of equations is set up to obtain the new-time level values for the void fraction and the other EOS variables. The system is based on the solution of products of void and density and void-density-energy from the mass and energy stabilizer step of the SETS method plus the EOS with pressure and temperature as independent variables. The discussion given in the manuals is correct and the system of non-linear equations can easily be discovered and the iterative Newton-Raphson method applied to the solution of the system to get the pressure, phase temperatures, and void fraction.

But, here is the basic issue. As discussed on page 2-61, the system of non-linear equations is treated as a system of un-coupled linear equations and a one-shot step, without iteration is all that is done. Note that coupled linear equations require iteration to obtain a solution. Most importantly, all the quantities determined by this one-shot rough estimate are discarded except for the void fraction. This means that none of the results from the one-shot evaluation will satisfy the equations from which they were obtained and only the void fraction from this "solution" is retained. Thus, just as in the case of the RELAP5 code, the non-linear EOS is not satisfied. Additionally one must wonder exactly what the "void fraction" "calculated" in the TRAC SETS manner actually represents.

I continue to investigate the properties of the numerical solution method used in the TRAC-P code, which is the same as in the current version of the TRACE code. The references are (1) TRAC-P manual of NUREG/CR-5673, LA-12031-M, 1993, and (2) the draft of the manual for the first F-90 version of TRAC/M given in the LA-UR-00-910, 2000. The latter document is basically the former only re-arranged. I have so far found out the following.

(1) At the top of page 2-20 of (2) the comments seem to say that the nonlinear equations for the fluid T/H are not solved for the single-phase liquid and single-phase vapor and non-condensable gas cases. That is when a single phase is in the flow system a linearized version of the basic equations is solved. Solution of the non-linear algebraic FDEs by an iterative method is not used; a one-shot through solution method is used instead. I recall that Wolfgang Wulff has written a long article about the use of linearized equations in T/H codes for safety analysis. My recollection is that he thought that this process was not fundamentally sound and that instead the fully non-linear T/H equations should be solved.

(2) I have been thinking some more about the fact that neither RELAP5 or TRAC/TRACE attempt to satisfy the non-linear EOS for the two-fluid model. The solution methods in both these codes are basically slight variations on the MAC-ICE-ALE methods developed at LANL almost three decades ago. While the two-phase equations, especially the non-conservative form of the "momentum" equations, can be made to look a lot like the momentum equation for single-phase flows for which the LANL methods were developed, there is a very significant departure from the single-phase equations. The two-fluid model

momentum equations contain the void fraction, for which there is no analogy in the LANL methods. What both RELAP5 and TRAC/TRACE apparently try to do in order to be able to use "two"- step solution methods, in contrast to fully-implicit methods, is not address completely the pressure-void coupling that is the essence of the two-fluid problem.

I further suspect that both codes have tried to solve the non-linear EOS in developmental versions of the codes but have found that that process leads to unsatisfactory performance of the numerical methods. I suspect this because both codes have apparently come to the same conclusion regarding the non-linear EOS. Other investigations of application of the MAC-ICE-ALE "two"-step methods, in which the non-linear EOS is solved, to the two-fluid model equations have found that the resulting methodology is not unconditionally stable. I do not have the software that could be used to investigate the properties and characteristics of such complex numerical methods as needed for the two-fluid model so that this hypothesis can be tested. A direct question to the developers of the codes might be the most efficient way to get a handle on how the solution methods have come to relay on non-solution of the EOS.

(3) The numerical methods used for both RELAP5 and TRAC/TRACE have been developed with the focus on the CPU time needed to complete a calculation. In REALP5 this has taken the form of solving systems of non-linear algebraic equations without iteration, in addition to not solving the non-linear EOS. In TRAC/TRACE this has taken the form of trying to avoid solution of coupled systems of equations by use of various time-levels during the several steps used in the solution. The number of different time-level values used in the SETS method is sometimes very bewildering. All of the following appear in the equations (2-70) through (2-91) on pages 2-22 through 2-31: old-time values, new-time values, tilde-level values and caret-level values. Plus, the spatial weighting factor for the flux terms, beta, takes on different values as a function of what is calculated to be happening in a given region of the flow field. Beta is a function of tilde- and old-time velocity and velocity-gradient values and phase-change rates and the formulation for beta changes during the various steps in the SETS method. Thus (1) the basic equations cannot be actually satisfied unless all the various time-level quantities attain the same numerical values, including beta; (2) beta can have different values in proximate regions of the flow; and (3) quantities at different time-levels, caret, tilde and new, are taken as the "solution" from the various steps in SETS. It seems to me that there is a high probability that convergence of the numerical equations to the continuous equations cannot be demonstrated.

I continue to investigate some properties of the TRAC-P code, which is the same as in the current version of the TRACE code. The references are (1) TRAC-P manual of NUREG/CR-5673, LA-12031-M, 1993, and (2) the draft of the manual for the first F-90 version of TRAC/M given in the LA-UR-00-910, 2000. The latter document is basically the former only re-arranged.

The authors of the TRAC codes have chosen to use the nomenclature "motion equation" for what is generally considered to be a momentum equation model. My experience is that all these systems-analysis codes generally return solutions consistent with the mechanical energy, or Bernoulli, equation.

There is an option in the TRAC codes to apply a COSINE factor to the momentum-flux term in the motion equation. The factor can be applied to the flux term at either, or both, contributions which appear in each cell edge (or link, flowpath, junction) motion equation. The factor is designed to be applied at locations at which the flowpath might split, or merge, such as at TEEs, and Ys.

This seems to be a problem as follows. If the motion equation returns the correct solution, ie returns the Bernoulli results, for one value of the COSINE factor, like the case for flows in straight pipes, it cannot return the correct solution for another value. Unless, of course, the motion equation is modified by some term. I do not see that the equation is modified in the manuals.

I suspect that if the COSINE factor is set to make the flux contribution null, pressure peaks will be present in the solution. This is the usual situation of stopping a flow so that the flux gives rise to a pressure increase.