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

July 10, 2002

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: SUMMARY REPORT - 493rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, JUNE 6-8, 2002 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 493rd meeting, June 6-8, 2002, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports and letters.

REPORTS:

The following reports were issued to Chairman Meserve, NRC, from George E. Apostolakis, Chairman, ACRS:

- Proposed Revision to 10 CFR 50.48 Endorsing NFPA-805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" dated June 17, 2002
- Policy Issues Related to Advanced Reactor Licensing dated June 17, 2002
- Vessel Head Penetrations and Vessel Head Degradation dated June 20, 2002

LETTERS:

The following letters were issued to William D. Travers, Executive Director for Operations (EDO), NRC, from George E. Apostolakis, Chairman, ACRS:

- Proposed Technical Assessment of Generic Safety Issue-168, "Environmental Qualification of Low-voltage Instrumentation and Control Cables" dated June 17, 2002

- Recommendations Proposed by the Office of Nuclear Regulatory Research for Resolving Generic Safety Issue-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident" dated June 17, 2002

HIGHLIGHTS OF KEY ISSUES

1. CRDM Cracking of Vessel Head Penetrations and Vessel Head Degradation

The Committee heard presentations by and held discussions with representatives of the Electric Power Research Institute/Materials Reliability Program (EPRI/MRP), First Energy Nuclear Operating Company (FENOC), and the NRC staff regarding vessel head penetration cracking and degradation and the data relevant to these issues.

FENOC presented information on the Davis-Besse reactor pressure vessel (RPV) closure head activities including replacement considerations, examinations of the Midland head which is to replace the current degraded head, plans for a new head purchase, and additional information on their root cause analysis. The MRP provided data on the crack growth rate curve and its application, probabilistic fracture mechanics analysis, and the industry inspection plan. The staff presented information on its assessment of margin available at Davis-Besse, the priority ranking of plants responding to Bulletin 2002-01, the implementation goals of the Inspection Manual Chapter 0350 Panel, the objectives and scope of the Lessons Learned Task Force, the inspection methods and frequencies being considered for management of the cracking and degradation issues, and comments on the industry's inspection plan.

The staff and industry are working together to resolve differences related to the inspection plan and other cracking and degradation issues.

Committee Action

The Committee issued a report to Chairman Meserve on this matter dated June 20, 2002, concluding that understanding of the issues related to cracking and degradation are progressing, but additional work is still needed. The Committee plans to hold additional meetings with the staff on these issues.

2. Technical Assessment of Generic Safety Issue (GSI)-189, Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident

The Committee heard presentations by and held discussions with representatives of the NRC staff and their consultants, the Union of Concerned Scientists (UCS), and members of the public regarding the recommendations proposed by the Office of Nuclear Regulatory Research (RES) to resolve Generic Safety Issue (GSI)-189. The Committee members, the staff, a representative of UCS, and members of the public discussed the background for the GSI, the options evaluated by RES for resolving the issue, RES cost benefit analysis and associated bases, and the uncertainties associated with the benefit cost analysis.

Committee Action

The Committee issued a letter to the EDO on this matter dated June 17, 2002, recommending that RES complete its additional analyses to quantify the uncertainties prior to providing the technical assessment results to the Office of Nuclear Reactor Regulation (NRR), and that NRR factor the uncertainties into the final resolution of GSI-189.

3. Technical Assessment of GSI-168, Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the technical assessment of GSI-168. The Committee discussed RES's conclusion that the current environmental qualification process for I&C cables is adequate for the duration of the current license term of 40 years. The Committee also discussed RES's proposed recommendations to NRR to resolve GSI-168.

Committee Action

The Committee issued a letter to the EDO on this matter dated June 17, 2002. The Committee recommended that cable test results and other pertinent information be disseminated to the nuclear industry through a generic communication, and that the generic communication also include a discussion of the treatment of I&C cables during the license renewal term.

4. Development of Reliability/Availability Performance Indicators and Industry Trends

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the agency's pilot program to risk-inform the performance indicator (PI) process associated with the revised Reactor Oversight Process (ROP). The pilot program is aimed at monitoring the unavailability and unreliability of selected mitigating safety systems and attempts to revamp results into one that produces a risk-informed PI; i.e. Pilot-MSPI (Mitigating System Performance Indicator). This effort is part of the Mitigating Systems Cornerstone.

Committee Action

This was an information briefing. No Committee action was taken at this meeting. The Committee will keep informed of staff activities with a follow-up joint meeting of the ACRS Subcommittees on Plant Operations and on Reliability and Probabilistic Risk Assessment in the fall of 2002.

5. Technical and Policy Issues Related to Advanced Reactors

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding issues that have potential policy implications for advanced reactor licensing and the staff's plans for seeking Commission's guidance for resolving these issues. The staff identified five policy issues: event selection and safety classification; fuel performance and qualification; source term; containment versus confinement; and emergency evacuation. In addition, the staff identified two overarching issues: the implementation of the Commission's "expectation" that advanced reactors will provide enhanced safety margins; and the relationship between the NRC's safety requirements and international safety requirements.

Committee Action

The Committee issued a report to Chairman Meserve on this matter dated June 17, 2002, recommending that the existing agency positions on some of these policy issues be reevaluated because of new perspectives on risk-informed regulation and defense in depth, as well as anticipated new reactor designs. In addition, the need for greater specificity in the application of defense in depth should be made a separate overarching issue.

6. Proposed Rulemaking to Endorse National Fire Protection Association 805 Standard, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants"

The Committee heard presentations by and held discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding the proposed revision to 10 CFR 50.48, endorsing National Fire Protection Association (NFPA) Standard 805 as a risk-informed and performance-based standard for determining fire protection requirements in commercial nuclear power plants. The Committee members, the NRC staff, and NEI representatives discussed the proposed rule, the NFPA 805 Standard, the plan for implementation of the rule, the development of implementation guidance by NEI, and the resolution of concerns identified by NEI.

Committee Action

The Committee issued a letter to the EDO on this matter dated June 17, 2002, recommending that the proposed rule be issued for public comment.

7. Generic Resolution of Voids in the Concrete Containment Walls

The issue of voids in the concrete containment walls was raised by a member of the public during the Committee's review of the Turkey Point license renewal application. The Committee discussed this issue during its April 2002 meeting and concluded that it has been properly addressed for Turkey Point. However, the Committee requested that the staff address the generic resolution of voids in the concrete containment walls. During the June 2002 ACRS meeting, the staff provided details of how voids would normally be detected during construction and testing of the containment. In addition, the staff provided information in support of the staff's conclusion that potential undetected voids in concrete structures do not have any generic safety implications.

Committee Action

The Committee found the staff's answer to be responsive to its concern. No further concerns were identified.

8. Format and Content of the 2003 ACRS Report on the NRC Safety Research Program

The Committee discussed the format, content, schedule, and member assignments for the 2003 ACRS report to the Commission on the NRC Safety Research Program.

Committee Action

The Committee will continue its efforts regarding the 2003 ACRS report on the NRC Safety Research Program during future meetings.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee discussed the EDO's response of May 13, 2002 to comments and recommendations included in the ACRS March 19, 2002 report related to Risk-Informing Special Treatment Requirements of 10 CFR Part 50 (Option 2).

  Several Committee members continue to disagree with the staff on the value of uncertainty analysis to support development of importance measures and changes in core damage frequency (ΔCDF).

  The Committee decided to continue its discussion of this matter during the July 2002 meeting.

- The Committee considered the response from the EDO, dated May 6, 2002, to comments and recommendations included in the ACRS report dated March 14, 2002, concerning Arkansas Nuclear One, Unit 2 Extended Power Uprate.

  The Committee decided to continue its discussion of this matter during the July 2002 meeting.

- The Committee considered the response from the EDO dated May 31, 2002, to comments and recommendations included in the ACRS report dated April 19, 2002, concerning a report on the safety aspects of the license renewal application for the Turkey Point Nuclear Plant, Units 3 & 4.

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

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from May 2, 2002 through May 31, 2002, the following Subcommittee meetings were held:

- Reliability and Probabilistic Risk Assessment and Plant Operations - May 30, 2002

The Subcommittees reviewed the staff's initiatives to integrate the NRC program for risk-based analysis of reactor operating experience into the reactor oversight process, specifically the development of reliability/availability performance indicators and industry trends.

- Planning and Procedures - May 30, 2002

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

- Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and Probabilistic Risk Assessment - May 31, 2002

The Subcommittees discussed the status of staff efforts at risk-informing 10 CFR 50.46 and related changes to Appendix K and Generic Design Criterion 35 (GDC-35) for emergency core cooling systems (ECCS). The discussion centered on the four principal technical areas related to risk-informing ECCS requirements; namely a) ECCS reliability requirements, b) ECCS acceptance criteria, c) ECCS evaluation model requirements, and d) the spectrum of break sizes to be considered.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee plans to review the proposed resolution of GSI-168, "Environmental Qualification of Low-Voltage Instrumentation and Control Cables."

- The Committee plans to review the additional analyses and the proposed RES recommendations to NRR for resolving GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."

- The Committee plans to continue its discussion of the May 13, 2002 ACRS report related to Risk-Informing Special Treatment Requirements of 10 CFR Part 50 (Option 2) during the July 10-12, 2002 ACRS meeting.

- The Committee plans to continue its discussion of the May 6, 2002 EDO response to the March 14, 2002 ACRS report concerning Arkansas Nuclear One, Unit 2 Extended Power Uprate during the July 10-12, 2002 ACRS meeting.

- The Committee plans to continue its discussion of issues associated with CRDM cracking of vessel head penetrations and vessel head degradation during future meetings.

- The Committee plans to continue its discussion of the staff's progress in developing reliability/availability performance indicators during future meetings.

- The Committee plans to continue its interaction with the staff on possible approaches and options for resolving policy issues related to advanced reactor licensing.

PROPOSED SCHEDULE FOR THE 494th ACRS MEETING

The Committee, in addition to meeting with the NRC Commissioners, agreed to consider the following topics during the 494th ACRS meeting, July 10-12, 2002:

- Pressurized Thermal Shock (PTS) Reevaluation Project: Risk Acceptance Criteria

- Draft Final Revision 1 to Regulatory Guide 1.174, "An Approach to Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and Associated Standard Review Plan Chapter 19

- Risk-Informed Regulation Implementation Plan

- Advanced Reactors Research Plan

- Overview of NRC Research Activities in the Seismic Area

- Development of Review Standard for Reviewing Core Power Uprate Applications

- Application of the Probabilistic Fracture Mechanics Methodologies to Reactor Vessel Integrity Assessment
- Format and Content of the 2003 ACRS Report on the NRC Safety Research Program

Sincerely,

George E. Apostolakis
Chairman

CERTIFIED

Date Issued: 8/12/2002
Date Certified: 8/21/2002

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MINUTES OF THE 493rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
June 6-8, 2002
ROCKVILLE, MARYLAND

The 493rd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on June 6-8, 2002. Notice of this meeting was published in the *Federal Register* on May 16, 2002 (65 FR 34961) (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 Public Document Room at the One White Flint North Building, Mail Stop 1F-15, Rockville, MD, 20852-2738. [Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc., 1323 Rhode Island Avenue, NW, Washington, DC 20005-3701, and on the ACRS/ACNW Web page at (www.NRC.gov/ACRS/ACNW).]

ATTENDEES

ACRS Members: ACRS Members: Dr. George Apostolakis (Chairman), Dr. Mario V. Bonaca (Vice Chairman), Dr. F. Peter Ford, Dr. Thomas S. Kress, Mr. Graham M. Leitch, Dr. Dana A. Powers, Dr. Victor H. Ransom, Mr. Stephen L. Rosen, Dr. William J. Shack, Mr. John D. Sieber, and Dr. Graham B. Wallis. 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. George E. Apostolakis, 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. CRDM Cracking of Vessel Head Penetrations and Vessel Head Degradation (Open)

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

Dr. F. Peter Ford, Chairman of the Materials and Metallurgy Subcommittee, introduced this topic to the Committee. Mr. John D. Sieber, Chairman of the Plant Operations Subcommittee, co-chaired this effort. The Committee heard presentations by representatives of EPRI/MRP, First Energy Nuclear Operating Company for Davis-Besse, and the NRC staff. The purpose of this meeting was to hear information regarding vessel head penetration cracking and degradation, and the data relevant to these issues.

NRC Staff and Industry Presentations

Staff presentations were made by Mr. Bill Bateman, NRR, Mr. John Grobe, Region III, and Mr. Edwin Hackett, Office of Nuclear Regulatory Research (RES). The industry presentations were made by Mr. Larry Mathews, Electric Power Research Institute Materials Reliability Program EPRI/MRP, and Mr. Jim Powers, Mr. Mark McLaughlin, Mr. Bob Schrauder, and Mr. Steve Loehlein and Mr. Ken Byrd, First Energy Nuclear Operating Company (FENOC).

During previous presentations to the ACRS on vessel head penetration cracking and degradation, information was provided regarding the status of Bulletins 2001-01 and 2002-01, the Augmented Inspection Team (AIT) Report, the findings and root cause of the corrosion at Davis-Besse, and proposed approaches to managing these issues. At those presentations, the ACRS requested that in a subsequent meeting, data be provided to support some of the comments made and conclusions drawn. Some of that data was presented at this meeting.

FENOC presented information on their reactor pressure vessel (RPV) closure head activities including replacement considerations, specifics regarding the Midland head which they plan to use temporarily, visual radiography (RT), liquid penetrant (PT), ultrasonic (UT), magnetic particle (MT) and additional non-destructive examinations (NDE) of the Midland head, plans for a new head purchase, and additional information on their root cause analysis.

The materials reliability program provided data on the derivation of the crack growth rate curve, the application of the curve, probabilistic fracture mechanics analysis of the CRDM nozzles, the industry's technical assessment of the Davis-Besse degradation, and the industry inspection plan.

The staff presented information on their assessment of margin available at Davis-Besse, the priority ranking of plants responding to Bulletin 2002–01, implementation goals of the Inspection Manual Chapter 0350 Panel, the objective and scope of the Lessons Learned Task Force, a discussion of the inspection methods and frequencies being considered for management of these issues, and comments on the industry's inspection plan.

The staff and industry are working together to resolve differences related to the inspection plan and other cracking and degradation issues.

Committee Action

The Committee concluded that issues related to cracking and degradation are progressing, but additional work is needed. They expect additional meetings with the staff on these issues. The Committee issued a report to Chairman Meserve on this matter dated June 20, 2002.

III. Technical Assessment Generic Safety Issue (GSI)-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident" (Open)

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

Dr. Kress, the cognizant member for this issue, introduced the topic. The Committee heard presentations by with representatives from the NRC staff and their consultants, the Union of Concerned Scientists (UCS), and a member of the public, regarding RES recommendations to resolve GSI-189. The Committee members, the staff, and representatives of UCS and the public discussed the background for the GSI, the options evaluated by RES for resolving the issue, RES's cost benefit analysis and associated bases, and the uncertainties associated with RES's analysis.

The Committee was reminded that this issue deals with hydrogen concentrations in ice condenser and Mark III containments during a severe accident. In order to control

hydrogen concentrations to prevent detonable concentrations, these types of plants are provided with hydrogen igniters located throughout the containment area outside the ice condenser chamber and in the drywell for Mark IIIs. The ice condenser containments also include fans to prevent stratification and ensure that the hydrogen can get to the igniters. One of the severe accident sequences that contribute significantly to risk is station blackout (SBO). The igniters and the fans are powered by AC power, and during SBO, they would lose their source of power. So the basic issue is, should igniters and fans (fans applicable to ice condenser plants only) in ice condenser and Mark III containments be equipped with backup power in the event of an SBO event. If so, this would constitute a backfit, and the staff would be required to perform a backfit regulatory analysis. Dr. Kress stated that RES has performed the backfit analysis and will be appearing before the Committee to discuss their findings.

He also noted that the Committee had received requests from two members of the public to address the Committee. David Lochbaum (representing Ken Bergeron, both with UCS) and Ms. Ann Harris, a member of the public living near the Watts Bar nuclear plant. Dr. Kress then turn the meeting over to Mr. Al Notafrancesco, RES, to begin the presentation.

Mr. Notafrancesco stated that he is the RES Task Manager for GSI-189. The major points of Mr. Notafrancesco's presentation are as follows:

- As part of the process for risk informing 10 CFR 50.44, the Commission issued a Staff Requirements Memorandum (SRM) on December 31, 2001 requiring the staff to resolve GSI-189 expeditiously. In February 2002, RES completed its generic issue screening process for GSI-189, and created a task action plan for resolution. RES is currently completing its technical assessment of the issue, and the results of the technical assessment is the subject for this presentation.

- The population of plants affected by GSI-189 includes nine PWRs with ice condenser containments (four dual unit sites and one single unit site), and four BWR plants with Mark III containments (four single unit sites). In the 1980s, these plans were retrofitted with AC-powered igniters to mitigate the consequences of hydrogen accumulation as part of the post Three Mile Island action plan.

- RES's approach for resolution was to use existing studies to perform the cost analysis, the benefits analysis, and the plant analysis. Options evaluated include a pre-staged diesel generator design, and a commercial off-the-shelf portable

generator that could be placed in service with minimal plant modifications. For the purposes of the analysis, backup power was assumed to be needed for 48 hours.

- The cost benefit analyses show that the low cost equipment option for ice condensers is clearly beneficial, and provides flexibility for response to external events. For Mark IIIs, the low cost option is marginally beneficial. RES, therefore, concludes that further regulatory action is warranted.

- A MELCOR study was utilized to demonstrate that igniters alone are effective in controlling hydrogen build-up in the ice condenser plants. In addition, the MELCOR study shows that use of one air return fan would provide marginal improvement in controlling hydrogen, but would accelerate ice melt. The uncertainty portion of the study is still ongoing.

- RES believes that igniters alone are adequate to protect the containment because there are more than 30 igniters distributed throughout the containment

In response to questions from the Committee, the staff provided the following additional information:

- In NUREG-1150, the SBO was approximately 90 percent of the total core damage frequency. For the ice condensers, there was some variation, but SBO is still a significant part of the total core damage frequency. For Catawba, the conditional containment failure probability was about 0.3. The large early release frequency was somewhat lower than the conditional containment failure probability, but still a significant contributor to risk. External events contribute about half of the risk associated with an SBO event.

- RES did not evaluate hooking up the igniters and/or fans to the emergency diesel generators because they were assumed to not be working during an SBO event.

- The scenario studied is as follows: at time T_0, a hurricane, earthquake, or other external event hits the plant. A common mode failure of the diesel generators is assumed. The backup source of the power to the igniters would be diverse; therefore, not subject to that common mode failure for the emergency diesel generators. Given an SBO event, the plant will cope with the event for several hours by either living off the station batteries, or by supplying water to your

steam generators with the steam driven auxiliary feedwater pumps. At some point in the event, the core will start to uncover, heating the core, and generating hydrogen. The igniters will need to be continuously powered, so that they can burn off the hydrogen in small amounts over the period of hours during which hydrogen is being generated. For the purposes of the study, it was assumed that the igniters would need to have power for 48 hours. Initially, the station crew will likely be focused on attempting to restore main power. If they are unable to restore main power, at some point, they would opt to connect the backup power source.

- The low cost option (portable generator) was assumed to a functional reliability of 0.8. The higher priced fixed generator option was assumed to have a functional reliability of 0.9. These reliability numbers are a combination of unreliability, unavailability, and human reliability. The human reliability drives both numbers. No uncertainty analysis was performed.

- Providing backup power to the igniters would not eliminate potential early containment failures, but would virtually eliminate early containment failures due to hydrogen combustion. There are other potential mechanisms that could still cause early containment failure, such as direct containment heating.

- Use of air return fans with the low cost portable diesel generator option was not considered because the power requirements are too large.

- There has been a substantial amount of work over the last 10 years looking at criteria for the transition to detonation. Specifically, the work has focused on criteria for determining an intrinsic measure of the detonability of a mixture. Work performed by the Russian Academy of Sciences and others has led to the development of correlations for the ratio of characteristic dimensions to cell size, which give an indication of the likelihood that a mixture can undergo a detonation. The state of the art for assessing detonability of mixtures is improved, and for certain kinds of geometries, the staff believes that these rough measures can give a picture of the detonability.

- Inerting the containment was not considered as part of the current GSI-189 study, but was considered after Three Mile Island. It was considered to be detrimental to safety for ice condensers and Mark III containments because the operations and maintenance activities associated with these plant types require much more frequent personnel access. Inerting would limit this access.

In response to the staff's presentation, the Committee expressed concerns that for the purposes of the study, the staff selected a solution and tried to justify it technically, because it's the only one that was affordable. Specifically, the issue of return air fans concerned the Committee. The Committee noted that the staff's judgment is made based on the absence of a detonation in the ice chamber. If the fans could prevent a detonation in the ice chamber, then a different cost-benefit ratio would likely result. The Committee also expressed concerns about the ability of MELCOR to adequately predict hydrogen concentrations throughout the containment, the ability to predict deflagration and detonation in an ice bed full or water droplets, and the uncertainties associated with the staff's calculations.

David Lochbaum, of the Union of Concerned Scientists, was introduced next to discuss some comments forwarded to the ACRS by Ken Bergeron. Mr. Bergeron was unable to attend the meeting, and Mr. Lochbaum was speaking on his Mr. Bergeron's behalf. Mr. Bergeron had forwarded a letter with his comments to the ACRS on June 2, 2002. Mr. Lochbaum noted that Mr. Bergeron's concerns had been discussed during the meeting. Mr. Lochbaum noted an inconsistency in the staff's presentation. Specifically, for non-station blackout events, the staff assumed that the igniters and the air return fans are functional. For station blackout events, they performed a MELCOR study to demonstrate that igniters alone are effective in controlling hydrogen. He noted that the staff's conclusion would lead one to believe that for non-station blackout events that you don't need air return fans either. He stated that he believes that the low-cost option proposed by the staff is insufficiently justified, and may actually be setting the operators up for a worse accident than the one they are dealing with. Unless a stronger justification is made for not including the air return fans in the station blackout provisions, Mr. Bergeron and Mr. Lochbaum would oppose providing backup power to the igniters only. He also noted a security concern that the 55-gallon drums of diesel fuel and diesel generators on wheels may make it easier for saboteurs to attack a plant without bringing their own explosives.

Ms. Ann Harris was then recognized by the chair to make a statement. The key points of her statement is that she expressed concern about the safety of the ice condenser plants and about the ability of the hydrogen igniters to protect against containment failure due to a hydrogen detonation. She challenged whether the portable generator option included in the RES technical assessment provided an adequate level of safety. She urged the Committee and the staff to take appropriate actions to fix the problems.

Mr. Bob Bryan of the Tennessee Valley Authority was introduced next to provide some comments from the licensee's perspective. Specifically, Mr. Bryan noted that the cost-benefit study was estimated based on power requirements that were lower than the power requirements for igniters at Watts Bar and Sequoyah. He stated that Watts Bar and Sequoyah igniters require about 600 watts each. He indicated that this comment was from a preliminary look at RES's findings, and that the licensee was only beginning to perform a more detailed evaluation of the GSI-189 results.

Mr. Paul Gunter, Nuclear Information Research Service, made a comment that the RES analysis of GSI-189 did not account for potential common mode failure caused by contaminated fuel.

Committee Action

The Committee issued a letter to the Executive Director of Operations on this matter dated June 17, 2002, recommending RES complete its additional analyses to quantify the uncertainties prior to providing the technical assessment results to the Office of Nuclear Reactor Regulation (NRR), and that NRR should factor the uncertainties into the final resolution of GSI-189.

IV. Technical Assessment of GSI-168, Environmental Qualification of Low-Voltage Instrumentation and Control Cables (Open)

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

Mr. Leitch, the cognizant ACRS Member for this issue, introduced the topic to the Committee noting that the environmental qualification (EQ) of instrumentation and control (I&C) cables is very important to plant operations since, if the cables fail, misleading and confusing information may be transmitted to the plant operators. Mr. Leitch also provided "as tested" cable samples to the Committee for examination.

The Committee heard presentations by representatives of the NRC staff regarding the technical assessment of GSI-168 "Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables." The technical assessment was prepared by RES. The Committee also discussed RES's proposed recommendations to the NRR to resolve GSI-168.

The staff began the briefing by presenting an overview of the history of EQ in I&C cables. The staff noted that it had performed research to resolve issues related to the EQ process for low-voltage I&C cables. Initially, a list of 43 issues were identified. Twenty four of the issues were resolved by a review and analysis of past research and 19 issues remained unresolved. Of the latter, 6 issues required additional cable testing to resolve.

The staff concluded that the current EQ process for low-voltage I&C cables is adequate for the duration of the current license term of 40 years. Therefore, the overall EQ process provides reasonable assurance that, when qualified in accordance with the regulations, the cables will perform their intended function during their qualified life. The staff also noted that the benefits of a full regulatory program compared to voluntary industry action were very small.

The Committee agreed with the staff's conclusions that the current EQ process for low-voltage I&C cables is adequate for the duration of the current license term of 40 years. The conservatism in the operating environment, as compared to the qualification environment, coupled with observation of the condition of the cables could be used to extend the qualified life of the cables. The Committee also agreed that monitoring techniques should continue to be developed by the industry and that a combination of condition monitoring techniques should be used since no single technique is effective to detect degradation of I&C cables.

During the above discussions the ACRS Members noted the following issues:

- Mr. Leitch and Mr. Rosen were concerned that, based on recent information, the Arrhenius model for thermal aging may not be conservative as stated in the staff's technical assessment. Mr. Rosen noted that the aging is performed quickly and does not allow for the diffusion of oxygen into the cable insulation. The staff stated that there are some uncertainties with the Arrhenius model with some materials; however, the class of materials tested during this research give reasonably conservative predictions of aging which have been confirmed by other studies.

- Mr. Rosen expressed some concern that the test cables may not be representative of all cables in service at nuclear plants. The staff responded that three types of cables were tested which were representative of between 75 and 77 percent of the I&C cables in nuclear power plants.

- Dr. Powers questioned whether the research took into account other plant conditions such as dirt, oil, and other contaminants adhering to the cables. The staff stated that these types of plant conditions were not included in the research program.

- Dr. Powers expressed concern that mechanical influences such as vibrations and cable movements resulting from main steam line breaks were not included in the research and recommended that the staff review these types of issues.

Committee Action

The Committee recommended that cable test results and other pertinent information be disseminated to the nuclear industry through a generic communication, which should also include a discussion of the treatment of I&C cables during the license renewal term. The Committee plans to review the proposed resolution of GSI-168. The Committee issued a letter to the Executive Director for Operations on this matter dated June 17, 2002.

V. Development of Reliability/Availability Performance Indicators and Industry Trends (Open)

[Note: Mrs. Maggalean W. Weston was the Designated Federal Official for this portion of the meeting. Dr. August W. Cronenberg was the Cognizant Staff Engineer]

The Committee heard presentations by representatives of the NRC staff regarding the agency's pilot program to risk-inform the performance indicator (PI) process associated with the revised Reactor Oversight Process (ROP). The pilot program is aimed at monitoring the unavailability and unreliability of selected mitigating safety systems and attempts to revamp results into one that produces a risk-informed PI; i.e., Pilot-Mitigating System Performance Indicator. This effort is part of the Mitigating Systems Cornerstone of the ROP.

Committee Action

This was an information briefing. The Committee will be kept informed of staff activities and a follow-up joint meeting of the ACRS Subcommittees on Reliability & PRA and Plant Operations will be scheduled in the fall of 2002. No Committee action was taken at this meeting.

VI. Technical and Policy Issues Related to Advanced Reactors (Open)

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

Dr. Kress, Chairman of the Subcommittee, stated that the Committee would hear presentations by the NRC staff regarding technical issues that have potential policy implications for advanced reactor licensing.

Dr. F. Eltawila, RES, stated that to facilitate licensing of advanced reactors, the Commission has encouraged the staff to interact with the designers at the pre-application stage to identify, early in the process, key safety and licensing issues. The Nuclear Energy Institute has submitted a white paper to the NRC Chairman on this topic.

The industry's approach to safety for future plants emphasizes prevention of radionuclide release and proposes to use PRA in the design process to a greater extent than current generation LWRs. The staff identified five areas:

- Event selection and safety classification
- Fuel performance and qualification
- Source term
- Containment vs. Confinement
- Emergency evacuation

Related to each of the above five areas, the staff considers two overarching policy issues. They are:

- How to implement the Commission's expectation that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions?

- What should be the relationship of NRC safety requirements to international safety requirements?

The staff plans to engage stakeholders on the above five issues prior to providing recommendations to the Commission. Currently, the staff plans to provide recommendations to the Commission in the Fall 2002.

Committee Action

The Committee issued a report to Chairman Meserve on this matter, dated June 17, 2002, recommending that the existing agency positions on some of these policy issues should be reevaluated because of a new perspective on risk-informed regulation and defense in depth, as well as new reactor designs. In addition, the need for greater specificity in the application of defense in depth should be made a separate overarching issue.

VII. Proposed Rulemaking to Endorse National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Station (Open)

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

During this session, the Committee heard presentations by representatives from the NRC staff and the Nuclear Energy Institute (NEI) regarding the proposed revision to 10 CFR 50.48, endorsing National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," as a risk-informed performance-based standard for determining fire protection requirements in commercial nuclear power plants. The Committee members, the staff, and NEI representatives discussed the proposed rule, the NFPA 805 Standard, the plan for implementation of the rule, the development of implementation guidance by NEI, and the resolution of concerns identified by NEI.

Mr. Rosen, Chairman of the Subcommittee on Fire Protection, provided introductory remarks for this session and summarized the Subcommittee meeting on June 4, 2002. He reminded the Committee that they had previously been very critical of the NFPA 805 standard because the standard did not go far enough in risk-informing fire protection requirements. He indicated that he believed that the NRC staff had been responsive to the Committee's comments in the development of the current proposed rule.

The staff then made its presentation on the proposed rule. Mr. John Hannon, NRR, provided introductory remarks for the staff. Mr. Eric Weiss, NRR, gave the staff's presentation. The key points of Mr. Weiss's presentation were:

- Background: The existing fire protection regulations contain prescriptive requirements. The ACRS was critical of NFPA 805 during its development

because it did not allow risk assessments to be used to alter basic fire protection requirements. The staff considered the Committee's comments in the development of the proposed rule. Regulatory Guide (RG) 1.189, "Fire Protection for Operating Nuclear Power Plants," was recently issued to lay out criteria for an adequate fire protection program. NFPA 805 was developed as a risk-informed national consensus standard, and was issued in February of 2001. The proposed revision to 10CFR50.48 endorsing NFPA 805 is consistent with the National Technology Transfer and Advancement Act and OMB Circular A119 which require federal agencies to use national consensus standards in lieu of agency developed or specific criteria when they serve the needs of the agency. NFPA 805 takes advantage of the advances in probabilistic risk assessment (PRA) and fire science since Appendix R was issued 20 years ago. Advances in fire modeling and PRA since the issuance of Appendix R have been substantial and the proposed rule permits fire protection to be risk-informed and performed-based, without requiring exemptions from licensees.

- Advantages of the proposed rule: The NFPA 805 standard allows licensees to maintain safety through more flexible, efficient and rational processes. A net result of this is that the staff believes the rulemaking will reduce the number of exemptions and license amendment requests.

- NFPA 805 structure: NFPA 805 allows the transition of the existing Appendix R licensing basis, including the exemptions and Generic Letter 86-10 equivalencies. Future changes to the plant may be either deterministic or risk-informed. However, if changes are made (either deterministic or risk-informed), the licensee must still evaluate the risk associated with the change. Acceptable changes in core damage frequency (CDF) or large early release frequency (LERF) are consistent with the guidelines of Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." If the licensee meets the risk CDF or LERF criteria associated with a given change, then NRC approval of that change is not required. The standard also includes a change control process. New performance-based risk-informed requirements may be utilized for meeting the standard once they have been approved by the NRC. Chapter 3 of NFPA 805 still retains a list of fundamental fire protection requirements that are prescriptive. The staff could allow new risk-informed performance-based methods to replace these requirements in the future.

- Rule structure: The rule is structured as an amendment to 10 CFR 50.48. The adoption of the NFPA 805 standard is voluntary. Licensees may elect to keep their existing Appendix R licensing basis. The existing licensing basis configuration and procedures would convey to the new risk-informed performed-based environment. Licensees would document their licensing basis changes and retain the records on site for inspection. The reactor oversight process would monitor future changes. The NRC would review/approve any new risk-informed performance-based methods used by licensees or a licensee may opt to use the methodologies described in the appendices to NFPA 805.

- Summary/schedule: NEI will develop a guidance document for implementation of the rule which the staff will review, and if found acceptable, endorse in a regulatory guide. NFPA 805 addresses the existing fleet of light water reactors only. The proposed rule was provided to the Commission in July 2002. The proposed rule will be published in the *Federal Register* for public comment for a period of one month. The final rule is scheduled to go to the Commission 15 months after the close of the public comment period, and the final rule will be published in the *Federal Register* one month after the staff requirements memorandum.

Mr. Weiss concluded by stating that the staff views this rulemaking as a necessary first step in providing an opportunity for licensees and NRC to be more efficient and effective in this regulatory environment. In response to questions from the Committee, the staff provided the following additional information:

- The NFPA 805 standard does not address getting the plant to cold shutdown. Instead, it specifies a safe and stable condition.

- The staff does not currently have any performance indicators for measuring the effectiveness of the performance based approaches allowed by the proposed rule.

- The staff has granted approximately 900 exemptions (industry wide) to the Appendix R requirements over the last 20 years. The staff recently evaluated the impact on CDF of the accumulated effects of all of the exemptions and found the change in CDF to be negligible.

- A licensee does not have to adopt NFPA 805 to change a requirement. They can always submit a risk-informed exemption request utilizing the methodologies in NFPA 805.

- Once the staff approves a new performance-based, risk- informed methodology for one licensee, others will be able to utilize the same methodology.

Mr. Fred Emerson, NEI, was then introduced to discuss the proposed rule from the industry perspective, and NEI's implementing guidance for the rule. Mr. Emerson stated that when the final NFPA 805 standard was approved in the fall of 2000, the industry still had some concerns related to the final rule and the final standard. These concerns were addressed during in the rulemaking process. Specifically,

- The industry believed that there should be an allowance for the use of performance-based methods to address the deterministic requirements of Chapter 3. Chapter 3 of the standard specifically prohibits the use of performance-based methods to address Chapter 3 requirements. The proposed rule provides an exception to allow the use of performance-based methods that have been approved by the staff.

- The industry also wanted to allow the use of a licensing bases as previously docketed instead of the "previously approved alternatives" currently included in the language of Chapter 3.

- The industry wanted the NRC to review performance-based methods instead of the NFPA Technical Committee, and the staff has agreed to do so.

Other major points from Mr. Emerson's presentation include:

- How NEI 00-01 will be used. NEI 00-01 is intended to be a risk-informed method for resolving the circuit failures issue. The development of this document is going on in parallel with the NFPA 805 rulemaking. NEI hopes that all open issues related to this document will be resolved before the rule is made final so that NEI 00-01 can be included as an acceptable method of performing engineering analysis for circuit failures (this would be one element of a licensee's risk-informed program).

- NEI agreed to develop the implementing guidance for the rulemaking, and the development of this guidance is moving forward in parallel with the rulemaking.

The implementing guidance is one of the vehicles for addressing issues as they arise during the rulemaking process. Other methods for addressing emergent issues during the rulemaking may include changes to the rule language directly and/or by placing the information in the statements of consideration for the rule. Once the NRC has accepted the implementing guidance, it is anticipated that they would endorse it in a regulatory guide. The guidance is being developed by a multi-discipline contractor team which addresses the various areas of fire protection that need to be addressed in a new rule (e.g., classic fire protection, safe shutdown, PSA, etc.).

- There are four fundamental industry positions related to the proposed rule. The first position relates to the increased use of risk information in fire protection. The intent of the proposed rule is that a licensee can choose an alternate licensing basis. However, many licensees have a fire protection licensing basis which they have been comfortable with for the last 20 years. The industry believes that if a licensee chooses not to adopt NFPA 805, they should still have the ability to use risk-informed, performance-based tools in a structured regulatory environment. The industry views this as an evolutionary process where licensees may gradually utilize performance-based, risk-informed methods to make changes to their fire protection licensing bases. The second position is that the rulemaking should be optional. This has never been a contentious issue with the staff. The third position is that the transition to the NFPA 805 standard as a licensing basis should be uncomplicated. The transition process is of great interest to any licensee who is going to be contemplating changing licensing bases. The fourth is the licensee should have a thorough understanding of the licensing basis through the transition.

- The industry is pretty positive about the current proposed rule language. They still have some concerns, but believe they will be addressed during the development of the final rule.

- The implementation guidance is currently under development. The major focus of the guidance is the transition process. The key elements of the guidance include what the process is, what are the licensee's options, guidance for licensees choosing to maintain their existing licensing basis with the possible use of 805 tools, guidance on adopting the new licensing basis, guidance of maintaining licensing basis configuration control once the transition has been made.

- The implementation guidance includes appendices on the use or interpretation of the provisions of NFPA 805 itself. Specifically, they contain such guidance as how to implement the program fundamentals, establish performance criteria, identify fire hazards in various systems, structures and components, and perform an evaluation against the performance criteria. The schedule for completing the implementation guidance is inconsistent with the rulemaking schedule.

- There are some potential barriers or hurdles. These need to be addressed during the development of the rule language and/or the implementation guidance. Examples of potential barrier issues include whether a license amendment is required for analytical methods, and the definition and use of the current licensing basis. An example of a potential hurdle is that NFPA 805 requires some new elements that are not currently part of Appendix R, such as monitoring at shutdown and low power modes, these are classified as potential hurdles because the industry must create a successful rule and matching implementation guidance.

- The major benefits of the proposed rule from the industry perspective include allowance for the use of risk methods for resolving current fire protection issues, and creating a licensing basis for fire protection that addresses the four NRC pillars of maintaining safety, increasing public confidence, reducing regulatory burden, and increasing agency efficiency and effectiveness. The rule will allow licensees to focus fire protection programs on those areas that are more risk-significant. Currently, this capability does not exist under Appendix R. The rule will provide a consistent method for supporting exemptions, deviations,] 10 CFR 50.59 and 86-10 evaluations. The rule will provide a seamless transition process from the deterministic to the risk-informed regulatory framework. Finally, the use of fire risk methods, integrating consideration of fire risk into overall plant risk will help licensees resolve issues for fire protection on an equal risk basis when considered against other plant issues that are competing for resources.

- In response to a question from the Fire Protection Subcommittee, Mr. Emerson explored the issue of whether licensees would utilize the proposed rule. He believes that there are a few plants that are considering using this rulemaking. The plants that are most likely to use it are the ones that are using risk techniques in their normal plant operations, and have established PRAs that they utilize. These plants will most likely be the first to adopt the new standard. Once the rest of the industry sees successful use and successful regulatory

interactions, using risk tools in the fire area, Mr. Emerson stated that more plants will move to adopt the new standard as they see the benefits.

From a perspective of public confidence, Mr. Paul Gunter, Nuclear Information Resource Service, discussed some specific concerns regarding the new rule. It provides a complicated overlay to the Appendix R requirements and he does not see how inspection and enforcement are being addressed in the overall picture. He stated that the public is looking for confidence in these areas. He also stated that the inspection process may be confounded by the new standard which would not enhance public confidence. He requested the staff to address the issue of how the NFPA 805 standard would make the inspection process more efficient and how it could be utilized for effective enforcement.

Mr. Weiss, NRR, provided the staff's response. Specifically, he stated that the staff recognized Mr. Gunter's concerns. The staff does have a vision that the proposed rule will work. However, training is needed. The staff is currently conducting quarterly training for inspectors and the staff is prepared to give the inspectors tools and training in this area. He pointed out that inspectors are very good at determining whether approved methods have been used, whether the people that use them are qualified to use them, and whether the configuration design was implemented properly in the plant. These strengths will be utilized to inspect in this area. Mr. Weiss stated that the staff would be able to construct an effective set of inspection criteria that would be tied to these strengths.

Committee Action

The Committee issued a letter to Dr. Travers on this matter dated June 17, 2002, recommending that the draft rule be issued for public comment.

VIII. Generic Resolution of Voids in the Concrete Containment (Open)

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

In the ACRS meeting of May 2-4, 2002, the Committee asked a question regarding voids in concrete containments. The issue was raised during the Committee's review of the Turkey Point license renewal application. Specifically, the Committee raised an issue related to the potential for the existence of undetected voids in concrete structures at nuclear power plants and their safety significance. The staff returned to

the Committee on June 7, 2002, to provide an answer to the Committee's question. The staff provided details of how voids would normally be detected during construction and testing of the containment. In addition, the staff provided information in support of the staff's conclusion that potential undetected voids in concrete structures do not have any generic safety implications. The specifics of the staff's presentation are explained in a memorandum from Mr. Eugene V. Imbro to Mr. Pao-Tsin Kuo, "Advisory Committee for Reactor Safeguards (ACRS) Inquiry Regarding Voids in Concrete Structures in Nuclear Power Plants," dated June 6, 2002.

Committee Action

The Committee found the staff's answer to be fully responsive to their question. No further concerns were identified.

IX. Format and Content of the 2003 ACRS Report on the NRC Safety Research Program (Open)

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

The Committee discussed the format and content of the 2003 ACRS report to the Commission regarding the NRC Safety Research Program. Dr. Ford indicated that some ACRS members would like the focus of the report to address new projects (new reactors, synergistic effects; i.e., license renewal/power uprate, fuels/fires, and possible security issues). The new projects could enhance NRC long-term capabilities in advanced reactor areas. Some of the key technical issues in the advanced reactor areas could be the adequacy of early site permits, core damage frequency/large early release frequency for multiple modules, and security.

Preliminary discussions indicate that one possible way to structure the content of the 2003 ACRS report on the NRC's safety research program is for the ACRS members to start from the following list of regulatory challenges that Dr. Kress developed as a result of the June 4-5, 2001 ACRS workshop on advanced reactors:

- Should we "force-fit" non-LWR designs into the current regulatory structure that is heavily focused on LWRs and the structuralist approach to defense in depth or should licensing take a "clean-sheet" risk-based approach?

- Do we need additional risk acceptance criteria (e.g., frequency-consequence curves) for designs for which core damage frequency and large, early release frequency are ill-posed concepts?

- How do we quantify and deal with PRA uncertainty for the new concepts?

- How robust must the containment (if any) be in view of the lack of experience with new designs and potentially large uncertainty in risk assessments?

- What is the acceptability of significantly reduced emergency response for designs with low source terms and long warning times before release?

- What is the acceptability of no additional emergency core cooling system for designs that have a great deal of water in the primary vessel and no pipes to break at locations that would drain the water (IRIS)?

- What role will "licensing by test" play in the regulatory process?

- Can the frequency of air ingress and a graphite/fuel fire be demonstrated to be acceptably low for those designs that use coated particle fuel?

- How do we establish LOCA frequencies for new plants for which there is not an extensive data base? The reliability of simplified passive safety systems?

- For the new designs, will we have appropriately qualified/validated probabilistic risk assessments and thermal hydraulic, neutronic, and safety assessment codes?

- Are there any new human performance and instrumentation and control issues?

- What process will be used to develop design basis accidents for the new designs? How will risk-significant structure, system, and components be identified? What will be the new regulatory treatment of non-safety systems?

- What use can be made of previous NRC reviews of derivative designs?

- For the new designs, what will be the regulatory approach with respect to: (a) many new plants and (b) multi-unit/module sites?

- How can we assure the required fuel quality for designs for which the safety case relies heavily on fuel integrity (focus on process versus product)? Validation of fuel performance? Accident source terms for design basis events? High burnup?

- How will once-through cores that last 5-10 years be monitored and inspected? How will acceptability of new "smart" on-line health monitoring systems?

- How will financially related requirements apply to the new concepts and business environment? Extension of Price-Anderson? NRC fee structure for multi-modules? Decommissioning?

- What is the acceptability of natural circulation cooling as an accident recovery strategy? What is the availability and validation status of appropriate analytical models?

- What data base exists for the lifetime temperature and irradiation behavior of graphite? How do the "new" graphites compare to the "old" ones?

- Can the NRC develop the manpower, resources, and technical expertise required for assessing the future concepts?

- Is there a sufficient data base on high temperature material behavior for the gas-cooled concepts with high exit temperatures? What are the gas turbine safety issues for high temperature helium cycles?

- What new NRC research is needed? Participation with industry? Use of international research and data bases?

- Are new approaches/criteria needed for licensing multi-purpose plants (e.g., power plus desalination, industrial/residential heating, hydrogen production, coal conversion)?

- What will be the spent fuel storage requirements for new fuel types/geometries? Potentially damaged fuel spheres/particles?

The ACRS members are expected to view the list of the above regulatory challenges and determine if these challenges could be addressed using the existing RES tool projects.

Committee Action

The Committee will continue its effort on the 2003 ACRS report to the Commission regarding the NRC Safety Research Programs during future meetings.

X. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations

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

- The Committee discussed the EDO's response of May 13, 2002 to comments and recommendations included in the ACRS March 19, 2002 report related to Risk-Informing Special Treatment Requirements of 10 CFR Part 50 (Option-2).

  Several Committee members continue to disagree with the staff on the value of uncertainty analysis to support development of importance measures and changes in core damage frequency (Δ-CDF). The Committee decided to continue its discussion of this matter during the July 2002 meeting.

- The Committee considered the response from the EDO, dated May 6, 2002, to comments and recommendations included in the ACRS report dated March 14, 2002, concerning Arkansas Nuclear One, Unit 2 Extended Power Uprate.

  The Committee decided to continue its discussion of this matter during the July 2002 meeting.

- The Committee considered the response from the EDO, dated May 31, 2002, to comments and recommendations included in the ACRS report dated April 19, 2002, concerning a report on the safety aspects of the license renewal application for the Turkey Point Nuclear Plant, Units 3 & 4.

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

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

The Committee heard a report from ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on May 30, 2002. The following items were discussed:

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through September 2002 was discussed. The objectives were:

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

During this session, the Subcommittee discussed and developed recommendations on the items that require Committee decision.

Quadripartite Meeting Update

The Quadripartite meeting is scheduled to be held October 23-25, 2002, in Berlin, Germany. On May 16, 2002, the RSK sent the latest agenda for this meeting.

The following technical papers will be discussed at the Quadripartite meeting:

- Safety Culture and Safety Management
- Risk-Informed Regulation
- Thermal-Hydraulic Analysis and Code Issues

- Stress Corrosion Cracks in Pressure Retaining Components in Nuclear Power Plants
- Risk Analysis of Spent Fuel Storage

Dr. Larkins has informed the Commission regarding the Quadripartite meeting, including forwarding the draft agenda. Coordination of this trip is being done through the Office of International Programs. In connection with the Quadripartite meeting trip, several members will visit a MOX facility in France.

Staff Requirements Memorandum

As a result of the ACRS meeting with the Commission on December 5, 2001, a Staff Requirements Memorandum dated December 20, 2001, requested the following:

- The ACRS should continue to review staff efforts on risk-based PIs and improvements to the significance determination process.

- The staff, with ACRS input, should provide recommendations for resolving, in a transparent manner, apparent conflicts and discrepancies between aspects of the revised reactor oversight process that are risk-informed (e.g., significance determination process) and those that are performance-based (e.g., performance indicators).

- The ACRS should continue its efforts to ascertain regulatory challenges for future reactor designs. The Committee should also ensure that it is prepared to review NRC staff efforts on advanced reactors in the near term, including issues related to Westinghouse's AP1000, General Atomics' GTMHR, and Exelon's pebble bed modular reactor.

During the February 2002 ACRS meeting, cognizant members developed a plan for addressing the reactor oversight process issues in the SRM. In May 2002, the NRC staff developed a standard development plan to improve strategies and an associated plan to improve the Task Action Plan. During the May meeting, Dr. Kress provided a draft plan to address the SRM issues on future plant designs and included an ACRS review of early site permit applications. In addition, Dr. Apostolakis and Mr. Sieber met with representatives of the NRC staff and discussed the staff plans to address the reactor oversight process issues specified in the SRM.

ACRS Meeting with the NRC Commissioners

The ACRS Members will meet with the Commissioners on Wednesday, July 10, 2002, between 2:00 and 4:00 p.m. The topics that will be discussed are:

- Overview by the ACRS Chairman
  - Status of ACRS activities on power uprates and license renewal
- Advanced reactor designs
- Risk-Informing Special Treatment Requirements of 10 CFR Part 50
- PTS reevaluation project

Celebration of the 500th ACRS Meeting

As agreed to by the members, invitations were sent to the NRC Commissioners to participate at the 500th ACRS meeting ceremony, which is scheduled for March 4-5, 2003. (This is also coincidental with the Committee's 50th Anniversary). So far, Chairman Meserve, Commissioner Dicus, and Commissioner McGaffigan have agreed to participate. Dr. Hal Lewis, Dr. Robert Seale, Dr. Bill Stratton, and Mr. Dave Ward have also agreed to participate in the celebration.

ACRS Senior Fellow Position

Dr. Hossein Nourbaksh has accepted the Senior Fellow position and will report to work subsequent to completion of all necessary paper work. He has a Ph.D. in chemical engineering; and, he is the President and founder of Energy and Environmental Science, Inc., a consulting company.

Arrangements for the Visit to Watts Bar Nuclear Plant and Region II

ACRS Members are scheduled to tour the Watts Bar Nuclear Power Plant on June 18 and hold a meeting with Region II personnel on June 19, 2002.

Revised ACRS Subcommittee Structure

A revised Subcommittee structure was discussed. This revision reflects the creation of a new Subcommittee on Safeguards and Security, establishment of two Plant License

Renewal Subcommittees, and member/staff reassignments. The ACRS Chairman has approved the revision and it will become effective on June 10, 2002.

Workshop on Nuclear Regulatory Decisionmaking Process in Switzerland

The Swiss Federal Safety Inspectorate (HSK), in coordination with the International Atomic Energy Agency (IAEA), is organizing a Workshop on Nuclear Regulatory Decisionmaking Process which will be held in Switzerland on October 16-18, 2002. Drs. Bonaca, Kress, and Ransom will participate at this Workshop. This will include a presentation on ACRS Perspective on Regulatory Decisionmaking and a Panel Discussion. Dr. Kress will prepare a paper, for Committee review and presentation at the workshop, on the ACRS Perspectives on Regulatory Decisionmaking. Dr. Bonaca will represent the ACRS on the Panel discussion on Issues and Opportunities for the Evaluation and Improvement of Nuclear Regulatory Decisionmaking Process.

Member Compensation Reports

The agency is operating under a new time and labor reporting system, Starfire. The time by labor category reported in that system will be used to make management decisions regarding each office's contribution to the agency's strategic goals. Members' time by labor category must be reported in the agency's Starfire system so that the true ACRS/ACNW office contribution is realized. To accomplish this, a data entry clerk from the Payroll Office will use the members' compensation sheets to enter each members' time by labor category into the Starfire system. Since the system has standardized reporting categories referred to as "activity codes," it is very important that the members' time be captured by the same activity codes when reporting their time. Each member will receive a laminated card with the labor category title and corresponding activity code as well as new compensation forms. The information on the laminated card (also on the backside of the compensation report) must be used when filling out the member compensation report. In addition, time must be attributed to an activity code for all Committee meetings and for preparation time. The nature of the work such as travel, meeting preparation, and type of meeting must also continue to be identified.

Role and Use of PRA in the ACRS Review Process

Mr. Sieber provided several comments on the Committee's use of PRA results in its review of regulatory issues (e.g., power uprates, license renewal), including the following:

- The Committee tries to use risk information in making decisions on regulatory matters. He continues to believe that risk information is one of the tools we can use to decide what is correct.

- The Commission's Safety Goal Policy defines "what is safe enough." The issue of the use of PRA in relation to the Safety Goal Policy seems to continually arise. In his view, the ACRS has addressed this issue only in a fragmented way.

- Adding individual opinions on PRA and Safety Goal Policy to our reports on licensee applications is not the appropriate avenue to transmit our concerns to the staff or the Commission.

Mr. Sieber recommended that a "White Paper" be prepared to address the adequacy of PRAs and issues associated with the Safety Goal Policy. The Committee should plan on discussing this matter during its 2003 retreat and subsequently develop comments and recommendations to the Commission as warranted.

In addition, Dr. Powers discussed several issues about how PRAs are used in License Renewal, Power Uprates and in the future. Basically, Dr. Powers suggests that because of enhanced computational capability that the ACRS could propose that PRAs could be expanded to cover issues and phenomena that are either excluded or treated in a less than complete manner.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 494th ACRS Meeting, July 10-12, 2002.

The 493rd ACRS meeting was adjourned at 3:00 pm on June 8, 2002.

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

August 21, 2002

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

FROM: George E. Apostolakis, Chairman
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFIED MINUTES OF THE 493rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), JUNE 6-8, 2002

I certify that based on my review of the minutes from the 493rd 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.

George E. Apostolakis, Chairman

August 21, 2002
Date

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

August 8, 2002

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 493rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - JUNE 6-8, 2002

Enclosed are the proposed minutes of the 493rd 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.

Attachment:
As stated

University of Virginia in Charlottesville, Virginia, and the Louisa County Public Library in Mineral, Virginia, have agreed to make the draft supplement to the GEIS available for public inspection.

Any interested party may submit comments on the draft supplement to the GEIS for consideration by the NRC staff. To be certain of consideration, comments on the draft supplement to the GEIS and the proposed action must be received by August 1, 2002. Comments received after the due date will be considered if it is practical to do so, but the NRC staff is able to assure consideration only for comments received on or before this date. Written comments on the draft supplement to the GEIS should be sent to: Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, Mail Stop T–6D 59, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

Comments may be hand-delivered to the NRC at 11545 Rockville Pike, Rockville, Maryland, between 7:45 a.m. and 4:15 p.m. on Federal workdays. Electronic comments may be submitted to the NRC by the Internet at *NorthAnnaEIS@nrc.gov*. All comments received by the Commission, including those made by Federal, State, and local agencies, Indian tribes, or other interested persons, will be made available electronically at the Commission's Public Document Room in Rockville, Maryland and from the Publicly Available Records (PARS) component of NRC's document system (ADAMS).

The NRC staff will hold a public meeting to present an overview of the draft plant-specific supplement to the GEIS and to accept public comments on the document. The public meeting will be held at the Public Room in the Louisa County Office Building, 1 Woolfolk Avenue, Louisa, Virginia, on June 25, 2002. There will be two sessions to accommodate interested parties. The first session will commence at 1:30 p.m. and will continue until 4:30 p.m. The second session will commence at 7:00 p.m. and will continue until 10:00 p.m. Both meetings will be transcribed and will include (1) a presentation of the contents of the draft plant-specific supplement to the GEIS, and (2) the opportunity for interested government agencies, organizations, and individuals to provide comments on the draft report. Additionally, the NRC staff will host informal discussions one hour prior to the start of each session at the Louisa County Office Building. No comments on the proposed scope of the supplement to the GEIS will be accepted during the informal discussions. To be considered, comments must be provided either at the transcribed public meetings, or in writing as discussed above. Persons may pre-register to attend or present oral comments at the meeting by contacting Mr. Andrew J. Kugler by telephone at 1–800–368–5642, extension 2828, or by Internet to the NRC at *NorthAnnaEIS@nrc.gov* no later than June 18, 2002. Members of the public may also register to provide oral comments within 15 minutes of the start of each session. Individual oral comments may be limited by the time available, depending on the number of persons who register. If special equipment or accommodations are needed to attend or present information at the public meeting, the need should be brought to Mr. Kugler's attention no later than June 18, 2002, to provide the NRC staff adequate notice to determine whether the request can be accommodated.

FOR FURTHER INFORMATION, CONTACT: Mr. Andrew J. Kugler, License Renewal and Environmental Impacts Program, Division of Regulatory Improvement Programs, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Mr. Kugler may be contacted at the aforementioned telephone number or e-mail address.

Dated at Rockville, Maryland, this 23rd day of April, 2002.

For the Nuclear Regulatory Commission.

Pao-Tsin Kuo,

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

[FR Doc. 02–12255 Filed 5–15–02; 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 June 6–8, 2002, in Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

Thursday, June 6, 2002

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.: CRDM Cracking of Vessel Head Penetrations and Vessel Head Degradation (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding issues related to the investigation of circumferential cracks in PWR control rod drive mechanism (CRDM) penetration nozzles and weldments, and reactor pressure vessel head degradation at the Davis-Besse Nuclear Power Plant.

10:45 a.m.–12:15 p.m.: Technical Assessment Generic Safety Issue (GSI)–189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding its technical basis and proposed recommendations for resolving GSI–189.

1:15 p.m.—2:15 p.m.: Technical Assessment of GSI–168, Environmental Qualification of Low-Voltage Instrumentation and Control Cables (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding its technical basis and proposed recommendations for resolving GSI–168.

2:15 p.m.–3:30 p.m.: Development of Reliability/Availability Performance Indicators and Industry Trends (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the staff's initiatives to integrate the NRC programs for risk-based analysis of reactor operating experience into the reactor oversight process, specifically the development of reliability/availability performance indicators and industry trends.

3:45 p.m.–4:45 p.m.: Technical and Policy Issues Related to Advanced Reactors (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding technical and policy issues related to advanced reactors.

5:00 p.m.–7:15 p.m.: Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting. In addition, the Committee may discuss a proposed report regarding confirmatory research program on high burnup fuel.

Friday, June 7, 2002

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.

8:35–10 a.m.: Proposed Rulemaking to Endorse National Fire Protection

Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the Nuclear Energy Institute regarding the proposed rulemaking to endorse NFPA 805 fire protection standard, and related matters.

10:15 a.m.–11:15 a.m.: Generic Resolution of Voids in the Concrete Containment (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the generic resolution of the issue of voids in the concrete containment walls.

11:15 a.m.–12 Noon: 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, and organizational and personnel matters relating to the ACRS.

12–12:15 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.

1:15 p.m.–7:15 p.m.: Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, June 8, 2002

8:30 a.m.–10 a.m.: Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

10:15 a.m.–11:30 a.m.: Discussion of Topics for Meeting with the NRC Commissioners (Open)—The Committee will discuss topics for meeting with the NRC Commissioners, which is scheduled for July 10, 2002.

12:45 p.m.–1:45 p.m.: Format and Content of the 2003 ACRS Report on the NRC Safety Research Program (Open)—The Committee will discuss the format, content, schedule, and assignments for the 2003 ACRS report to the Commission on the NRC Safety Research Program.

1:45 p.m.–2:45 p.m.: Proposed papers for the Quadripartite Meeting (Open)—The Committee will discuss proposed technical papers on specific topics that will be discussed at the Quadripartite meeting scheduled to be held on October 23–25, 2002, in Berlin, Germany.

2:45 p.m.–3:00 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 3, 2001 (66 FR 50462). 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 and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify Dr. Sher Bahadur, ACRS, 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 Dr. Sher Bahadur 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 Dr. Sher Bahadur if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting Dr. Sher Bahadur (telephone 301–415–0138), between 7:30 a.m. and 4:15 p.m., EDT.

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/NRC/ADAMS/index.html*.

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

Dated: May 10, 2002.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 02–12256 Filed 5–15–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Fire Protection; Notice of Meeting

The ACRS Subcommittee on Fire Protection will hold a meeting on June 4, 2002, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The agenda for the subject meeting shall be as follows:

Tuesday, June 4, 2002—8:30 A.M. Until the Conclusion of Business

The Subcommittee will review (1) the proposed revision to 10 CFR 50.48 to endorse the National Fire Protection Association (NFPA) standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," and (2) the Nuclear Energy Institute (NEI) guidance document NEI 00–01, "Guidance for Post-Fire Safe-Shutdown Circuit Analysis". The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

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

May 9, 2002

SCHEDULE AND OUTLINE FOR DISCUSSION
493RD ACRS MEETING
JUNE 6-8, 2002

THURSDAY, JUNE 6, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - ~~10:30~~ 10:45 A.M. CRDM Cracking of Vessel Head Penetrations and Vessel Head Degradation (Open) (FPF/MWW)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff regarding issues related to the investigation of circumferential cracks in PWR control rod drive mechanism (CRDM) penetration nozzles and weldments, and reactor pressure vessel head degradation at the Davis-Besse Nuclear Power Plant.

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

~~10:30~~ - ~~10:45~~ 10:45 - 11:00 A.M. ***BREAK***

3) ~~10:45~~ 11:00 - ~~12:15~~ 1:04 P.M. Technical Assessment Generic Safety Issue (GSI)-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident" (Open) (TSK/RBE/SD)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding its technical basis and proposed recommendations for resolving GSI-189.

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

~~12:15~~ - ~~1:15~~ 1:04 - 1:40 P.M. ***LUNCH***

4) ~~1:15~~ 1:40 - ~~2:15~~ 2:40 P.M. Technical Assessment of GSI-168, Environmental Qualification of Low-Voltage Instrumentation and Control Cables (Open) (GML/TJK/SD)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff regarding its technical basis and proposed recommendations for resolving GSI-168.

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

5) 2:50 - 4:15 ~~2:15 - 3:30~~ P.M. Development of Reliability/Availability Performance Indicators and Industry Trends (Open) (MVB/AWC/MWW)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's initiatives to integrate the NRC programs for risk-based analysis of reactor operating experience into the reactor oversight process, specifically the development of reliability/availability performance indicators and industry trends.

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

4:15 - 4:30 ~~3:30 - 3:45~~ **P.M.** ***BREAK***

6) 4:30 - 5:40 ~~3:45 - 4:45~~ P.M. Technical and Policy Issues Related to Advanced Reactors (Open) (TSK/MME)

6.1) Remarks by the Subcommittee Chairman

6.2) Briefing by and discussions with representatives of the NRC staff regarding technical and policy issues related to advanced reactors.

~~**4:45 - 5:00 P.M.** ***BREAK***~~

7) 5:00 - 7:45 ~~7:15~~ P.M. Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

7.1) CRDM Cracking of Vessel Head Penetrations and Vessel Head Degradation (FPF/MWW)

7.2) Technical Assessment of GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident" (TSK/RBE/SD)

7.3) Technical Assessment of Generic Safety Issue-168, "Environmental Qualification of Low-Voltage I&C Cables" (GML/TJK/SD)

7.4) Development of Reliability/Availability Performance Indicators and Industry Trends (MVB/AWC/MWW))

7.5) Confirmatory Research Program on High Burnup Fuel (Tentative) (TSK/TJK/MME)

7.6) Technical and Policy Issues Related to Advanced Reactors (Tentative) (TSK/MME)

FRIDAY, JUNE 7, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 - ~~10:00~~ 10:20 A.M. Proposed Rulemaking to Endorse National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" (Open) (SLR/RBE/SD)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Energy Institute regarding the proposed rulemaking to endorse NFPA 805 fire protection standard, and related matters.

~~10:00~~ - ~~10:15~~ 10:20 - 10:30 A.M. ***BREAK***

10) ~~10:15~~ - ~~11:15~~ 10:30 - 10:55 A.M. Generic Resolution of Voids in the Concrete Containment (Open) (MVB/RBE/SD)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff regarding the generic resolution of the issue of voids in the concrete containment walls.

11) ~~11:15~~ - ~~12:00~~ 11:05 - 12:20 Noon. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/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, and organizational and personnel matters relating to the ACRS.

12) ~~12:00~~ - ~~12:15~~ 1:20 - 1:42 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GEA, et al./SD, et al.)

Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

~~12:15~~ - ~~1:15~~ 12:20 - 1:20 P.M. ***LUNCH***

13) 1:15 - 7:15 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS Reports on:

7:00-7:30 13.1) CRDM Cracking of Vessel Head Penetrations and Vessel Head Degradation (FPF/MWW)

6:15-7:00 13.2) Technical Assessment of GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident" (TSK/RBE/SD)

1:57-3:00 & 3:15-3:45 13.3) Technical Assessment of GSI-168, "Environmental Qualification of Low-Voltage I&C Cables" (GML/TJK/SD)

13.4) Development of Reliability/Availability Performance Indicators and Industry Trends (MVB/AWC/MWW)

5:10-6:05 13.5) Proposed Rulemaking to Endorse NFPA 805 Fire Protection Standard (SLR/RBE/SD)

13.6) Confirmatory Research Program on High Burnup Fuel (Tentative) (TSK/TJK/MME)

3:50-4:30 & 4:35-4:45 13.7) Technical and Policy Issues Related to Advanced Reactors (Tentative) (TSK/MME)

SATURDAY, JUNE 8, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 10:00 A.M. Proposed ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 13.
9:50-12:50 CRDM letter discussion

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

15) 10:15 - 11:30 A.M. Discussion of Topics for Meeting with the NRC Commissioners (Open) (GEA, et al./JTL, et al.)
Discussion of topics for meeting with the NRC Commissioners on July 10, 2002.

11:30 - 12:45 P.M. ***WORKING LUNCH***

16) ~~12:45~~ 1:40 - ~~1:45~~ 2:00 P.M. Format and Content of the 2003 ACRS Report on the NRC Safety Research Program (Open) (FPF/MME)

16.1) Remarks by the Subcommittee Chairman

16.2) Discussion of the format, content, schedule, and assignments for the 2003 ACRS report to the Commission on the NRC Safety Research Program.

17) 1:45 - 2:45 P.M. Proposed Papers for the Quadripartite Meeting (Open) (GEA, et al./JTL, et al.)
Discussion of proposed papers on the following:

17.1) Safety Culture and Safety Management (MVB/DAP)

17.2) Risk-Informed Regulation (GEA/TSK)

17.3) Thermal-Hydraulic Analysis and Code Issues (GBW/VHR)

17.4) Stress Corrosion Cracks in Pressure Retaining Components in Nuclear Power Plants (FPF/WJS)

17.5) Risk Analysis of Spent Fuel Storage (TSK/DAP)

18) 2:45 - 3:00 P.M. Miscellaneous (Open) (GEA/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) copies of the presentation materials should be provided to the ACRS.**

APPENDIX III: MEETING ATTENDEES

493rd ACRS MEETING
JUNE 6-8, 2002

NRC STAFF (June 6, 2002)

J. Grobe, RIII
D. Snowberger, NRR
B. Bateman, NRR
K. Wichman, NRR
A. Hiser, NMRR
S. Coffin, NRR
A. Lee, NRR
S. Blou, NRR
E. Hackett, RES
K. Chang, NRR
T. Markley, NRR
M. Snodderly, NRR
H. Vandermolen, RES
A. Malliakos, RES
M. Marshall, NRR
S. Feld, RES
R. Dudley, NRR
S. Uttaz, OGC
M. Rubin, NRR
B. Palla, NRR
M. Franovich, NRR
A. Rubin, RES
C. Ader, RES
J. Ridgely, RES
M. Johnson, NRR
A. Levin, OCM/RAM
P. Shemanski, NRR
N. Chokshi, RES
S. Lee, NRR
P. Kang, NRR
P. T. Kuo, NRR
D. Hickman, RES
J. Thompson, NRR
M. Satorious, NRR
R. Frahm, NRR
H. Hamzehee, RES
C. Carpenter, NRR
I. Jung, NRR
T. Boyce, NRR
W. Liu, NRR
B. Brady, RES
A. Druzo, NRR
F. Eltawila, RES
A. Cubbage, NRR
J. Wilson, NRR
J. Mitchell, RES
J. Flack, RES
J. Lee, RES
S. LaVie, NRR
M. Hart, NRR
S. Steele, NMSS
M. Drouin, RES
S. Magruder, NRR
J. Muscara, RES
S. Ali, RES
P. Bennett, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

D. Raleigh, LIS Scientech
R. Schranter, FENOC
J. Meyer, ISL
J. Lehner, BNL
M. Barrett, Duke Energy
B. Bryan, TVA
C. Brinkman, Westinghouse
P. Gunter, NIRS
B. Horin, Winston & Strawn
T. Houghton, NEI
S. Eich, INEEL
B. Youngblood, ISL
P. Holzman, STAR, Inc.
J. Stevens, ManTech
P. Negus, GE
J. Riccio, Green Peace
J. Weil, McGraw-Hill
J. Rycyna, CNS

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
D. Dellario, CNS
J. Roe, Scientech
K. Spencer, FENOC
D. Labott, BEG
P. McCluskey, FENOC
M. McLaughlin, FENOC
L. Mathews, Southern Nuclear
S. Hunt, Dominion
A. Marion, NEI
R. Huston, LSS
J. Powers, FENOC
S. Loehlein, FENOC
S. Fyfitil, FRA-ANP
J. Hickling, EPRI
P. Riccardella, Structural Integrity
D. Horner, McGraw-Hill
K. Colens, NEI
T. Tseng, Tecro
T. Henry, Toledo Edison

NRC STAFF (June 7, 2002)
A. Singh, RES
E. Weiss, NRR
S. Black, NRR
J. Hannon, NRR
E. Connell, NRR
L. Whitney, NRR
J. Hyslys, RES
P. Kang, NRR
P. T. Kuo, NRR
J. S. Ma, NRR
J. Strasma, NRR
G. Imbro, NRR
D. Terao, NRR
G. Klein, NRR
G. Bagchi, NRR
H. Graves, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC (June 7, 2002)

H. Thornburg, Consultant
F. Emerson, NEI
R. Huston, LSS
P. Gunter, NIRS

APPENDIX IV

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 18, 2002

SCHEDULE AND OUTLINE FOR DISCUSSION
494th ACRS MEETING
JULY 10-12, 2002

WEDNESDAY, JULY 10, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:00 A.M. Pressurized Thermal Shock (PTS) Reevaluation Project: Risk Acceptance Criteria (Open) (TSK/FPF/MWW)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff regarding the risk metrics and associated criteria that can be used in reevaluating the technical basis of the PTS rule.

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

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

3) 10:15 - 11:15 A.M. Draft Final Revision 1 to Regulatory Guide 1.174, "An Approach to Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and Associated Standard Review Plan Chapter 19 (Open) (GEA/AWC/PAB)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final revision 1 to Regulatory Guide 1.174 and the associated Standard Review Plan Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance."

4) 11:15 - 12:45 P.M. Discussion of Topics for Meeting with the NRC Commissioners (Open) (GEA, et al./JTL, et al.)
   Discussion of topics for meeting with the NRC Commissioners on July 10, 2002.

12:45 - 2:00 P.M. ***LUNCH***

5) 2:00 - 4:00 P.M. Meeting with the NRC Commissioners (Open) (GEA, et al/ JTL, et al.)
   The Committee will meet with the NRC Commissioners, Commissioners' Conference Room, One White Flint North, to discuss the following:

   - Overview (GEA)
     - Core Power Uprates and License Renewals
     - Future Committee Activities

- Advanced Reactors (TSK)
- Risk-Informing Special Treatment Requirements of 10 CFR Part 50 (GEA)
- Pressurized Thermal Shock Technical Basis Reevaluation Project (FPF)

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

6) 4:15 - 5:15 P.M. Risk-Informed Regulation Implementation Plan (Open) (GEA/AWC/HJL)
   6.1) Remarks by the Subcommittee Chairman
   6.2) Briefing by and discussions with representatives of the NRC staff regarding the updated version of the Risk-Informed Regulation Implementation Plan.

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

7) 5:30 - 7:15 P.M. Proposed ACRS Reports (Open)
   Discussion of proposed ACRS reports on:
   7.1) PTS Reevaluation Project: Risk Acceptance Criteria (TSK/FPF/MWW)
   7.2) Draft Final Revision 1 to Regulatory Guide 1.174 and SRP Chapter 19 (GEA/AWC/PAB)
   7.3) Risk-Informed Regulation Implementation Plan (GEA/AWC/HJL)

THURSDAY, JULY 11, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 - 10:00 A.M. Advanced Reactors Research Plan (Open) (TSK/MME)
   9.1) Remarks by the Subcommittee Chairman
   9.2) Briefing by and discussions with representatives of the NRC staff regarding the modifications and updates to the Advanced Reactors Research Plan.

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

10) 10:15 - 12:00 Noon Overview of NRC Research Activities in the Seismic Area (Open) (DAP/TJK/SD)
   10.1) Remarks by the Subcommittee Chairman
   10.2) Briefing by and discussions with representatives of the NRC staff regarding ongoing and proposed research activities as well as new research needs in the seismic area.

12:00 - 1:00 P.M. *****LUNCH*****

11) 1:00 - 2:30 P.M. Development of Review Standard for Reviewing Core Power Uprate Applications (Open) (GBW/JDS/PAB)

11.1) Remarks by the Subcommittee Chairman

11.2) Briefing by and discussions with representatives of the NRC staff regarding the development of a "Review Standard" for use in future reviews of core power uprate applications.

2:30 - 2:45 P.M. ***BREAK***

12) 2:45 - 6:00 P.M. Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

12.1) PTS Reevaluation Project: Risk Acceptance Criteria (TSK/FPF/MWW)

12.2) Draft Final Revision 1 to Regulatory Guide 1.174 and SRP Chapter 19 (GEA/AWC/PAB)

12.3) Risk-Informed Regulation Implementation Plan (GEA/AWC/HJL)

12.4) Advanced Reactors Research Plan (TSK/MME)

12.5) Development of Review Standard for Reviewing Core Power Uprate Applications (Tentative) (GBW/JDS/PAB)

FRIDAY, JULY 12, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

13) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GEA/JTL/SD)

14) 8:35 - 10:15 A.M. Application of the Probabilistic Fracture Mechanics Methodologies to Reactor Vessel Integrity Assessment (Open) (FPF/AWC/MME)

14.1) Remarks by the Subcommittee Chairman

14.2) Briefing by and discussions with representatives of the NRC staff regarding application of the probabilistic fracture mechanics methodologies (including the FAVOR computer code) to assess reactor pressure vessel integrity.

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

15) 10:30 - 2:30 P.M. Proposed ACRS Reports (Open)

(12:00 - 1:00 P.M. LUNCH) Continue discussion of proposed ACRS reports listed under Item 12.

2:30 - 2:45 P.M. ***BREAK***

16) 2:45 - 3:30 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL/SD)

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

16.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

17) 3:30 - 3:45 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GEA, 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.

18) 3:45 - 4:45 P.M. Format and Content of the 2003 ACRS Report on the NRC Safety Research Program (Open) (FPF/RPS)
18.1) Remarks by the Subcommittee Chairman
18.2) Discussion of the format, content, schedule, and assignments for the 2003 ACRS report to the Commission on the NRC Safety Research Program.

4:45 - 5:00 P.M. *****BREAK*****

19) 5:00 - 6:00 P.M. Proposed Papers for the Quadripartite Meeting (Open)) (GEA, et al./ JTL, et al.)
Discussion of proposed papers on the following:
19.1) Safety Culture and Safety Management (MVB/DAP)
19.2) Risk-Informed Regulation (GEA/TSK)
19.3) Thermal-Hydraulic Analysis and Code Issues (GBW/VHR)
19.4) Stress Corrosion Cracks in Pressure Retaining Components in Nuclear Power Plants (FPF/WJS)
19.5) Risk Analysis of Spent Fuel Storage (TSK/DAP)

20) 6:00 - 6:30 P.M. Miscellaneous (Open) (GEA/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) copies of the presentation materials should be provided to the ACRS.**

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
493RD ACRS MEETING
JUNE 6-8, 2002

[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 June 6-8, 2002
2	CRDM Cracking of Vessel Head Penetrations and Vessel Head Degradation
	2. Update of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Closure Head Activities presentation by FENOC [Viewgraphs]
	3. Time Line presentation by FENOC [Viewgraphs]
	4. NRC Staff Activities on Vessel Head Penetration Nozzle Cracking and Vessel Head Degradation presentation by B. Bateman, NRR [Viewgraphs]
	5. Ongoing NRC Regulatory Activities at Davis Besse presentation by J. Grobe, Region III [Viewgraphs]
3	Technical Assessment Generic Safety Issue (GSI)-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident"
	6. Technical Assessment of GSI-189 presentation by RES [Viewgraphs]
4	Technical Assessment of GSI-168, Environmental Qualification of Low-Voltage Instrumentation and Control Cables
	7. Technical Assessment of GSI-168 Environmental Qualification of Low-Voltage I&C Cables presentation by RES [Viewgraphs]
5	Development of Reliability/Availability Performance Indicators and Industry Trends
	8. Revised Oversight Process Performance Indicator Pilot Program Mitigating System Performance Index presentation by RES [Viewgraphs]
	9. Industry Trend Programs presentation by NRR [Viewgraphs]
6	Technical and Policy Issues Related to Advanced Reactors
	10. Potential Policy Issues for Advanced Reactors presentation by RES [Viewgraphs]

9 Proposed Rulemaking to Endorse National Fire Protection Associated (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants"

11. Proposed Agenda [Handout]
12. NFPA 805 Rulemaking presentation by NRR [Viewgraphs]
13. Current Draft Proposed Rule Language [Handout]
14. Risk-Informed, Performance-Based Fire Protection Implementing Guidance presentation by NEI [Viewgraphs]

10 Generic Resolution of Voids in the Concrete Containment

15. Potential Undetected Voids in Concrete Structures of Nuclear Power Plants presentation by NRR [Viewgraphs]

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

16. Future Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - May 30, 2002 [Handout #11]

12 Reconciliation of ACRS Comments and Recommendations

17. Reconciliation of ACRS Comments and Recommendations [Handout #12]

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TAB	DOCUMENTS
2	CRDM Penetration Cracking and RPV Head Degradation
	1. Table of Contents
	2. Proposed Schedule
	3. Status Report
	4. FENOC Press Release, FENOC Acquires Unused Reactor Vessel Head
	5. In The News, May 24, 2002, Commentary on NRC and Davis-Besse
	6. News Release, May 20, 2002, NRC Lessons Learned Task Force
	7. Inside NRC, May 20, 2002, Congressional Investigation
	8. NRC RES-MRP Alloy 600 Meeting Slides, May 22, 2002
	9. Letter to J. E. Dyer, Region III Administrator, from H. Bergendahl, FENOC, Subject: "Transmittal of Davis-Besse Nuclear Power Station, Unit 1 Return to Service Plan, " May 21, 2002
	10. Letter to J. E. Dyer, Region III Administrator, from H. Bergendahl, FENOC, Subject: "Supplemental Information in Response to NRC Question Number 24 on the Preliminary Probable Cause Summary Report dated March 22, 2002," May 15, 2002
3	GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident"
	11. Table of Contents
	12. Agenda
	13. Status Report
	14. Memorandum to John T. Larkins from Farouk Eltawila, "RES Proposed Recommendation for Resolving Generic Safety Issue 189: "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident," dated May 13, 2002
	15. Information Systems Laboratories, Inc. report entitled, "Backup Power for PWRs with Ice Condenser Containments and for BWRs with Mark III Containments under SBO Conditions: Impact Assessment," dated May 1, 2002
	16. Brookhaven National Laboratory draft letter report entitled, "Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants," dated April 25, 2002
	17. Abstract from NUREG/CR-5586, "Mitigation of Direct Containment Heating and Hydrogen Combustion Events in Ice Condenser Plants," October 1990
4	Proposed Technical Assessment of Generic Safety Issue-168, "Environmental Qualification of Low Voltage Instrument and Control (I&C) Cables"

18. Table of Contents
19. Proposed Schedule
20. Status Report
21. Letter to R. Meserve, Chairman, NRC, dated November 15, 2000, from D. Powers, Chairman, ACRS, Subject: License Renewal Guidance
22. Letter to D. Powers, Chairman, ACRS, dated December 22, 2000, from W. Travers, EDO, Subject: Response to ACRS letter to Chairman Meserve, dated November 15, 2000, "License Renewal Documents"
23. Memorandum from M. Mayfield to J. Larkins, dated May 6, 2002, Subject: Generic Safety Issue (GSI) "Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables, including enclosure of Technical Assessment of Generic Safety Issue GSI-168, Generic Safety Issue Environmental Qualification of Low-Voltage Instrumentation and Control Cables.

5 Development of Reliability/Availability Performance Indicators and Industry Trends

24. Table of Contents
25. Proposed Schedule
26. Status Report

6 Technical and Policy Issues Related to Advanced Reactors

27. Table of Contents
28. Proposed Schedule
29. Status Report
30. DRAFT Commission Paper (predecisional) dated May 23, 2002
31. NUREG-1226, Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants (NRC)

NRC

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

493rd FULL COMMITTEE MEETING
JUNE 6-8, 2002

June 6, 2002
Today's Date

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JUNE 6-8, 2002

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
493RD FULL COMMITTEE MEETING

JUNE 6-8, 2002

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493RD FULL COMMITTEE MEETING

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493rd FULL COMMITTEE MEETING
JUNE 6-8, 2002

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
493RD FULL COMMITTEE MEETING

JUNE 6-8, 2002

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FRED EMERSON	NEI
ROGER HUSTON	LSS
PAUL GUNTER	NIRS

ITEMS OF INTEREST

493RD ACRS MEETING

JUNE 6-8, 2002

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
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An Overview of the Nuclear Industry

Dr. Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission

2002 Nuclear Energy Assembly
Naples, Florida
May 3, 2002

Good morning. I am very pleased to be able to join you again this year at this annual meeting of U.S. and international nuclear energy industry leaders.

Much has changed since last year's Nuclear Energy Assembly, both for the Nuclear Regulatory Commission and for you in the nuclear industry. Obviously, the most striking change has arisen as a result of the events of September 11. This event has affected us in many ways, and will continue to do so for the foreseeable future. The protection of nuclear facilities from potential threats from terrorists has become one of our chief concerns. However, it would be a serious overstatement to say that physical security has been an "overriding" concern or a "preoccupation." Although the issue of security still requires much attention from the NRC and our licensees, safety issues more generally never go away for either the NRC or the industry.

The 103 operating nuclear power plants that supply almost 20% of our electricity continue to require careful regulatory oversight, both on a routine basis and to deal with emergent safety issues, as we have recently seen at Davis Besse. Our nuclear plant licensees continue to apply to renew their operating licenses beyond the original 40-year term and to increase the power output of their facilities. There continues to be interest in the possibility of new construction. And after decades of technical studies and political debate, we may be on the threshold of a new era in dealing with the accumulation of spent fuel at our power reactor sites. I would like to address some of these issues briefly in order to give you a sense of the NRC's perspectives.

Security

Last September's terrorist attacks brought home, in a horrific and graphic way, some of the vulnerabilities of our open society. The American people, and their representatives in Congress, have come to realize the damage that could be done to critical elements of our national infrastructure and they have asked for reassurance that public health and safety can be protected. My colleagues and I on the Commission believe that nuclear power plants are among the most hardened potential targets of terrorist attacks, but, in light of September 11, we have sought to reduce vulnerabilities. We recognize, however, that there are aspects of nuclear plant security and protection that must be within the purview of local, state or national government, and should not be the responsibility of licensees. In establishing interim compensatory measures for plant security and in our re-evaluation of the design basis threat, we are working to define where and how that line should be drawn. .

As you know, we have recently established a new Office of Nuclear Security and Incident Response. Our purpose was to pull together groups that were previously in our other offices to provide a single

focused organization for security, safeguards, and emergency response. This office also serves as a central point of contact for coordination between the NRC and other government agencies in these areas. Nuclear plant security cannot be viewed in isolation, but must be an integral part of an overall strategy for the protection of critical infrastructure. As a result, we continue to work with the Office of Homeland Security, other Executive Branch agencies, and Congress to formulate and implement strategies to deal with the terrorist threat. This is obviously an area that will continue to be of great importance to both the NRC and to you in the industry for the foreseeable future.

The Future of Nuclear Power

As I noted at the start of my remarks, security cannot be our preoccupation. Over the past decade, we have seen remarkable progress in the overall safety and economic performance of the nuclear power industry. As a result, we are seeing great interest from licensees in renewing their operating licenses beyond the original 40-year term. We also see continuing interest in the possibility of building a new generation of nuclear power plants. Let me address each of these areas briefly.

License renewal has been, from virtually every perspective, one of the NRC's most conspicuous success stories. We established the license renewal process to permit the staff to conduct a thorough and rigorous technical review to assure that a plant will be able to operate safely through the term of its extended license. The results of the license renewal program to date speak clearly of strong licensee interest.

To date, we have granted four renewals, covering 8 units. All of these reviews were carried out within the projected schedule: 25 months without a hearing, 30 months with one. Eight more plants, with 15 units, are currently under review. And the total number of plants that are formally in the renewal process - that is, either a renewal has issued, an application is under review, or the licensee has formally notified the NRC of its intent to submit an application - currently comprise roughly half of the U.S. operating fleet. We expect that almost all of the operating plants in the U.S. will ultimately apply to renew their licenses. As a result, we expect that nuclear power will continue to make a significant contribution to the Nation's electricity supply for at least the next several decades.

Future Reactors

If nuclear power is to contribute substantially to our energy supply beyond the next two to three decades, new construction will eventually be necessary. As you are well aware, it is not the NRC's job to promote such new construction, but if and when it comes about, we must be prepared to perform to ensure public health and safety in such a way that we provide a predictable, efficient, and effective process with a minimum of unnecessary burden on applicants and licensees. The NRC has established a process for licensing new plants, in Part 52 of our regulations, and we have put in place the organization that will manage new plant reviews. We believe we have the necessary framework in place. Although many of you are already somewhat familiar with Part 52, let me review its elements and indicate where we are in their implementation.

Part 52 is designed to provide a more streamlined licensing process than was the case under our "old" licensing regime, based on 10 CFR Part 50. It is built on three distinct elements: design certification, early site permits, and combined construction permits and operating licenses. The design certification process begins with NRC review of a standardized plant design. If the staff finds that the design meets technical licensing requirements, the design can then be certified through a rulemaking. In a similar fashion, Part 52 allows for approval of a prospective site in advance of a specific application to build a plant there. The review for an early site permit is similar to the site review performed under the old Part 50 process, but can be undertaken long in advance of a decision to undertake construction. If the review finds the site to be acceptable, the site can then be "banked" for a future application. An important feature of both design certification and the early site permit process is that technical issues resolved at these early stages cannot be reopened in later proceedings.

The final element of the Part 52 process involves combined licenses, or C-O-Ls in the NRC's terminology. A COL application may reference a certified design, or an early site permit, or both - although it is not required to do so. If the application does, in fact, reference a certified design or an early site permit, the scope of the review and any associated hearings would be limited to issues that could not previously be resolved. If the application is granted, an applicant may proceed directly from construction to operation without a separate application for an operating license. The

crucial factor in making this transition is the satisfactory completion of the inspections, tests, analyses and acceptance criteria–ITAAC–that demonstrate the constructed plant reflects the certified design.

We are currently reviewing Westinghouse's application for certification of its AP1000 design. If granted, this would be the fourth design certified by the NRC, after the ABWR, the System 80-plus, and the AP600. We expect to begin our first early site permit reviews in 2003 for Dominion's North Anna site, Entergy's Grand Gulf site, and an as-yet-unspecified Exelon site. And we are cognizant of the Department of Energy's 2010 initiative, which foresees the possible application for a COL perhaps as early as 2005. The NRC continues to prepare for these upcoming reviews, and to consider even more far-reaching changes to our licensing process to deal with non-light-water-reactor designs and to incorporate risk-insights. A technology-neutral, risk-informed regulatory approach is our ultimate goal.

You are all of course aware of the decision by Exelon to back away from the development of the pebble bed modular reactor. The PBMR has been very prominent in media reports discussing the construction of a potential new generation of nuclear plants and Exelon's decision may be seen by some to reflect a change in the attitudes toward new construction. The impact of Exelon's decision on the public has yet to be determined, but I do not detect any sweeping change in the overall industry interest in new reactors. The NRC's work on issues related to future reactors will continue so that we are prepared to fulfill our responsibilities in an efficient manner.

Yucca Mountain

Let me now move on to my final topic, the status of our national effort to dispose of spent fuel and high-level waste from civilian reactors.

As you know, just weeks ago, President Bush accepted the Secretary of Energy's recommendation that the Yucca Mountain site be developed as a potential repository for the disposal of high-level nuclear wastes and spent nuclear fuel. On April 8th, Governor Kenny Guinn of Nevada provided the Congress with the State's "Notice of Disapproval of the Proposed Yucca Mountain Project." The President's recommendation will become a final decision if, within 90 calendar days of continuous session, Congress approves a resolution of siting approval. Congress is currently holding hearings on the site recommendation and if it approves the President's recommendation, DOE is authorized to apply to the NRC for construction authorization.

We are preparing for the possibility of such an application. Specifically, as required under the Energy Policy Act of 1992, the NRC has promulgated the health and safety regulations to guide a licensing decision on Yucca Mountain. We are confident that any repository that can be shown by DOE to comply with these demanding regulations will protect the public today and in the future. In order to prepare for a potential license application review, the NRC staff recently published a draft of the Yucca Mountain Review Plan. Thus, NRC's regulatory framework for a review of a license application, should one be submitted, is largely in place.

If a license application is filed, I believe the litigation before the NRC will constitute one of the largest administrative proceedings in which any agency has ever engaged. To prepare for those proceedings, our Atomic Safety and Licensing Board Panel has begun work, including the exploration of possible arrangements for a hearing location and the development of automation tools necessary to facilitate a hearing. These activities include development of an electronic hearing docket and completion of the Licensing Support Network (LSN) that will provide access on the Internet to all the key documents. The NRC staff also is working to provide guidance to DOE on developing an electronic high level waste repository license application.

One issue that has been prominent in recent discussions of Yucca Mountain relates to the safety of the transportation of spent fuel. Federal regulation of spent fuel transportation safety is shared by the U.S. Department of Transportation (DOT) and the NRC. DOT regulates the transport of all hazardous materials, including spent fuel, and has established regulations for shippers and carriers regarding among other things, radiological controls, hazard communication, and training. For its part, NRC establishes design standards for the casks used to transport licensed spent fuel, and reviews and certifies cask designs prior to their use. NRC also conducts an inspection and enforcement program, and reviews and approves physical security plans for spent fuel shipments.

The safety record associated with the current regulatory system is exemplary – approximately

1,300 shipments of civilian fuel and 920,000 miles without an accidental radioactive release. Nonetheless, we continually examine the transportation safety program. Over two years ago, NRC began the Package Performance Study to study cask performance under severe impact and fire accident conditions. The study plan calls for full-scale testing of a cask to confirm computer models of cask response to severe accident conditions. As a part of its evaluation, the NRC staff is analyzing appropriate national transportation accidents, such as the 2001 train accident in Baltimore, Maryland, to determine if lessons learned from that event should be included in our transportation requirements or analyses. Finally, NRC is sponsoring a study to update its evaluation of cask response to acts of sabotage. NRC plans to utilize the results of these studies as input into its comprehensive review of security. These studies will provide further confidence that our national system for the transportation of spent fuel is safe long before shipments to a possible repository at Yucca Mountain.

Conclusion

I hope I have been able to provide a perspective on some of the challenges facing both the industry and the NRC in the coming years. There has been much that I have not had the time to cover: for example, our work on power uprates, continuing efforts to risk-inform our regulations, and progress with the reactor oversight process. We clearly have much on our plate. Nonetheless, speaking for myself, I believe the nuclear industry is in better shape, with more favorable prospects, than has been the case for more than 25 years. Speaking for the NRC, we look forward to fulfilling our appointed role: that of a tough but fair regulator that is fulfilling our primary mission of ensuring the health and safety of the public.

Thank you.

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PROVIDING CERTAINTY IN LOW-LEVEL RADIOACTIVE WASTE DISPOSAL: THE CONTINUING CHALLENGE

CHAIRMAN RICHARD A. MESERVE
U.S. NUCLEAR REGULATORY COMMISSION

KEYNOTE ADDRESS:
17th ANNUAL LOW-LEVEL RADIOACTIVE WASTE
DECISIONMAKERS' FORUM & TECHNICAL SYMPOSIUM
SCOTTSDALE, ARIZONA
MAY 14, 2002

Good morning. I am pleased to be here with you this morning to share my thoughts on the spectrum of issues facing nuclear licensees in their attempts to secure reliable options for disposal of low-level radioactive waste. The uncertainty as to the availability of safe and economical disposal options for low-level waste has become a key challenge for many categories of licensees.

Before discussing these issues, however, I would like to provide a status report on two related matters that could potentially affect low-level waste disposal activities. First, I will discuss the security measures that the NRC has taken in response to the terrorist attacks of September 11. And, second, I will provide a summary of recent activities related to the proposed high-level waste repository at Yucca Mountain, Nevada.

Security of Civilian Nuclear Activities

The terrible events of September 11 have had a profound impact on the NRC and its licensees. It has compelled all of us to reassess the full range of security issues involving NRC-related activities. This means, among other things, the examination of civilian uses of nuclear materials that have long been assumed to present minimal risks. As a result, the NRC has reassessed its assumptions about the nature of the threats with which our licensees might have to cope. In that regard, the events of September 11 have reinforced the fact that for all civilian nuclear facilities and activities, security concerns and safety are inextricably connected. Nuclear security is not and cannot be an afterthought or an add-on; rather, it is integral to all our activities, across the board.

In the immediate aftermath of September 11, we advised reactor and fuel cycle licensees to proceed to the highest level of security -- a posture they have maintained in the intervening months. Many state and local government officials also took steps to augment power plant security forces with law enforcement personnel or National Guard troops. The NRC has worked closely with the Office of Homeland Security, the FBI, and other government agencies to provide our licensees with timely information about potential security threats. It is my view that nuclear facilities provide the strongest defenses that are to be found at civilian infrastructure in this country.

Although the NRC's primary safeguards focus has been on those facilities that constitute the highest risk, such as civilian nuclear power plants and fuel cycle facilities, we also are increasing security for those nuclear materials that could theoretically be used as radiological dispersal devices or "RDDs." Although there has been significant media attention to the hazards of such devices, our

assessments indicate that RDDs are not particularly effective weapons in terms of the casualties that their use might be likely to cause. Large sources tend to be self-protecting and the dispersal of the material generally tends to reduce the hazard. RDDs are not very effective weapons, which is presumably the reason that such devices are not part of the arsenal of any country. Nonetheless, RDDs might be employed by a terrorist because of the fear that surrounds nuclear materials. Moreover, such a weapon could impose significant costs for cleanup and would obviously disrupt the use of the contaminated area until cleanup is accomplished. As a result, we have to assume that a terrorist might seek to use such a weapon.

The NRC has issued advisories to thousands of materials licensees and State agencies urging a number of specific actions to enhance the protection of nuclear materials. Generally speaking, these were directed to giving increased attention to unusual activities, ensuring conscientious use of security controls on radioactive materials, and restricting access during the use, storage, and transport of radioactive materials. The Commission is currently considering additional security enhancements as part of our comprehensive safeguards review.

The concern for security will result, over time, in a somewhat different focus in our regulation of nuclear materials. Our existing regulatory scheme is premised on ensuring the protection of the workers and the public from the licensed uses of radioactive materials, with little specific emphasis on preventing terrorist diversion and misuse of such materials. We clearly will have to build greater awareness that such materials could be used as weapons and to include elements in our regulations to reduce that threat.

Another area that we have had to reassess relates to the information that we make available to the public. September 11 made clear the need to rethink just how open we can and should be with respect to physical security issues. In assessing these issues, we must give due regard to two vital but competing interests. The first is the public's right to know, a right that is grounded in law and that is one of the most cherished principles of our democracy. The other is the need to keep sensitive information away from those whose purpose is to destroy that democracy. We are endeavoring to strike an appropriate balance between openness and security.

As many of you may already know, we initially brought down our website in order to give ourselves the opportunity to review its contents carefully. We needed to make sure that we were not inadvertently providing terrorists with information that could be valuable to them. In the intervening months, we have restored much of that information to public view. We realize that this has been an inconvenience to many who deal with the NRC, and we appreciate the patience and understanding that the agency has received in this critical period.

Status of Activities Related to High-Level Waste Disposal

Let me now briefly discuss recent events related to the proposal to site a high-level waste repository at Yucca Mountain.

As you know, several months ago, President Bush formally accepted the Secretary of Energy's recommendation that the Yucca Mountain site be developed as a repository for the disposal of high-level nuclear wastes and spent nuclear fuel. On April 8th of this year, the Governor of Nevada provided the Congress with the State's "Notice of Disapproval of the Proposed Yucca Mountain Project." Under the law, the President's determination will become a final decision if, within 90 calendar days of continuous session, Congress passes a resolution approving it. The House took such action last week and hearings on the site recommendation are scheduled in the Senate. If Congress endorses the President's approach, DOE will be authorized to apply to the NRC for a license to construct the repository.

Of course, we cannot know with certainty what the outcome of the Congressional consideration will be. Nonetheless, the NRC needs to be prepared for the possibility of such an application. Specifically, as required under the Energy Policy Act of 1992, the NRC has promulgated the health and safety regulations to guide a licensing decision on Yucca Mountain. These are demanding regulations, and we are confident that any proposed repository that can be shown by DOE to comply with them will adequately protect the public, now and in the future. In addition, in order to prepare for a potential license application review, the NRC staff recently published a draft of the Yucca Mountain Review Plan. Thus, NRC's regulatory framework for a review of a license application, should one be submitted, is largely in place.

If an application is submitted, the administrative proceeding before the NRC is likely to be extensive and complex. Our Atomic Safety and Licensing Board Panel has already begun work to prepare the ground, including development of the electronic tools that will make it possible for the parties to have access to the enormous volume of documents that can be expected. We plan to harness electronic technology to replace traditional document discovery and to expedite the hearing.

One issue that has been prominent in recent discussions of Yucca Mountain relates to the safety of the transport of spent fuel. Federal regulation of spent fuel transportation is shared by the U.S. Department of Transportation (DOT) and the NRC. DOT regulates the transport of all hazardous materials, including spent fuel, and has established regulations for shippers and carriers regarding, among other things, radiological controls, hazard communication, and training. For its part, NRC establishes design standards for the casks used to transport licensed spent fuel, and reviews and certifies cask designs prior to their use.
NRC also conducts an inspection and enforcement program, and reviews and approves physical security plans for spent fuel shipments.

The safety record associated with the current regulatory system is exemplary – approximately 1,300 shipments of civilian fuel and 920,000 miles without an accidental radioactive release. But, as elsewhere in our activities, a record of success does not preclude the possibility that undetected weaknesses may exist, and neither the NRC nor its licensees can afford to become complacent. We therefore continually examine the transportation safety program. Over two years ago, NRC began the Package Performance Study to study cask performance under severe impact and fire accident conditions. The study plan calls for full-scale testing of a cask to confirm computer models of cask response to severe accident conditions. As a part of its evaluation, the NRC staff is analyzing appropriate national transportation accidents, such as the 2001 train accident in Baltimore, to determine if our transportation requirements need to be modified. Finally, NRC is sponsoring a study to update its evaluation of cask response to acts of sabotage. These studies, together with any resulting changes to our security requirements, if necessary, should further ensure the safety of the transportation of spent fuel.

Low-Level Waste Disposal

I will now turn to the central focus of this meeting -- low-level radioactive waste disposal. It will not be news to any one here that the low-level waste siting program in this country is not working. Moreover, barring Congressional action, which is unlikely in the near term, the situation is unlikely to change. Access to low-level waste disposal sites affects many classes of licensees, including nuclear power plant licensees intending to decommission their plants. My remarks today will concentrate on the strategies that the NRC is considering to make sure the uncertainty in obtaining uninterrupted access to licensed low-level waste disposal sites does not adversely affect licensees' ability to decommission their plants safely.

Before I begin, let me say that the NRC staff will make a number of detailed presentations at this meeting on some of these topics, so I will discuss them only briefly.

It has been estimated that the volume of low-level radioactive waste generated over 40 years of operation is approximately 600 cubic meters for a pressurized water reactor and is approximately 2700 cubic meters for a boiling water reactor.[1] Thus, the total volume of low-level waste generated by existing reactors is on the order of 150,000 cubic meters. If decommissioning is also considered, however, the estimates are 8,000 cubic meters of waste for a pressurized water reactor and 15,000 cubic meters for a boiling water reactor.[2] The resulting total volume is on the order of a million cubic meters. Volumes of contaminated soil at fuel cycle sites and sites managed under NRC's Site Decommissioning Management Plan are less well known, but could be on the order of 650,000 cubic meters. Consequently, although the estimates carry considerable uncertainty, it is clear that there is a significant volume of low-level waste for which a disposal site or sites will be needed.

Sufficient disposal capacity currently exists to handle today's disposal needs, particularly in light of the trend towards license renewal of civilian nuclear power plants. (License renewal delays decommissioning and hence postpones the need to dispose of the waste associated with decommissioning.) In addition, waste minimization, volume reduction, and decay-in-place strategies reduce the overall volume of material. Nonetheless, the disposal situation is increasingly uncertain. With the eventual closure of the Barnwell disposal facility to states outside the Atlantic Compact, the absence of progress in other Compacts to site low-level waste disposal facilities, and few other disposal options, access to facilities for the disposal of low-level waste is increasingly

constrained. Although Envirocare of Utah may eventually obtain state approval for disposal of Class B and C wastes, the limited options for disposal are likely to keep disposal costs high. There is thus the potential that the decommissioning process for many sites and the medical use of radionuclides will be affected adversely.[3]

Even today, the reduced availability and the associated high cost of disposal at licensed low-level waste disposal sites are fostering exploration of a series of alternatives (e.g., assured isolation, rubblization, and entombment). However, all these alternatives face considerable uncertainty.

Moreover, the situation is aggravated by the reality that one important aspect of the license termination rule -- restricted release -- is not functioning as intended. You will recall that the LTR allows, under certain circumstances, license termination with restrictions on future site use, so long as there is an appropriate third- party custodian, such as a governmental entity, that will maintain the site in perpetuity and there are legally enforceable institutional controls that limit exposure to 25 mrem for an average member of the critical group. Those NRC licensees interested in restricted release are finding it extremely difficult to find an appropriate custodian. Thus, it now seems that the broad flexibility that the NRC's license termination rule was intended to provide is not being realized. As a result, licensees with sites to be decommissioned are being forced to pursue unrestricted release, which means increased volumes that must be shipped to offsite waste disposal sites. This, of course, simply serves to enhance the demand for increasingly unavailable low-level disposal sites.

The Commission is aware of and concerned by these developments. In November 2001, I sent a letter to Secretary Abraham of the Department of Energy noting that Section 151(b) of the Nuclear Waste Policy Act allows the Department to assume title and custody of low-level radioactive waste and the land on which such waste is disposed of, provided the NRC makes certain findings. We advised that NRC has identified a number of sites (currently about seven) that might be appropriate for future license termination under restricted conditions and that could be candidates for the exercise of DOE's statutory authority for long-term stewardship. I encouraged DOE to continue to work with NRC staff to develop a Memorandum of Understanding that addresses issues associated with long-term stewardship. In January 2002 Undersecretary Card responded to my letter by recommending that DOE and NRC work together with an appropriate Federal land management agency such as the Department of the Interior, as well as with the Office of Management and Budget, to seek a solution. DOE and NRC staff are preparing to begin discussions with OMB and DOI. In the meantime, it has become apparent that the Commission must also explore other ways to make restricted release a viable decommissioning option under the license termination rule.

The Commission is also taking other steps that may address, to some extent, the lack of disposal options. The NRC has reaffirmed that uranium mills can be used for processing alternative feed material, such as material from the Corps of Engineers Formerly Utilized Sites Remedial Action Program. Similarly, the Commission has voted to approve the use of mill tailings impoundments for disposal of other radioactive wastes similar to mill tailings, so long as the approvals of the long-term custodian and the relevant Low-Level Waste Compact are received prior to disposal. Also, because mill tailings impoundments and RCRA Subtitle C facilities can provide similar levels of public health and safety protection, we are beginning to work with EPA to consider a rule that would allow for disposal of certain radioactive material in RCRA impoundments. Finally, the NRC has agreed to help fund an assessment by the National Academies of the key issues affecting the safe and cost-effective management of civilian low-level waste. Many of these topics will be discussed in detail by other presenters at this meeting.

One further option that the NRC is exploring in its search for safe and otherwise suitable approaches to decommissioning is the unrestricted release of slightly contaminated material that poses no significant health threat. If this could be accomplished, it would serve to preserve scarce space in low-level waste sites. As many of you may recall, in June 1999, the NRC published an issues paper in the Federal Register to foster discussion about alternative courses of action for control of slightly contaminated materials. We indicated an interest in exploring options to address the issues, including whether to establish a standard for release by rule. In addition, the NRC staff held a series of public meetings during the fall of 1999 at four locations around the country to provide further opportunity for public input.

Over 800 written comments were received. Potential recipients of solid material, such as scrap metals and cement industry representatives, strongly objected to the release of contaminated solid materials. These commenters asserted that there would be severe economic impact on their

industries if consumers were to refuse to buy their products because of concerns over the presence of radioactivity. Citizen groups and other individuals expressed concern about health effects of the potential presence of this material in the environment and many argued that NRC should prohibit the release of any contaminated material. Others strongly favored the establishment of clearance levels.

The number and intensity of these comments indicated that further careful examination of the issue was needed. Moreover, it was clear from the comments that there was an element of distrust among some stakeholders as to the objectivity of the NRC in these matters. As a result, the Commission concluded that any successful resolution of the debate over the unrestricted release of slightly contaminated material would be facilitated by an independent assessment. Accordingly, the NRC deferred action so that the National Academies could study the issue and make recommendations to us. In the interim, the NRC and Agreement State staff would continue to handle releases on a case-by-case basis.

The NAS study is now complete and the NRC staff is currently evaluating the report and its recommendations.[4] The report finds that there is no immediate need for resolution of the issue, but that steady progress toward resolution should be undertaken. Although urging further careful engagement with stakeholders, the Academies clearly see conditional release -- that is release of slightly contaminated materials to unlicensed individuals for limited uses -- as an option that is worthy of careful exploration. I am very much looking forward to receiving the staff's analysis of the report and its recommendations for further steps on the part of the Commission. The staff's analysis is due to the Commission within a few months.

Conclusion

Let me note in closing that the Commission fully appreciates the importance of bringing greater clarity and predictability to the issue of access to low-level waste disposal for both current and future needs. The challenges are complex. Technology, science, economics, law, and public opinion all play a part. While none of us expects quick or easy answers, there is a need to bring greater certainty to low-level waste disposal. The Commission is committed to this task.

Thank you.

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[1]C.C. Miller, "Radwaste Management at U.S. Nuclear Power Plants: Where We Are Today (and How We Got There),", Radwaste Magazine, Volume 6, Number 6, at 8, November/December 1999.

[2]G. Konzek and others, "Revised Analysis of Decommissioning for the Reference Pressurized Water Reactor Power Station," NUREG/CR-5884, November 1995, and R. Smith and others, "Revised Analyses of Decommissioning for the Reference Boiling Water Reactor Power Station," NUREG/CR-6174, July 1996.

[3]National Research Council, Board on Radiation Effects Research, "The Impact of Low-Level Radioactive Waste Management Policy on Biomedical Research in the United States" 2001.

[4]National Research Council, Board on Energy and Environmental Systems, "The Disposition Dilemma: Controlling the Release of Solid Materials from Nuclear Regulatory Commission-Licensed Facilities," March 2002.

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U.S. PRESENTATION TO THE CONVENTION ON NUCLEAR SAFETY OF THE SECOND REVIEW MEETING OF THE CONVENTION ON NUCLEAR SAFETY

Dr. Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission

April 18, 2002
Vienna, Austria

Good morning, Mr. Chairman, ladies and gentlemen. I am pleased to join you today as the representative of the United States.

This convention presents a unique opportunity for representatives from more than 50 countries that have a common interest in improving nuclear safety to benefit by sharing our different approaches and experiences. In that spirit of cooperation, it is a pleasure for me to present the United States' National Report for your review and to respond to your questions.

I would like to begin by making some general comments before discussing the substance of the United States' National Report.

Current State of the Nuclear Industry In the United States of America

The nuclear industry in the United States is healthier than it has been for many years. Economic performance is strong, which is reflected in the fact that the production cost for nuclear power is less on average than that for electricity produced from coal or natural gas - its direct competitors. As a result, there is a strong interest in maintaining or even expanding nuclear power production. Even more important in this context, safety performance has improved in parallel with economic performance. An objective measure of this improvement is illustrated by the NRC's systematic tracking of significant events, which include failures of safety systems, unanticipated plant responses, degradation of key systems or components, and operator errors. Over the past 15 years, the number of significant events has dramatically declined. Other performance indicators, such as automatic scrams while critical, safety system actuations, and collective radiation exposure to plant personnel, have shown substantial improvement as well.

Perhaps as a consequence of this improved economic and safety performance, the nuclear industry in the United States appears to be entering a renaissance. After 25 years in which there have been no new reactor orders, several electric generating companies have expressed interest in building new plants. In addition, the increasing need for additional power and the improved performance of nuclear power plants over the past decade have caused an increasing number of licensees to apply for power uprates and to consider renewing their licenses instead of decommissioning their plants.

In general, our best-performing licensees also tend to be the most economically successful. This is not unexpected because improvement in both safety and economic performance stem from the same causes: an attention to detail, rigorous preventative maintenance, effective training, and the

maintenance of a strong safety culture. I believe as well that the NRC's efforts to establish a stable, predictable regulatory regime have played a role in establishing a climate that has encouraged the current levels of safety and economic performance.

The United States' Commitment to Continuous Learning and Improvement

We recognize, however, that complacency opens the door to decline. In the nuclear business, if you are not steadily struggling to improve, you will find that you are sliding backwards. Consequently, the NRC and our licensees are committed to continuous learning and improvement.

In recent weeks, the United States had a powerful reminder of the need for unremitting attentiveness to safety issues. I am referring to the discovery of corrosion damage to the reactor vessel head at the Davis Besse Nuclear Power Station. Most of you are probably familiar with the underlying facts, so I will describe them here only briefly.

The Davis Besse Nuclear Power Station shut down to conduct a refueling outage in February of this year. In response to an NRC Bulletin issued last August concerning the circumferential cracking of reactor pressure vessel head penetration nozzles, the licensee had committed to inspect the nozzles in the course of the outage. The particular focus was to be on the nozzles associated with the control rod drive mechanisms (CRDMs). The licensee found indications of cracking in the nozzles for five CRDMs and committed to repairing them. In the course of performing repairs, however, the licensee encountered anomalies that caused it to investigate the condition of the pressure vessel head. When the licensee removed boric acid deposits from the top of the head they found a large cavity encompassing an area of five inches by seven inches. The wastage had progressed all the way through the 6.5 inch carbon steel base material to the 0.3 inch stainless steel interior cladding. This discovery clearly constituted a serious degraded condition. And, as we have examined the matter further, it has become apparent that there were indications of the problem that the licensee failed to recognize. We will be looking at this systematically to determine if this problem represents a weakness in our oversight program or a failure by the licensee to take appropriate measures or both.

Although the United States believes that it has a comprehensive program for nuclear safety, the Davis Besse event reinforces the need to remain watchful. We cannot fall into the trap of assuming that safety is necessarily assured even with improved technology, good performance, and strong regulatory oversight. Continuous vigilance to consider what may have been missed is also necessary.

The Convention on Nuclear Safety is an excellent forum to encourage such vigilance and to learn from the experiences of others. We see this meeting as an opportunity to generate ideas and to share international experiences. It will also inform and enhance our collective efforts to improve worldwide safety. We all benefit because an accident anywhere in the world affects us all. Consequently, my fellow Commissioners and I have given the Convention a high priority. We fully support the Convention's goal of enhancing nuclear safety worldwide through national measures and international cooperation.

With that as a backdrop, let me now turn to the U.S. National Report. In preparing our report, we considered each of the Convention's obligations and focused on explaining our means of fulfilling those obligations. Each of the chapters relates to an article of the Convention and describes how the U.S. seeks to fulfill the obligations arising from this article. The United States also received some questions concerning our report. We have also sought to respond to each of your questions. Our written responses are provided in a supplement to our National Report, which has been made available to you. Additionally, we have prepared a CD-ROM containing certain reference documents and other information that may be of interest.

As you may know, this is the first time the United States is participating in a Review Meeting. Nonetheless, in preparing our National Report, we did consider the issues that were raised in the final report of the First Review Meeting.

SCOPE OF TODAY'S PRESENTATION

Given the number and depth of the questions that were presented to the United States, I believe it is apparent that most of you are very familiar with our report. Indeed, it has become apparent to me over the time that I have been at the Commission that many of our international colleagues

have an impressive knowledge of the U.S. regulatory program. In order to preserve ample time for a dialogue, I will not consume your valuable time by repeating information we have presented in our report. Instead, I believe it will be more meaningful and productive to focus on certain general themes in two broad areas that emerged from the questions you posed. Then I invite an extended discussion of these themes or of other matters of interest.

The broad areas that I will discuss are:

Nuclear Reactor Regulation at the NRC today, which includes:

- risk-informed regulation;
- reactor oversight process;
- safety culture; and
- safety reviews

and recent initiatives, which include

- license renewal; and
- new reactor licensing

Following my discussion of these themes, I welcome the opportunity to clarify aspects of the United States' report in response to your questions.

RISK-INFORMED REGULATION

Overview of Risk-Informed Regulation

Let me focus first on our agency's initiatives concerning a risk-informed approach to regulation.

The evolution to a more risk-informed approach to regulation is perhaps the most significant change occurring at the NRC today and is a theme central to the NRC's activities. This effort represents a significant shift away from our traditional approach.

Our historical regulatory framework is based on a "deterministic" approach that rests in part on a defense-in-depth philosophy, and employs conservative safety margins, accident analyses with prescribed acceptance criteria, and qualitative assessments of risk. We believe this approach has served us well. Nonetheless, there are modern analytical tools that should be more systematically applied. We now have more than 25 years of progress in the development of probabilistic risk assessment (PRA), stemming from the Rasmussen Report of 1975, and over 40 years of operating experience from which to develop input data on equipment reliability. A PRA provides a direct measure of safety and is connected to the fundamental purpose of our regulatory system. As a result, the NRC adopted a policy to promote the increased use of PRAs, to the extent practical.

I should emphasize, however, that our aim is to use risk insights to *complement* the existing deterministic approach. We take this incremental approach in recognition of the uncertainties in PRA analysis and the reality that we cannot impose a wholly new regulatory system to operating plants. This complementary aspect explains why the NRC refers to its actions as being "risk-informed" and not "risk-based." We do not intend to jettison the existing regulatory system, but instead to use risk insights as a tool for its modification and improvement.

We are introducing this new effort carefully and deliberately. As we have developed risk-informed regulations and regulatory practices, we have invited the public to comment so that we can gain the benefit of a broad range of input from outside the NRC. We hold public workshops and meetings with stakeholders as we consider risk-informed improvements. And as we evolve toward a risk-informed regulatory system, we will continue to require informed input from the nuclear industry and from other stakeholders, both within the United States and internationally.

Benefits of Risk-Informed Regulation

This group asked many questions about our risk-informed approach to regulation. For example, you asked what benefits the NRC has realized, how we measure them, and how many licensees are taking advantage of those benefits.

The NRC has initiated several risk-informed changes to our regulatory programs. We believe those changes have numerous benefits:

- They focus attention on the areas of highest safety priority and result in more realistic regulatory decisions. Risk insights can cut both ways -- justifying increased regulatory requirements in some cases and reductions in others. Let me emphasize a fundamental point: the elimination of regulatory requirements that do not affect safety can itself improve safety by encouraging increased attention to those requirements that are important. Thus both the reduction of requirements and the addition of requirements on the basis of risk considerations serve to enhance safety overall.

- Risk-informed reform enables the reduction of unnecessary regulatory burden. For example, recent risk-informed initiatives concerning in-service inspection and testing have allowed licensees to focus their resources on highly risk-significant systems and components, while systems and components that are less risk-significant receive less attention, consistent with their lower safety influence. Similarly, the improved standard technical specifications reduce the regulatory burden on both the licensee and the regulator without adverse risk impacts by generally allowing more appropriate surveillance testing and longer times to correct problems before requiring a plant to change modes. These allowances help to reduce the number of unnecessary scrams, power reductions, and plant shutdowns. Ultimately, these activities serve both to reduce needless cost and to increase safety.

- Risk-informed initiatives improve communication among the NRC, the nuclear industry, and the public. The careful consideration of risk enables the systematic and principled examination of the foundations for regulatory action. This enhances public acceptance because the reasons for and benefits of regulatory change are more transparent.

In most cases, risk-informed changes to regulations and regulatory practices are voluntary. That is, licensees may continue to apply existing deterministic criteria to show compliance with NRC requirements, or may voluntarily switch to alternative, risk-informed processes. (Of course, if risk insights reveal a significant weakness in the existing regulatory standards, the deterministic criteria will be amended.) The value of the new approach is best demonstrated by the number of licensees who have chosen to implement voluntary risk-informed alternative approaches to regulatory requirements. Risk-informed in-service inspection has been implemented at 43 units, risk-informed changes to technical specifications concerning allowed outage times have been implemented at 41 units, and standard technical specifications have been implemented at 63 units. The United States currently has 103 operating commercial reactors.

Challenges in Implementing Risk-Informed Regulation

In taking the first steps in risk-informing our regulatory system, we expected challenges along the way and we have encountered many. The shift from a traditional prescriptive, deterministic approach toward a risk-informed approach has challenged both the NRC and the regulated industry because the new approach requires rethinking the foundations of the entire regulatory structure. Moreover, our regulatory requirements contain intricate interconnections, so all of the implications of change must be carefully evaluated.

In order to guide and inform the industry, the public, and our own staff on how best to use risk information, the NRC has developed many guidance documents addressing various aspects of risk-informed regulatory methods. Several of these documents are included on the CD we have made available.[1] In addition, the NRC developed a Risk-Informed Regulation Implementation Plan (RIRIP) which is updated regularly to reflect both progress and new challenges. Our aim through this guidance is to manage these activities more efficiently in order to achieve results more quickly.

Let me provide an example of an application of risk insights - our revised maintenance rule. The rule added new requirements to:

- Verify the adequacy of risk assessments for the existing plant configuration;
- Consistently perform risk assessments before conducting maintenance activities;
- Ensure the availability of key safety functions through the use of risk assessment tools; and
- Identify and implement appropriate risk management activities.

The application of the rule presents challenges in that it requires an in-depth understanding of plant

equipment functions and interdependencies. But the rule provided a way to assure safety while a plant, as a result of maintenance activities, was in an abnormal configuration. And the assessment of risk associated with the maintenance activities has offered licensees a mechanism by which they may show in some cases that on-line maintenance is safer than maintenance during shutdown, thereby enabling both improved safety and efficiency. I am happy to report that industry has embraced the requirements of the rule and is seeking to use the methodologies for risk management developed in connection with the maintenance rule in other regulatory areas, such as in connection with the development of risk-informed technical specifications.

Various other risk-informed rule changes are underway. These include possible changes to our rules governing special treatment requirements (the special requirements governing safety-related equipment; 10 CFR 50.69, passim), combustible gas control (50.44), emergency core cooling systems (50.46), and pressurized thermal shock (50.61). Although the efforts to use risk insights to revise our regulatory system have proceeded somewhat more slowly than we initially anticipated, we continue to believe the potential gains in developing a more consistent and rational regulatory structure are worth pursuing.

REACTOR OVERSIGHT PROCESS

Overview of the Reactor Oversight Process

The most visible aspect of our efforts to apply a risk-informed philosophy to our regulatory system is our *reactor oversight process* (or "ROP"). The NRC developed the ROP to focus the inspection of operating plants on areas involving the greatest risk, while making our oversight of the nuclear industry more objective and transparent. A number of the documents associated with the development and implementation of the ROP are included on the CD that we have made available.[2] During my discussion, I will cover the following aspects of the NRC's Reactor Oversight Process:

- Implementation and resource requirements;
- Inspection program;
- Use of performance indicators; and
- Ongoing development and evaluation

We realized that, despite our successes in regulating and improving the performance of U.S. nuclear power plants over the past 40 years, our inspection, assessment, and enforcement processes did not always focus on the most important safety issues. In some situations, our inspection activities were inefficient and, at times, they were overly subjective. In addition, our regulatory actions were not always sufficiently understandable or predictable to either the public or the regulated industry.

The reactor oversight process addresses these concerns by defining several regulatory objectives:

- Focus inspections on activities involving the greatest potential risks;
- Devote greater regulatory attention to facilities with performance problems;
- Use objective measurements of plant performance;
- Give all stakeholders timely and understandable assessments of plant performance;
- Minimize unnecessary regulatory burdens on nuclear power plants; and
- Respond to violations of regulatory requirements in a predictable manner, consistent with their risk and safety impact.

The key features of the ROP are new methods for assessing and reporting performance and for conducting inspections to ensure safe operation. The process also clearly spells out what licensees can expect if they achieve good performance, as well as what actions the agency will take if performance declines.

The ROP is anchored in the NRC's fundamental mission to ensure public health and safety in the operation of nuclear power plants. Its objective is to monitor performance in three areas that are important to safety - reactor safety, radiation safety (by which we mean exposures in normal operations), and plant security - each of which is associated with one or more "cornerstones" of safe nuclear plant operation. These cornerstones are the fundamental building blocks of the ROP.

In addition, there are three "crosscutting elements" that apply to all areas of safety, and thus to all of the cornerstones. These elements are:

- Human performance
- A safety-conscious work environment to ensure management attention to safety and to protect the ability of workers to raise safety issues; and
- Corrective action programs to identify and fix problems.

These aspects of the ROP are shown graphically on the slide.

Implementation and Required Resources

I want to briefly touch on how the ROP has been implemented and how much staff effort has been required.

In developing the ROP, the NRC sought input from the nuclear industry, citizens groups, and the general public. We held public workshops to obtain feedback from interested stakeholders while we were developing the process and establishing the pilot program so that we could gain the benefit of stakeholder views before implementation. We continue to seek public input as we gain experience with the ROP and are using insights from experience to further revise the program.

The assessment of licensee performance in the ROP rests upon two major elements: data from performance indicators and results of inspections. The results are reported using a color-coded system that reflects the safety significance of the findings. "Green" indicates that the findings present very low risk significance. Findings characterized as white, yellow or red reflect the increasing levels of risk significance. With this as a backdrop, let me describe the inspection program and performance indicators in more detail and show how the NRC's actions are keyed to the assessment results.

Inspection Program

The inspection program includes a baseline effort and supplemental inspections that may be added for a variety of reasons. The baseline inspection effort reflects the required minimum level of inspection for a licensee. It is designed to assess licensee performance in areas in which performance indicators are not available and to verify the validity of the indicators. About 15% to 20% of the NRC's baseline inspection program is devoted to ensuring that licensees have effective programs for identifying and resolving deficiencies. The baseline inspection program requires a total of about 5,000 inspector hours per year for a two-unit site. In addition to the baseline program, licensees may be subject to additional inspections for any baseline inspection findings that are not color-coded as "green." Such inspections range from less than 30 hours to as much as 2,000 hours, depending upon the significance and complexity of the issue. Inspections are also conducted in response to events and to assess the resolution of generic safety issues, with the level of effort commensurate with the risk significance. Thus, the inspection effort for a two-unit site can be significantly more than 5000 inspector hours if performance indicators, inspection findings, or significant operational events cross established thresholds.

Use of Performance Indicators

Licensee performance is also assessed using information from performance indicators. For example, typical performance indicators are based on the number of unplanned reactor scrams, safety system unavailability, or effluent releases. Performance indicators use objective data to provide information on licensee performance in each of the cornerstones. Our licensees generate performance indicator data for submission to the NRC, and the NRC verifies the accuracy and completeness of the data as part of the baseline inspection program.

We are continuing the search for appropriate performance indicators. Ideally, of course, such indicators should serve as a means to identify emerging safety problems early, rather than merely to confirm the existence of a problem. Moreover, the development of a broader suite of indicators is desirable because the indicators cover only part of plant performance. The limited scope of performance indicators is part of the reason for baseline inspections.

Allow me to note in passing that Article 19 of the Convention requires the Contracting Parties to take appropriate steps to ensure that their programs promote the collection, analysis, and communication of operating experience. We believe that one effective and objective method for communicating and sharing international operating experience is the use of some form of common

performance indicators. Such indicators could provide a useful point of reference for evaluation of the national reports.

The Action Matrix

The ROP also specifies the regulatory action that follows from performance indicators and inspection findings. To do this, we developed an "action matrix"[3] to ensure that our regulatory response to declining licensee performance is applied in a consistent fashion. The decisions are guided by establishment of thresholds that enable the performance indicators and inspection results to be addressed in a consistent way across the fleet of plants. For example, if a single performance indicator or cornerstone inspection area crosses its threshold, the NRC will consider a supplemental inspection to ensure that the licensee has identified the root causes and specified appropriate corrective actions. More significant changes in performance could lead to more significant actions. The last performance band, "Unacceptable Performance," is reserved for plants at which the degradation in performance is so serious and pervasive that continued operation of such plants would threaten our ability to ensure reasonable protection of public health and safety.

Our premise is that licensees are responsible for ensuring the safety of their facilities, and our thresholds help to determine the level of regulatory engagement that is appropriate in each cornerstone. Our intention is that enhanced regulatory oversight will prod the correction of problems before an incident occurs. Nonetheless, past experience suggests that we can still expect a limited number of risk-significant events to continue to occur with little or no prior indication of overall declining performance. If such events do occur, we will conduct follow-up inspections to ensure that the causes of these events are understood and that licensee corrective actions are adequate to prevent recurrence. Similarly, if we receive allegations from licensee employees or public stakeholders, we may conduct follow-up inspections. We will then factor the results of such inspections into our assessment process, along with performance indicator data and the results of risk-informed baseline inspections.

Ongoing Development and Evaluation

On the basis of lessons learned and feedback from stakeholders, we are confident that the ROP has met the goal of being more objective, risk-informed, understandable, and predictable than our previous process. Nonetheless, we realize that we will have to continue to assess and modify the ROP, and we have established a self-assessment program to identify areas for improvement.

As we have made the transition from deterministic oversight methods to the ROP, we confronted a variety of issues, some of which we have resolved and others we are still addressing. One significant issue relates to the assessment of the risk-significance of inspection findings. Since some inspection findings cannot be analyzed using PRA tools -- such as those in the areas of security, radiation protection, and emergency preparedness -- the NRC has to develop and refine a process to assess the significance of the inspection findings. We are also still working to address the standardization of performance indicator definitions, minimizing the unintended consequences of the performance indicators, and establishing thresholds for documenting inspection findings. The Advisory Committee on Reactor Safeguards, an independent group of technical experts providing advice and guidance to the Commission, has been of considerable assistance in these efforts.

We recognize that the reactor oversight process that we are implementing in the US would be difficult to apply uniformly throughout the world. It works for us, in part, because we have a mature industry with more than 100 units that have collectively accumulated several thousand reactor-years of operating experience. That experience provides a sound basis for risk-informing our regulatory processes. We recognize that this approach may not be appropriate for all.

ASSESSMENT OF SAFETY CULTURE

Overview of Safety Culture

Let me turn now to safety culture.

I mentioned earlier that, in general, the best-performing licensees from a safety standpoint also tend to be the most economically successful. Conversely, breakdowns in economic and safety performance can often be traced to failures in safety culture. These facts suggest that a strong safety culture is a singularly important attribute of a licensee's organization.

Although safety culture is a broad concept, there is general agreement as to its basic elements. These include licensee emphasis on safety as the highest priority; training for all staff (at all levels) to ensure that each employee understands his or her responsibilities for ensuring safe operations; conservative, safety-conscious decision making; a philosophy of continuous improvement, including critical self-assessment and a questioning attitude; and a willingness to address promptly and effectively any problems that may arise.

How the NRC Assesses Safety Culture

I believe that the United States explicitly or implicitly addresses most of the elements of safety culture in the NRC's regulatory processes, despite the fact that we do not directly regulate safety culture. We believe that it is unnecessary to assess a licensee's safety culture as a distinct component because the concept of safety culture is similar, if not integral, to the licensee's more specific responsibilities. If a licensee has a poor safety culture, problems and events will continue to occur at that facility either causing various performance indicators to exceed their thresholds, or surfacing during the NRC's baseline inspection activities. In fact, the baseline inspection program specifically includes reviews of a licensee's programs to identify and correct problems, and also verifies that the licensee has properly implemented the Maintenance Rule, which ensures effective resolution of deficiencies that involve risk-significant systems and components. This inspection effort assesses a central aspect of safety culture -- the willingness to identify and correct problems. If necessary, the NRC can also assess the work environment at a licensee's facilities and require a licensee to conduct an independent or third-party survey of its safety-conscious work environment. Indeed, the health of the licensee's safety culture will be indicated by performance indicators or baseline inspections, and if necessary can be evaluated by special inspections.

Other processes, programs, and regulatory requirements also help provide the NRC with a basis by which to assess the health of a licensee's safety culture. Specific examples include requirements for a quality assurance program and for allegations management.

Our quality assurance (QA) regulatory requirements, as codified in Appendix B to 10 CFR Part 50, define 18 criteria, many of which are related to the elements of safety culture that I mentioned earlier. These include management involvement; training; prompt and effective corrective action; and critical self-assessment by means of an audit program. Rigorous implementation of an effective QA program helps to foster the development of a healthy safety culture.

Licensees are also required to deal effectively and conscientiously with issues and allegations raised by their employees or contractors. This responsibility promotes the questioning attitude and critical self-assessment that contribute to a healthy safety culture. If plant employees believe that their concerns are not being effectively addressed by licensee management, they can bring these issues to the NRC for further investigation. We believe that such an allegations program is an essential factor in nurturing nuclear safety as specified by Article 10 of the Convention. Indeed, we believe that future Conventions should urge Contracting Parties to report on how they ensure the open identification and discussion of safety issues without fear of retribution.

In sum, we believe that the NRC has the benefit of a variety of tools to probe the existence of an appropriate safety culture among our licensees.

HOW THE NRC FULFILLS PERIODIC SAFETY REVIEW OBJECTIVES

Let me turn now to periodic safety reviews. Most countries have decided to perform periodic safety reviews in accordance with the recommendations of the International Atomic Energy Agency (IAEA). The objective of these reviews is to ensure a high level of safety throughout the life of a nuclear power plant, given the cumulative effects of plant aging, modifications, operating experience, and technical developments.

The NRC agrees that vigilant oversight and regular review are essential to ensure a high level of safety throughout the life of a nuclear power plant. Although a number of our regulatory review processes, specifically our reactor oversight process, are periodic in nature, our regulatory environment calls for continuous oversight. We believe that the United States meets the obligation to provide safety assessments throughout a nuclear installation's life through our ongoing reviews and through upgrades to our requirements. Let me discuss these processes further.

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Prior to issuing the original operating license for a nuclear power plant, the NRC performs comprehensive reviews to ensure that the proposed design, construction, and operation of the plant meet the NRC's requirements and provide reasonable assurance of adequate protection of the health and safety of the public. After issuance of the license, the licensing basis of a plant does not remain fixed. Rather, the licensing basis evolves throughout the term of the license through the NRC's ongoing regulatory activities, as well as the activities of the licensee.

The NRC's activities that provide ongoing assurance of an acceptable level of safety include daily oversight by the resident inspectors, periodic regional inspections, audits, investigations, evaluations of operating experience, independent research, and regulatory actions to resolve identified safety issues. The NRC evaluates new information and determines if changes to the licensing basis are warranted from a safety perspective by applying certain regulatory criteria (termed the "Backfit Rule"). These processes continue through each plant's operating life.

A licensee may also request changes to the licensing basis for its plant. Such licensee-initiated changes are subject to the NRC's formal regulatory controls, which ensure that a documented basis exists for licensee-initiated changes, and that the licensee obtains NRC review and approval, if required, before implementing the proposed changes. (I should note that some changes might not require prior NRC approval if they do not raise new safety issues.) The evolving nature of the licensing basis for each plant is documented in periodic updates to the plant's final safety analysis report, which are required by NRC regulations.

In short, we believe that our comprehensive reviews, combined with our continuous oversight by way of the Reactor Oversight Process, meet the need for safety assessments throughout a nuclear installation's life.

LICENSE RENEWAL AND NEW REACTOR LICENSING

Now let me turn to two current initiatives -- license renewal and new reactor licensing.

NRC Requirements for License Renewal

In the United States, the Atomic Energy Act authorizes the agency to issue operating licenses to nuclear plants for up to 40 years. Because the U.S. Congress based the 40-year license period on economic considerations, rather than a technical assessment of the length of time these plants can operate safely, the act also authorizes the NRC to extend operating licenses.

As I mentioned earlier, the increasing need for electric power and the improved performance of nuclear power plants over the past decade have inspired an increasing number of licensees to consider renewing their licenses instead of decommissioning their plants. To date, the NRC has received license renewal applications for 23 units and, ultimately, we expect that most licensees of our 103 operating power reactors will apply to extend their licenses.

The NRC has made license renewal a high priority. We realize that the simultaneous review of many renewal applications is a considerable challenge. However, we also recognize the safety implications of license renewal, and are committed to devoting significant attention and resources to this effort. We have also recently revised our guidance for the license renewal process to improve our effectiveness and efficiency in the years ahead. Accordingly, we have set performance goals for the processing of applications — 30 months, if there is a hearing, and 25 months, if not.

The License Renewal Rule focuses on providing assurance that licensees will manage the aging of long-lived passive structures and components, in accordance with the plant's current licensing basis, throughout the renewal term in the same manner and to the same extent as during the original license period. We focus on passive components because we determined that the performance of active components is adequately controlled during operations. The NRC also assesses the scope and impact of environmental effects that would be associated with license renewal of U.S. plants.

To meet the requirements of the License Renewal Rule, licensees have developed aging management programs that generally focus on prevention, mitigation, condition monitoring, or performance monitoring. In some instances, licensees may implement more than one type of aging management program to ensure that the aging effects are adequately addressed. Under the license renewal process, licensees can now reference our report on generic aging lessons learned for aging

management programs, rather than having to submit details of their individual programs, because the NRC has previously determined that certain programs are acceptable on a generic basis.

Various documents associated with license renewal process are included on the CD.[4]

NRC Requirements for New Reactor Licensing

In addition to renewing licenses of existing facilities, a few electric generating companies have expressed interest in building new plants. As I am sure you are aware, there have been no new reactor orders in the U.S. for about 25 years. To prepare for new construction, the NRC revised its regulations in 1989 to provide a more stable and predictable process for licensing nuclear power plants. This process, incorporated into the NRC's regulations as 10 CFR Part 52, includes the use of early site permits, standard design certifications, and combined construction permits and operating licenses.

This process is shown graphically on the overhead figure.

Under the Part 52 process, the NRC has already certified three new designs – namely, the General Electric Advanced Boiling Water Reactor, the ASEA Brown Bovari/Combustion Engineering System 80+ which is now part of the BNFL/Westinghouse portfolio, and the Westinghouse AP600. In addition, the staff is currently reviewing the Westinghouse AP1000, design that was submitted for design certification earlier this month. Industry representatives have also expressed interest in applying for early site permits, other advanced reactor design certifications, possible combined licenses, and restarting of suspended construction projects.

To ensure that we can effectively carry out our regulatory responsibilities associated with an early site permit application, a license application, and the construction of a new nuclear power plant, the NRC recently formed its New Reactor Licensing Project Office. The NRC's Office of Research is also helping to develop the technical foundation to support our review of new, innovative reactor designs. This is necessary because of the need to prepare to review some unusual reactor types; such as gas-cooled reactor designs.

We face challenges in dealing with these new reactor concepts. Much of our current regulatory basis assumes that the plant has a nuclear steam supply system with a light water-cooled and -moderated reactor. Applying a risk-informed regulatory approach to alternative plant designs, such as gas-cooled designs, requires much more than just a revision of current regulations. We are currently studying options for developing proper risk-informed criteria for such designs. I must also note that we seek international partners in the conduct of research so as to make effective use of experience in other nations relevant to these reactor designs.

Our Policy Statement on Advanced Reactors states that advanced reactors must, at a minimum, provide at least the same degree of protection of the public and the environment as our current light-water reactors. The policy further states that we expect future reactor designs to achieve a higher level of safety for certain technical and severe accident issues than the designs of currently operating nuclear power plants. These expectations were realized in the three designs that we have already certified and we expect they will be satisfied with other new designs.

In sum, we have the prospect in the United States for the continued substantial contribution of nuclear energy to electrical supply through both the life extension of existing reactors and the possibility of new construction. We are dedicated to ensuring that these activities maintain very high safety standards.

CONCLUSION

Let me conclude this statement by noting that the U.S. Government and the NRC believe that this Convention represents a milestone in international cooperation in furtherance of nuclear safety. It presents an unprecedented opportunity for the Contracting Parties to work together to enhance the level of nuclear safety worldwide.

Our review of the reports of other Contracting Parties has challenged the NRC to think about the variety of approaches to nuclear reactor regulation. They have caused us to reflect upon our national regulatory processes and how they might be improved. I hope that I have clarified some of the areas of the United States National Report in which you expressed interest and that some of our

approaches have similarly stimulated interest in you. I look forward to additional discussion of our practices, both during today's session and in the years to come.

Mr. Chairman, this completes my statement. Thank you.

OFFICE OF PUBLIC AFFAIRS

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No. S-02-012

PDF Version (28 KB)

Plenary Statement of the United States at the Second Review Meeting of the Convention on Nuclear Safety

Dr. Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission

April 15, 2002
Vienna, Austria

The U.S. Government and our delegation to this Convention are pleased to have the opportunity to attend this Second Review Meeting of the Contracting Parties to the Convention on Nuclear Safety (CNS). Our participation reaffirms our view of the importance of international cooperation in the promotion of a high level of nuclear safety worldwide.

The next two weeks will provide a unique opportunity for discussions concerning activities to enhance nuclear safety in virtually all States utilizing nuclear energy for electric power production. I am confident that this forum will benefit all such States -- including the United States -- and will provide a better understanding of current issues and trends. Each of us bears the important responsibility of ensuring nuclear safety within our countries. I am hopeful that this review meeting will identify ways to improve nuclear safety and that, in response to our discussions, nuclear safety will be enhanced around the globe.

The United States acknowledges that the purpose of this meeting is to concentrate on the safety of nuclear installations. I recognize that the attacks of September 11th have focused world attention on the elevated threat of terrorism, including the possibility of nuclear-related terrorism. These terrible assaults have caused all of us to examine the risk of a terrorist effort to exploit any vulnerabilities in the defenses of nuclear power plants or to use nuclear materials for harmful purposes. It is not the purpose of this meeting to discuss the measures that countries are taking to address the need for heightened safeguards and security programs. But I do not want the absence of this matter on our meeting agenda to imply that such threats are being taken lightly. There have been aggressive efforts in the United States to respond to the terrorist threat. And the United States supports appropriate activities in other fora, including activities through the auspices of the International Atomic Energy Agency, that will provide focused attention on the security dimension of our safety responsibilities. This approach will allow this review meeting to focus more broadly on the challenges to nuclear safety arising from all types of risks.

As you are all well aware, the Convention on Nuclear Safety seeks to achieve the improvement of safety through the peer review process that will be undertaken over the next several days. It is apparent that the Convention will only be as effective as each of us makes it. Each of us should welcome the probing inquiry of other colleagues, for it is through the ensuing discussion that all of us can learn and improve our practices.

I also believe that all of us should view the Convention on Nuclear Safety as a living document. We have the opportunity to structure and focus these review meetings in ways that will better facilitate the achievement of the Convention's objectives. With that thought in mind, I urge all of you to consider during the course of this meeting how we might make the next review meeting even more

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effective. We are embarked on a common and important enterprise in improving nuclear safety and we should ensure that these review meetings are designed in a way that best furthers that objective.

I look forward to presenting the U.S. National Report, where I will describe how the United States seeks to meet its obligations under the Convention. I encourage your questions as I am hopeful that the resulting discussion will provide us all with insights that will be helpful in furthering efforts to enhance nuclear safety around the globe. Similarly, U.S. delegates will actively engage in the peer review of the programs of other Contracting Parties, and I hope that our questions and comments will be of use to you. It is only through a vigorous and candid review that the objectives of the Convention can be met.

I would like to conclude by expressing my country's support for, and confidence in, CNS President Gregoric. His experience in leading international groups addressing similarly complex nuclear safety issues will directly affect the successful outcome of this meeting. We pledge our fullest cooperation in assisting him with this important task.

Thank you.

PNO-IV-02-023 - Wolf Creek Nuclear Operating Corporation

May 16, 2002

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO-IV-02-023

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Headquarters staff on this date.

Facility	__ **Licensee Emergency Classification**
Wolf Creek Nuclear Operating Corporation	__ Notification of Unusual Event
Wolf Creek Nuclear Plant	__ Alert
Burlington, Kansas	__ Site Area Emergency
Dockets: 50-482	__ General Emergency
	X Not Applicable

Subject: ***Unplanned Shutdown Due to Loose Part in Steam Generator D***

Description:

On May 13, 2002, at 4:17 p.m. (CDT), the licensee commenced a reactor shutdown based on the indication of a loose part in the Steam Generator D hot leg bowl. The loose part was identified through acoustic monitoring instrumentation. The noise signatures from the acoustic monitors were reviewed by Westinghouse and, based on their recommendation, the licensee shut down the plant. The reactor entered Mode 5 (cold shutdown) at 10:48 a.m. on May 14. The loose part will be retrieved and the steam generator will be inspected for any potential damage that may have been caused by the loose part. Once the part is retrieved and identified, the licensee will finalize their plans for restarting the plant.

As of 12 p.m. (CDT) on May 16, the licensee was preparing to enter midloop operation and open up Steam Generator D for inspection.

The state of Kansas will be informed.

The licensee did make a press release regarding this issue.

The Resident Inspector received initial notification of this occurrence at 4:30 p.m. on May 13, 2002.

Region IV has informed EDO/NRR/PA.

This information has been discussed with the licensee and is current as of May 16, 2002, at 12:45 p.m. (CST).

CONTACTS: Ron Kopriva 817-860-8104
Elmo Collins 817-860-8291

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EA-02-010 - D.C. Cook, Unit 2 (Indiana and Michigan Electric Company)

May 6, 2002

EA-02-010

Mr. A. C. Bakken III
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan MI 49107

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT 50-315/02-02; 50-316/02-02) (D.C. COOK NUCLEAR POWER PLANT)

Dear Mr. Bakken:

The purpose of this letter is to provide you with the final results of our significance determination of the preliminary White finding identified in the subject inspection report. The inspection finding was assessed using the significance determination process and was preliminarily characterized as White, an issue with low to moderate importance to safety, which may require additional NRC inspections. This White finding involved your staff's failure to take corrective action to preclude a repetitive failure of the Unit 2 Turbine Driven Auxiliary Feedwater Pump, a significant condition adverse to quality.

During a meeting with Anton Vegel of NRC, Region III, on April 23, 2002, Joseph E. Pollock, Site Vice President for D.C. Cook, confirmed that American Electric Power did not contest the characterization of the risk significance of this finding and that you declined your opportunity to discuss this issue in a Regulatory Conference.

After considering the information developed during our inspections, the NRC has concluded that the inspection finding is appropriately characterized as White.

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

The NRC has also determined that the failure to take corrective action to preclude repetition of the August 10, 2001, failure of the Unit 2 Turbine Driven Auxiliary Feedwater Pump is a violation of 10 CFR Part 50, Appendix B, Criterion XVI, as cited in the enclosed Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

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

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

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you provide one, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Geoffrey E. Grant, Director
Division of Reactor Projects

Docket No. 50-316
License No. DPR-74

Enclosure: Notice of Violation

cc w/encl:
J. Pollock, Site Vice President
M. Finissi, Plant Manager
R. Whale, 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

Indiana and Michigan Electric Company
D. C. Cook Nuclear Power Plant, Unit 2

Docket No. 50-316
License No. DPR-74
EA-02-010

During NRC inspections conducted from November 21, 2001, through March 31, 2002, 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 Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.
>
> Contrary to the above, the licensee failed to take corrective action to preclude repetition of the August 10, 2001, failure of the Unit 2 Turbine Driven Auxiliary Feedwater Pump (TDAFWP), a significant condition adverse to quality. Specifically, the Unit 2 TDAFWP failed to start on August 10, 2001, because the trip throttle valve latch mechanism failed to remain engaged during pump start. On December 13, 2001, the licensee obtained information from the trip throttle valve vendor identifying critical parameters for the trip hook mechanism geometry and alignment. The licensee failed to promptly perform corrective actions to verify that the Unit 2 TDAFWP trip hook conformed to these critical parameters. Consequently, a second failure of the Unit 2 TDAFWP occurred on January 18, 2002, because the trip throttle valve latch mechanism failed to remain engaged during pump start. Subsequent investigation determined that the cause of the August 10, 2001 and January 18, 2002 failures was due to incorrect trip hook geometry and alignment.

This violation is associated with a White SDP finding.

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

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

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). 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 2.790(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 6th day of May 2002

EA-02-022 and EA-01-310 - Shearon Harris (Carolina Power & Light Company)

April 16, 2002

EA-00-022
EA-01-310

Carolina Power & Light Company
ATTN: Mr. James Scarola
Vice President - Harris Plant
Shearon Harris Nuclear Power Plant
P. O. Box 165, Mail Code: Zone 1
New Hill, NC 27562-0165

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (SHEARON HARRIS NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-400/00-09)

Dear Mr. Scarola:

The purpose of this letter is to provide you with the final results of our significance determination for the preliminary White finding and our conclusions related to the significance of two apparent violations associated with the Thermo-Lag fire barrier assembly at Carolina Power and Light Company's (CP&L) Shearon Harris Nuclear Power Plant. The fire barrier serves as the fire area separation barrier between Fire Area 1-A-SWGR-B [B Train Switchgear Room/Auxiliary Control Panel Room] and Fire Area 1-A-CSR-A [A Train Cable Spreading Room]. Based on your Thermo-Lag barrier fire resistance tests conducted in 1994 and 1995, this fire barrier did not have the required three-hour fire resistance rating. The inspection finding was assessed using the Significance Determination Process (SDP) and was preliminarily characterized as White (i.e., an issue with low to moderate increased importance to safety, which may require additional NRC inspections). At your request, an open regulatory conference was conducted with you and members of your staff on January 31, 2002, to discuss your views on this issue. In addition, a predecisional enforcement conference was also held to discuss a related matter involving a change that CP&L made to the Harris Fire Protection Program, as discussed below. Enclosure 2 lists the attendees at the regulatory and predecisional enforcement conferences. Enclosures 3 and 4 include copies of the material presented at the conference by the NRC and CP&L, respectively.

During the regulatory conference, CP&L representatives described the analytical approach used in determining the significance of the finding, provided a description of the physical configuration of the fire area, and discussed the key factors considered by CP&L in estimating the change in core damage frequency (CDF). CP&L's presentation highlighted the major differences between its best estimate of the incremental increase in CDF and the NRC's preliminary assessment. Based on the results of your analysis, CP&L concluded that the incremental increase in CDF was consistent with a Green finding.

After the regulatory conference, the NRC revised its risk assessment after considering the information discussed at the conference, and an additional review of the factors and assumptions used in the NRC's initial determination of the increase in risk. The NRC's letter of March 18, 2002, forwarded the NRC's revised risk estimate and the basis for the factors that support the estimate, and offered CP&L the opportunity to provide its perspectives on the updated information. CP&L subsequently informed the NRC by telephone that the risk information presented at the regulatory conference was sufficient to characterize the risk properly, and that it did not intend to provide

any additional technical information on the matter at this time.

After considering the information developed during the inspection, the information you provided at the conference, as well as the information developed and revised by the NRC after the conference, the NRC has concluded that the inspection finding resulted in an incremental increase in CDF of approximately $7x10^{-6}$ per year. The technical basis for this determination was fully discussed in the revised risk assessment forwarded to CP&L by letter dated March 18, 2002. Accordingly, the final significance of the finding is characterized as White. In addition, the NRC has concluded that the fire area separation barrier failed to comply with 10 CFR 50.48, Harris Operating License Condition 2.F, and Updated Final Safety Analysis Report (UFSAR) 9.5.1.2.2, in that the fire resistance rating was indeterminate instead of the required three hour rating. Additional enforcement aspects of this issue are discussed below.

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

A predecisional enforcement conference was also held on January 31, 2002, to discuss an apparent violation involving a change made by CP&L to the Fire Protection Program. This finding was not evaluated under the SDP but was considered for escalated enforcement action because it appears to have impacted the NRC's ability to perform its regulatory function.

Based on the information developed during the inspection and the information you provided during the predecisonal enforcement conference, the NRC has determined that the change made by CP&L to the Fire Protection Program resulted in a violation of License Condition 2.F of the Harris Operating License. Specifically, the change to the Fire Protection Program in 1997 involved revising the rating of the Thermo-Lag fire barrier assembly from three hours to that suitable for the hazard. The NRC concluded that this change increased the likelihood that both redundant divisions or trains of safety-related systems could be damaged by a single fire. As such, this change required prior NRC approval, in that it adversely affected the ability to achieve and maintain safe shutdown in the event of a fire. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions - May 1, 2000" (Enforcement Policy), NUREG-1600, this violation is characterized as a Severity Level III violation because the significance of the change resulted in a low to moderate increase in risk.

In accordance with the Enforcement Policy, a base civil penalty in the amount of $60,000 is considered for a Severity Level III violation. Because your facility has not been the subject of escalated enforcement action within the last two years, the NRC considered whether credit was warranted for *Corrective Action* in accordance with the civil penalty assessment process described in Section VI.C.2 of the Enforcement Policy. Your corrective actions included the initiation of fire watches in the area, the consideration of various modification options with the long term intent to restore the barrier to the required three hour fire rating, and a re-emphasis during the design change process to consider inputs from multiple site organizations and disciplines. Based on the above, the NRC concluded that your actions were prompt and comprehensive, and credit was warranted for the factor of *Corrective Action*.

Therefore, to encourage prompt and comprehensive correction of violations and in recognition of the absence of previous escalated enforcement action, I have been authorized to propose that no civil penalty be assessed in this case. However, similar violations in the future could result in further escalated enforcement action.

As CP&L discussed at the conferences, the non-compliance involving the indeterminate fire barrier rating and the non-compliance involving the inappropriate change to the Fire Protection Program both stemmed, in part, from incorrect decisions by CP&L regarding the acceptability of the fire barrier testing results. The NRC agrees with this determination, and has concluded that it is appropriate to cite these non-compliances as one violation in the enclosed Notice of Violation (Notice).

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 is already adequately addressed on the docket in this letter and in the information presented by CP&L at the regulatory and predecisional enforcement conferences. Therefore, you are not required to respond to the provisions of 10 CFR 2.201 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.

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

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room (PDR) or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/ADAMS.html (the Public Electronic Reading Room).

If you have any questions regarding this matter, please contact Charles Casto, Director, Division of Reactor Safety, at 404-562-4600.

Sincerely,

/RA/

Luis A. Reyes
Regional Administrator

Docket No.: 50-400
License No.: NPF-63

Enclosure: Notice of Violation

cc w/encls:

Terry C. Morton, Manager
Performance Evaluation and
Regulatory Affairs CPB 9
Carolina Power & Light Company
Electronic Mail Distribution

Robert J. Duncan II
Director of Site Operations
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
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Ben Waldrep
Plant General Manager--Harris Plant
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Richard H. Givens, Chairman
Board of County Commissioners
of Chatham County
Electronic Mail Distribution

NOTICE OF VIOLATION

Carolina Power and Light Company
Shearon Harris Nuclear Power Plant
Unit 1

Docket Nos.: 50-400
License Nos.: NPF-63
EA-00-022, EA-01-310

During an NRC inspection completed on December 12, 2001, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions - May 1, 2000," (Enforcement Policy), the violation is listed below:

10 CFR 50.48 requires that all operating nuclear power plants have a fire protection

program that satisfies Criterion 3 of Appendix A to 10 CFR 50.

Harris Operating License NFP-63, Condition 2.F, "Fire Protection Program," specifies, in part, that Carolina Power and Light (CP&L) implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report (UFSAR) for the facility as amended and as approved in the Safety Evaluation Report (SER) dated November 1983 (and supplements 1 through 4), and the Safety Evaluation dated January 12, 1987. License Condition 2.F permits the licensee to make changes to the approved fire protection program without prior NRC approval only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Harris UFSAR Sections 9.5.1.2.2, "Barriers and Access," states that fire barriers with a minimum fire resistance rating of three hours are provided such that both redundant divisions or trains of safety-related systems are not subject to damage from a single fire to the extent possible in accordance with NRC position C.5.b.(2) of Branch Technical Position Chemical Engineering Branch (CMEB) 9.5-1 (NUREG-0800), July 1981.

Harris UFSAR Section 9.5.1.2.2 and Section 9.5.1.4 of the SER dated November 1983 identifies the Thermo-Lag fire barrier wall assembly between the B Train Switchgear Room/Auxiliary Control Panel Room and the A Train Cable Spreading Room as a three-hour rated fire barrier.

Contrary to the above, the licensee failed to implement and maintain NRC approved fire protection program safe shutdown system separation requirements for the Thermo-Lag fire area separation barrier between the B Train Switchgear Room/Auxiliary Control Panel Room and the A Train Cable Spreading Room. The installed fire area separation barrier had an indeterminate fire resistance rating instead of three hours as referenced in the Harris UFSAR and NRC SERs that established the approved fire protection program. In addition, on August 18, 1997, the licensee made changes to the approved fire protection program without prior Commission approval, that adversely affected the ability to achieve and maintain safe shutdown in the event of a fire. In Safety Evaluation 97-255, the licensee accepted the condition of a degraded Thermo-Lag fire barrier assembly between the B Train Switchgear Room/Auxiliary Control Panel Room and the A Train Cable Spreading Room B in lieu of the intended three-hour fire rating. The licensee made changes to UFSAR Sections 9.5 and 9.5A by revising the rating of the Thermo-Lag fire barrier assembly from three-hour rated to that suitable for the hazard. This change increased the likelihood that both redundant divisions or trains of safety-related systems could be damaged by a single fire. Therefore, this change could adversely affect the ability to achieve and maintain safe shutdown in the event of a fire, and thus required NRC approval prior to its implementation.

This violation is characterized as a Severity Level III violation and is associated with a White SDP finding.

The NRC has concluded that information regarding the reason for the violations, the corrective actions taken and planned to correct the violations and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in this letter and in the information presented by Carolina Power and Light Company at the regulatory and predecisional enforcement conference. 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," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region RII, 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 Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC web site at

http://www.nrc.gov/reading-rm/ADAMS.html (the Public Electronic Reading Room). Therefore, to the extent possible, the response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

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

Dated this 16th day of April 2002

1Q/2002 ROP Inspection Findings Summary

Note: This summary provides the color designation of the most significant inspection findings over the previous 4 quarters

Plants	Initiating Events	Mitigating Systems	Barrier Integrity	Emergency Preparedness	Occupational Radiation Safety	Public Radiation Safety	Physical Protection
Arkansas Nuclear 1							
Arkansas Nuclear 2							
Beaver Valley 1							
Beaver Valley 2							
Braidwood 1							
Braidwood 2							
Browns Ferry 2							
Browns Ferry 3							
Brunswick 1							
Brunswick 2							
Byron 1							
Byron 2							
Callaway		White (1)					
Calvert Cliffs 1		Yellow (1)					
Calvert Cliffs 2							
Catawba 1							
Catawba 2							
Clinton				White (1)			
Columbia Generating Station				Yellow (1)			
Comanche Peak 1						White (1)	
Comanche Peak 2						White (1)	
Cooper		White (1)		White (3)			
Crystal River 3							
D.C. Cook 1							
D.C. Cook 2							
Davis-Besse							
Diablo Canyon 1							
Diablo Canyon 2							
Dresden 2							

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Dresden 3							
Duane Arnold							
Farley 1							
Farley 2							
Fermi 2		White (1)					
FitzPatrick							
Fort Calhoun							
Ginna							
Grand Gulf 1							
Harris 1							
Hatch 1							
Hatch 2							
Hope Creek 1							
Indian Point 2							
Indian Point 3							
Kewaunee							
La Salle 1							
La Salle 2							
Limerick 1				White (1)			
Limerick 2			White (1)	White (1)			
McGuire 1							
McGuire 2							
Millstone 2							
Millstone 3							
Monticello							
Nine Mile Point 1							
Nine Mile Point 2							
North Anna 1							
North Anna 2							
Oyster Creek							White (1)
Oconee 1		White (1)					
Oconee 2							
Oconee 3							
Palisades		White (1)					
Palo Verde 1							
Palo Verde 2							
Palo Verde 3							
Peach Bottom 2				White (1)			
Peach Bottom 3				White (1)			
Perry 1							
Pilgrim 1							
Point Beach 1							
Point Beach 2							
Prairie Island 1		White (1)					

Prairie Island 2		White (1)					
Quad Cities 1							
Quad Cities 2							
River Bend 1							
Robinson 2							
Saint Lucie 1							
Saint Lucie 2							
Salem 1							
Salem 2							
San Onofre 2							
San Onofre 3							
Seabrook 1		White (1)					
Sequoyah 1							
Sequoyah 2							
South Texas 1							
South Texas 2							
Summer							
Surry 1		White (1)					
Surry 2		White (1)					
Susquehanna 1				White (1)			
Susquehanna 2				White (1)			
Three Mile Island 1							
Turkey Point 3							
Turkey Point 4							
Vermont Yankee							Yellow (1)
Vogtle 1							
Vogtle 2							
Waterford 3							
Watts Bar 1							
Wolf Creek 1							

▲ PI Summary | Reactor Oversight Process

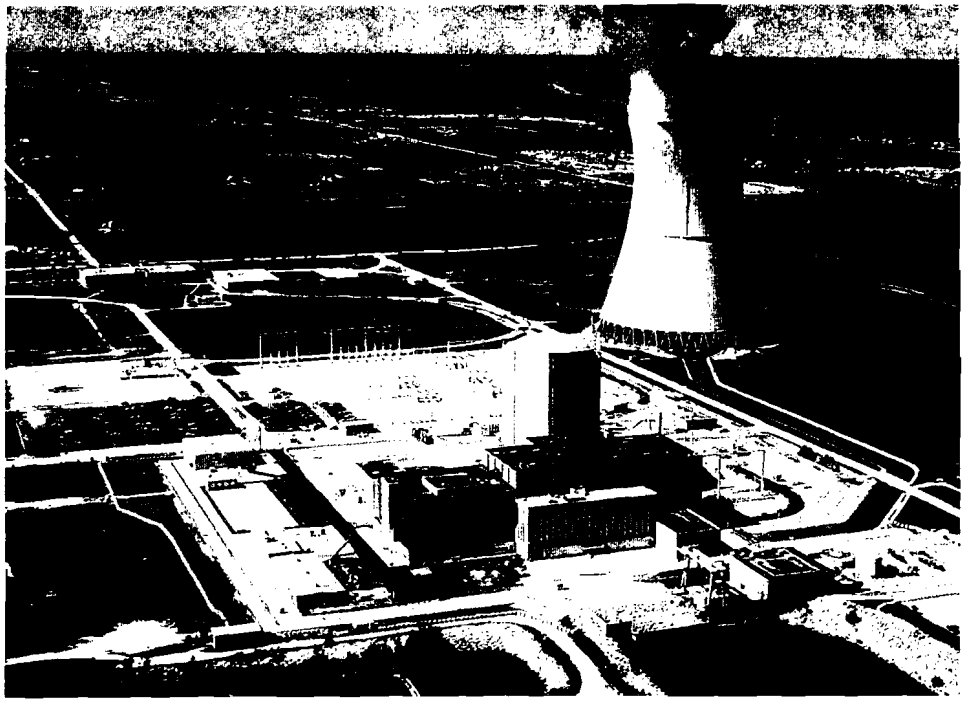
Last Modified: May 3, 2002

Advisory Committee on Reactor Safeguards Update of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Closure Head Activities

June 6, 2002

2

Agenda



Introduction

- Jim Powers

Update of RPV Closure Head

Field Activities

- Mark McLaughlin

RPV Closure Head Replacement

- Bob Schrauder

Root Cause Analysis

- Steve Loehlein

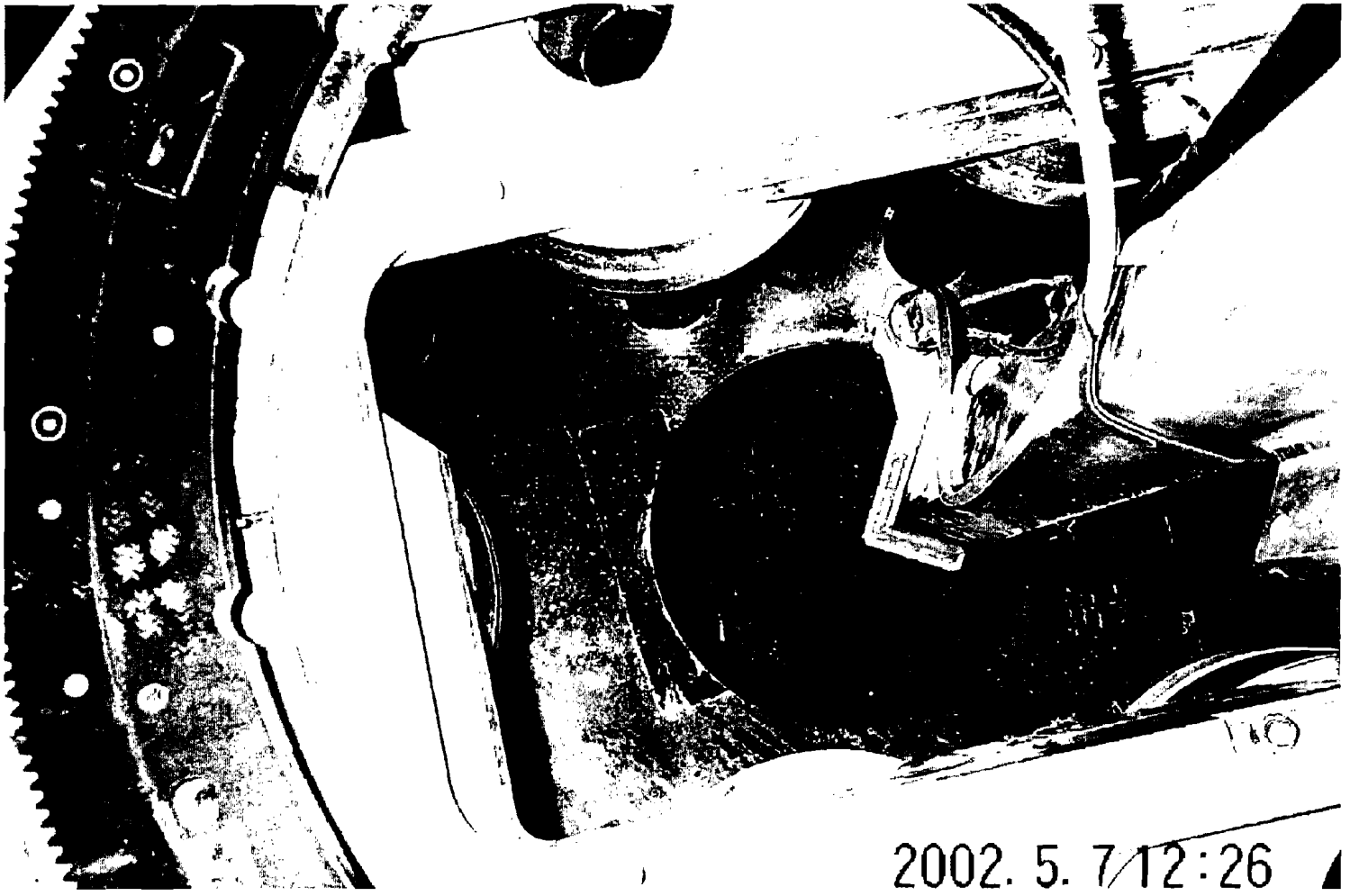
Concluding Remarks

- Jim Powers

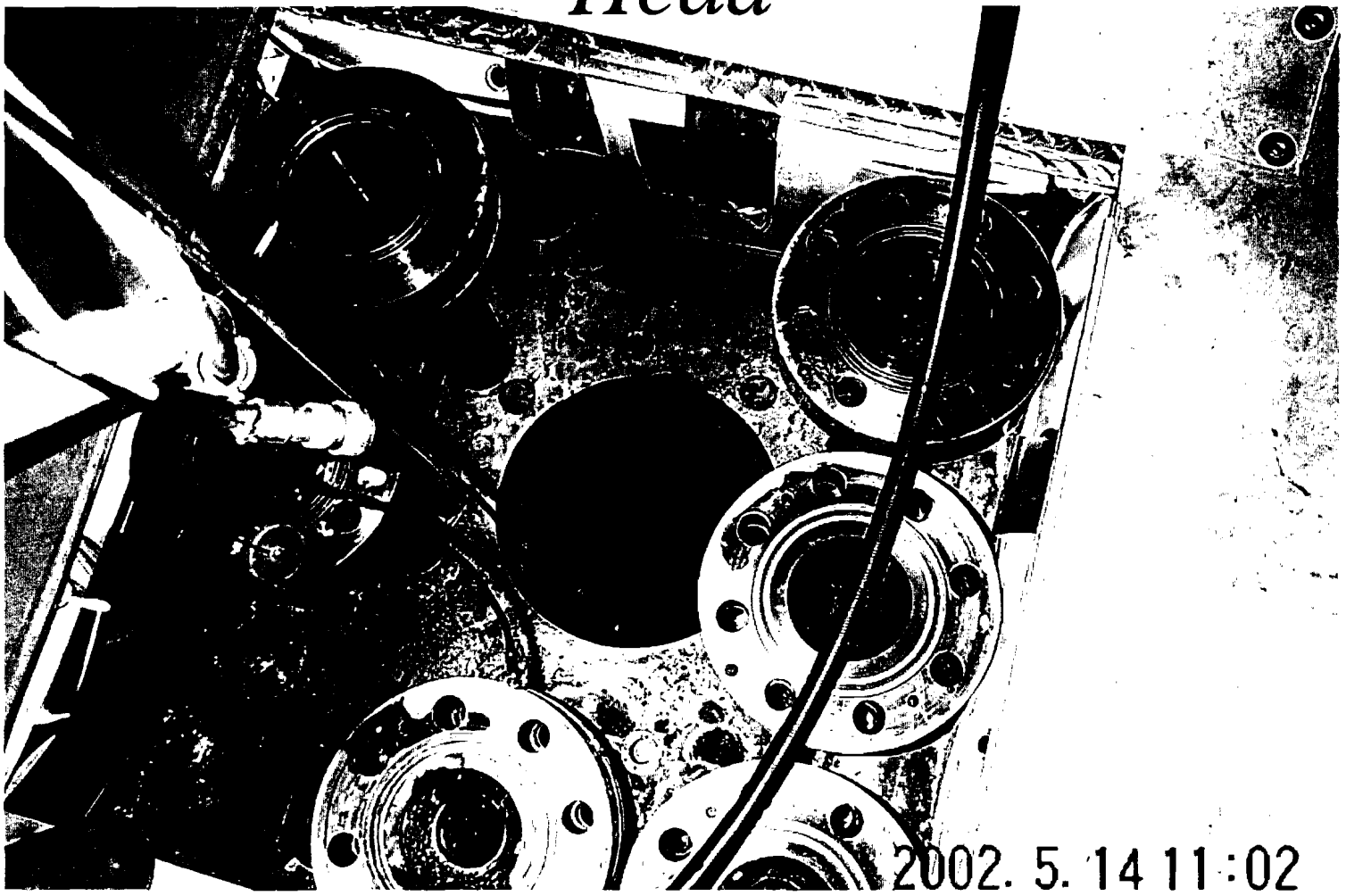
Update of RPV Closure Head Field Activities

Mark McLaughlin
Field Activities Team Leader

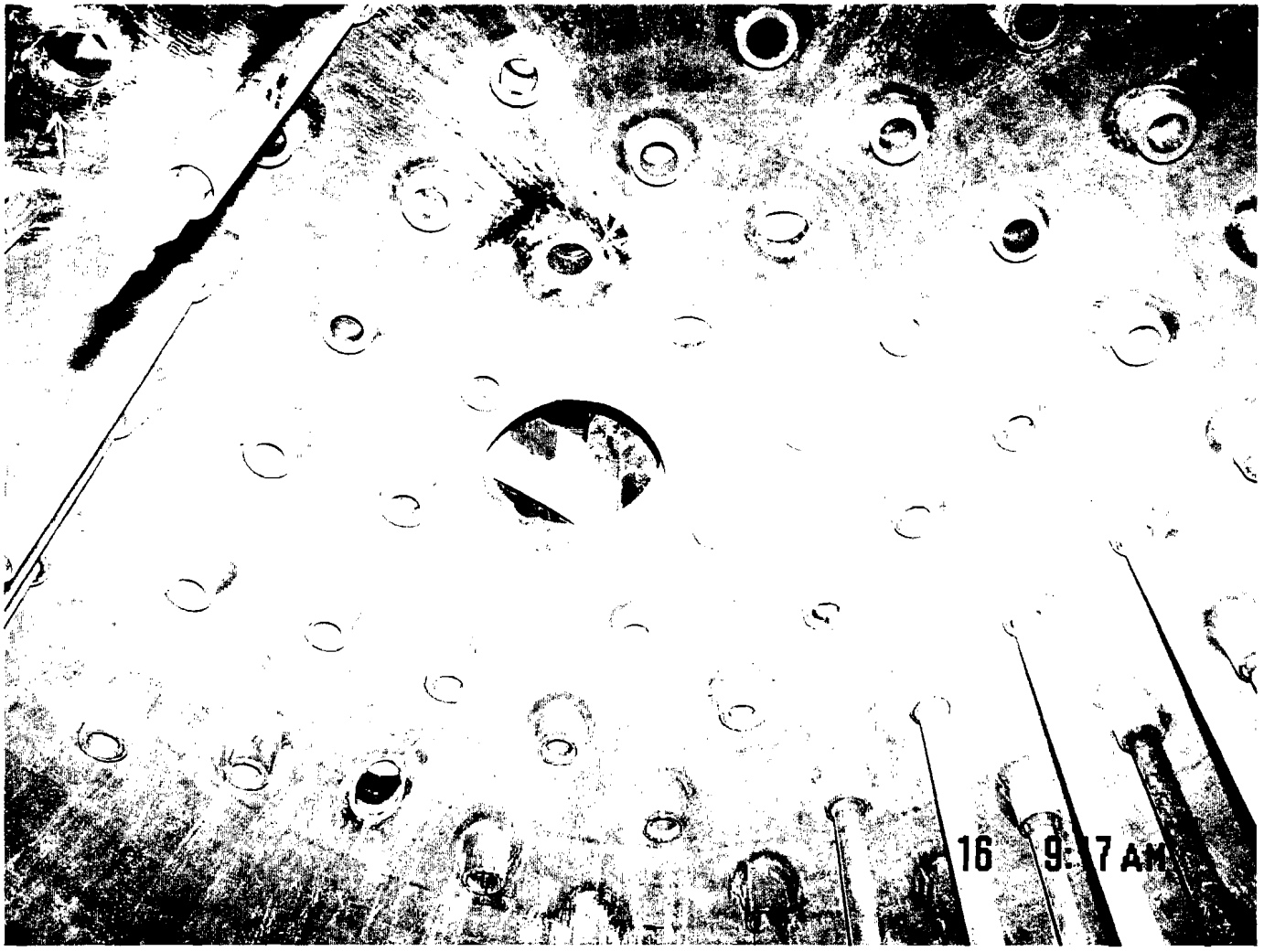
Abrasive Water Jet



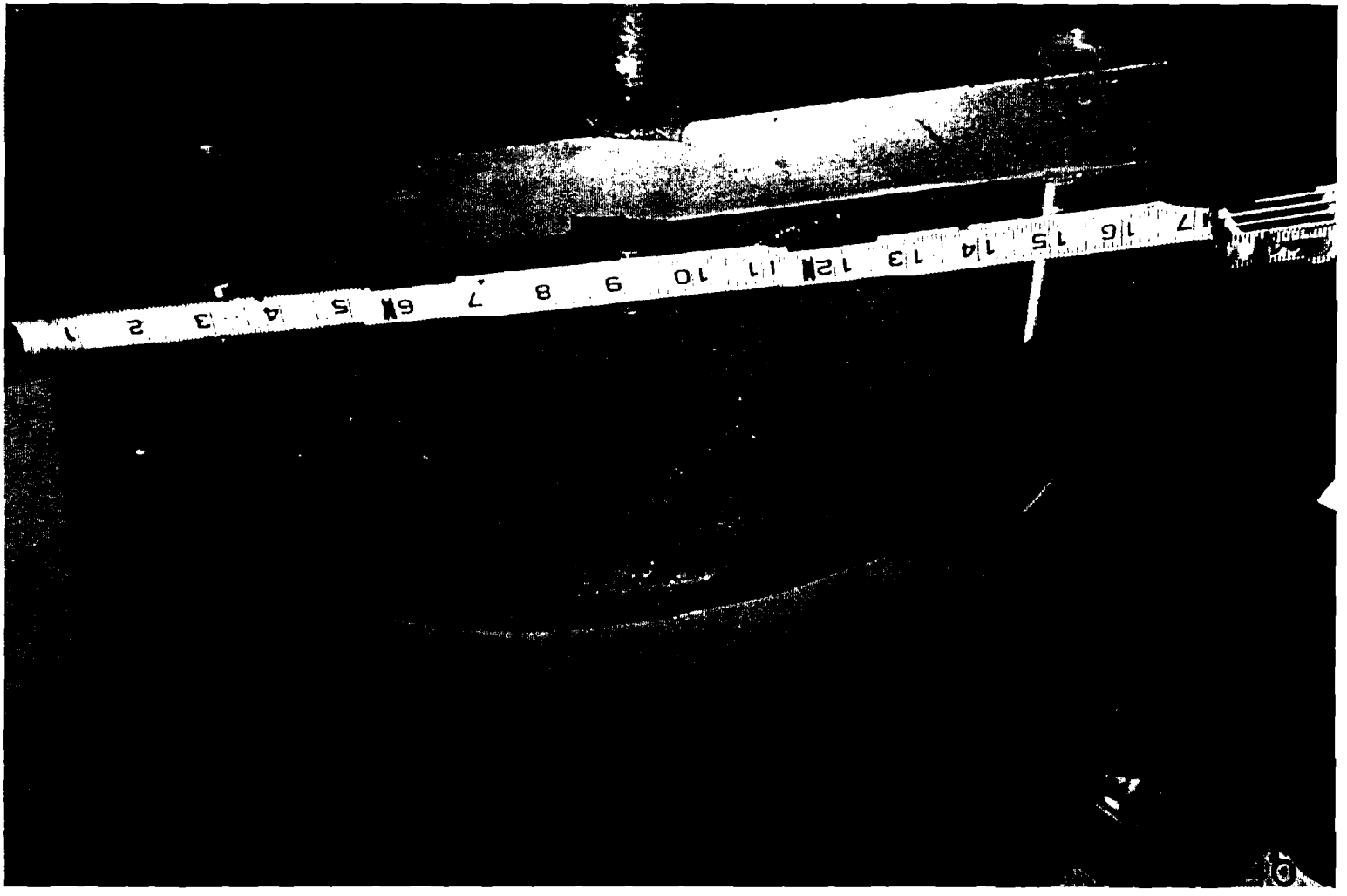
Area Removed from RPV Closure Head



Underneath RPV Closure Head



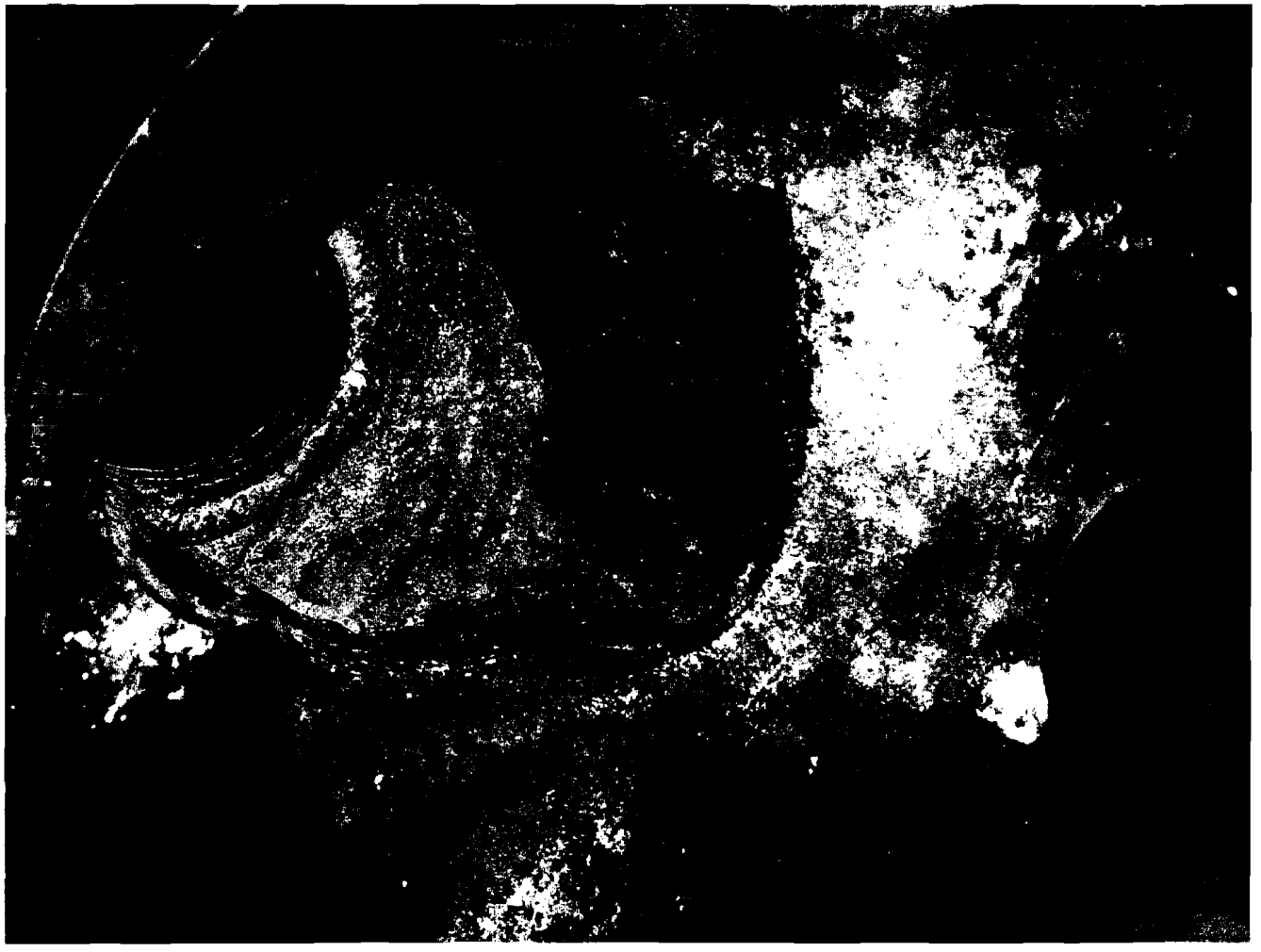
RPV Closure Head Cutout



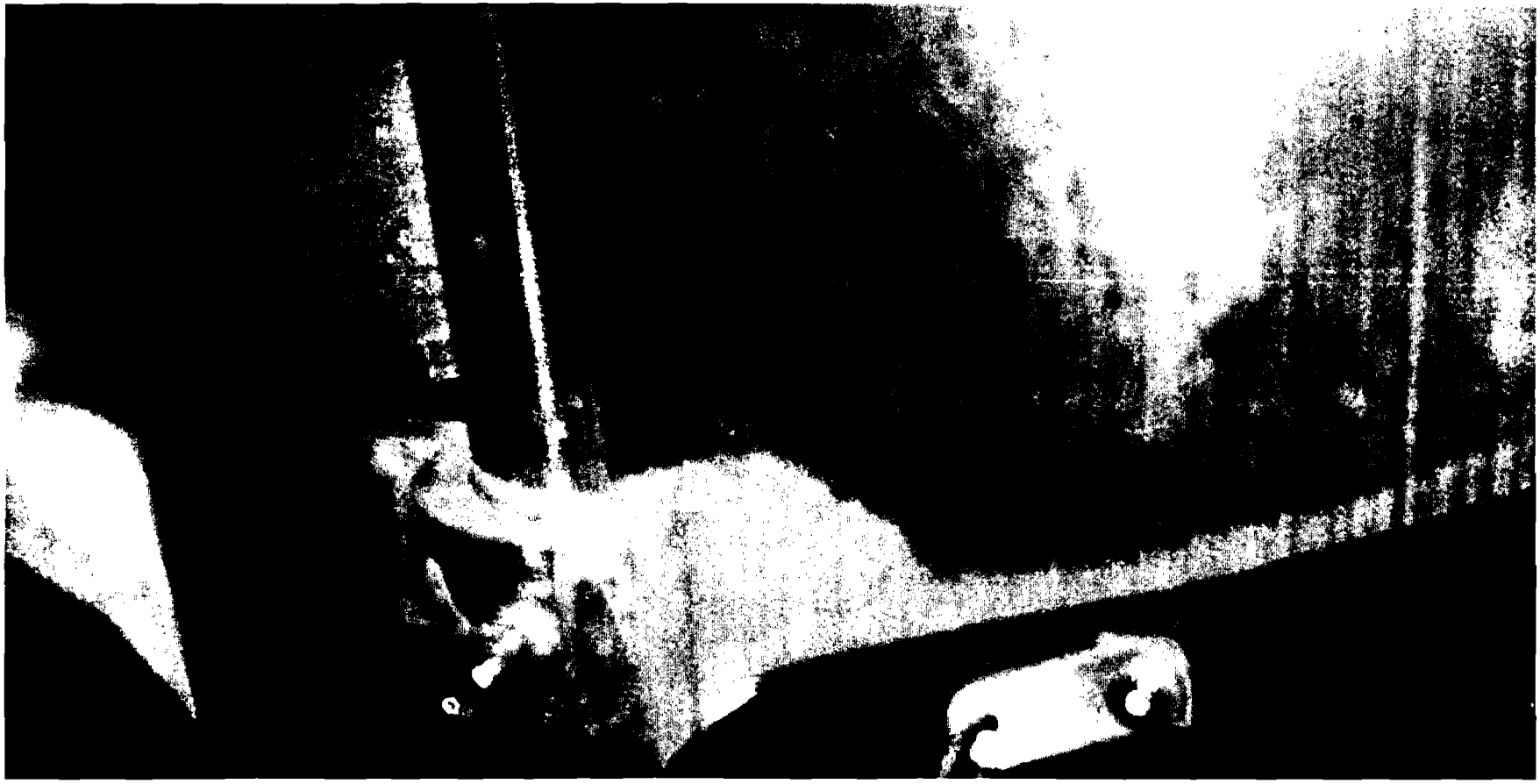
Sample Plan

- Phase 1
  - Corrosion products/boric acid deposits from top of head
  - Deposits scraped from CRD nozzle 3 below the flange
  - Draft report issued for Davis-Besse review
- Phase 2
  - Corrosion products/boric acid deposits from nozzle 2 removal
- Phase 3
  - Nozzle 3 and nozzle 3 corrosion area
  - Nozzle 2

RPV Closure Head Cutout



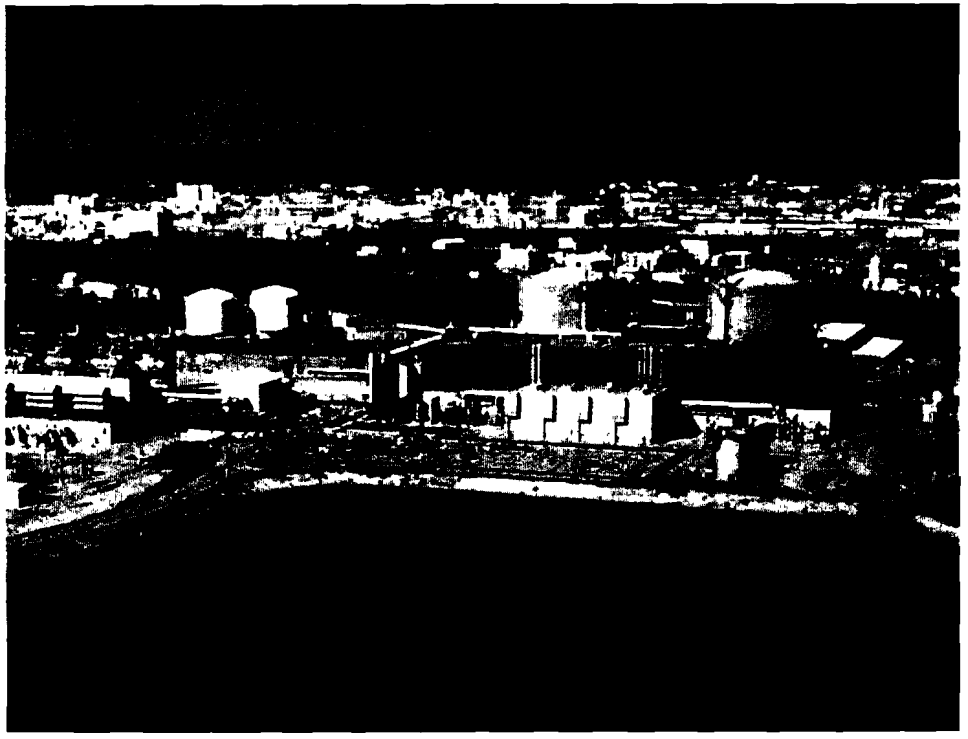
Nozzle 3 Cutout Cladding Interface



Reactor Pressure Vessel Closure Head (RVPCH) Replacement

Bob Schrauder
Engineering Services

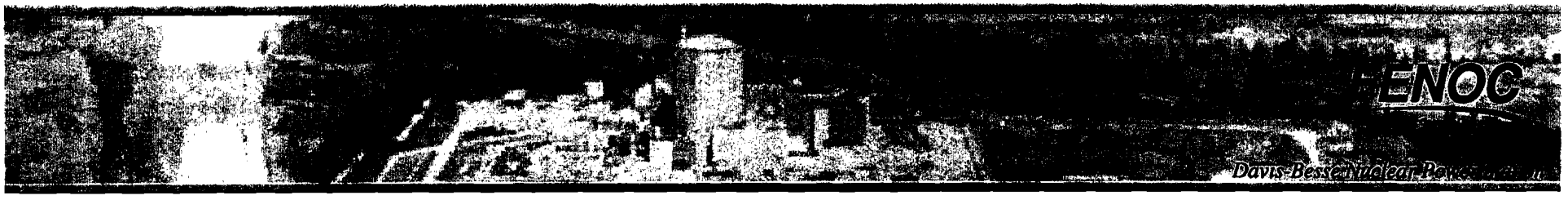
RPVCH Replacement Considerations



- The Midland RPVCH is
  - Similar in design to the Davis-Besse RPVCH
  - Readily available
  - Not contaminated

Replacement RPVCH

- Midland RPVCH was fabricated by Babcock and Wilcox
  - Manufactured to ASME Boiler & Pressure Vessel Code Section III, Code Class A, 1968 Edition, Summer 1968 Addenda
  - Accepted by Consumers Power and an Authorized Nuclear Inspector as an acceptable ASME component
  - Hydrostatically tested at 3125 psig per ASME Code Requirements



Replacement RPVCH

- Framatome-Advanced Nuclear Power (FRA-ANP) has purchased Midland RPVCH and is compiling/validating the ASME Code Data Package
- FRA-ANP is reconciling the Midland RPVCH against Davis-Besse design requirements
- FRA-ANP activities are governed by their safety-related Quality Assurance program, including 10CFR21 reporting

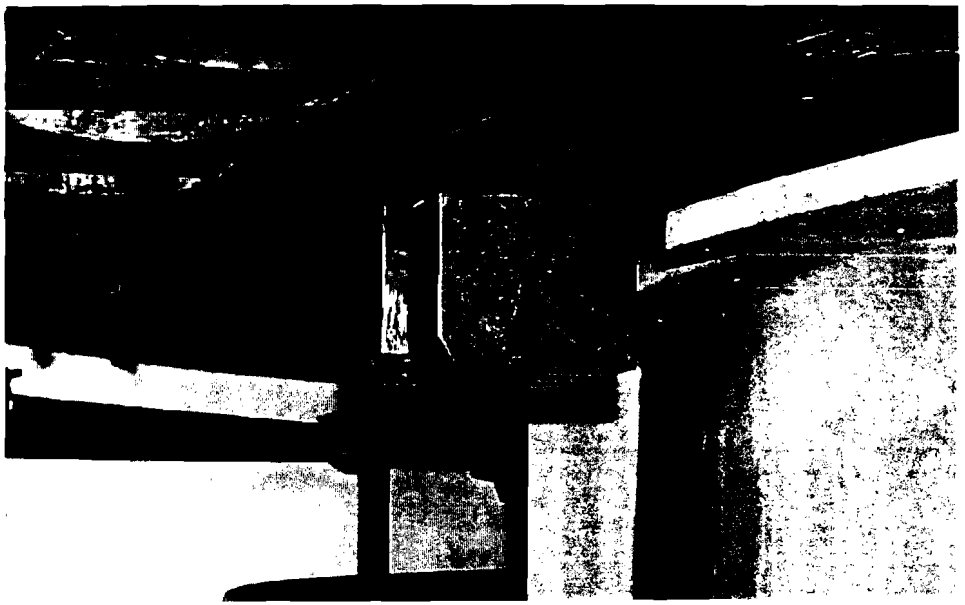
Replacement RPVCH Comparison to Davis-Besse RPVCH

	Davis-Besse	Midland
Material of Construction		
Closure Head	SA-533, GR B Cl 1	Same
Closure Head Flange	SA-508, Cl 2	SA-508-64, Cl 2
CRDM Nozzle	Inconel SB-167	Same
CRDM Flange	SA-182, F-304	Same
Design		
Pressure	2500 psig	Same
Temperature	650 degree F	Same

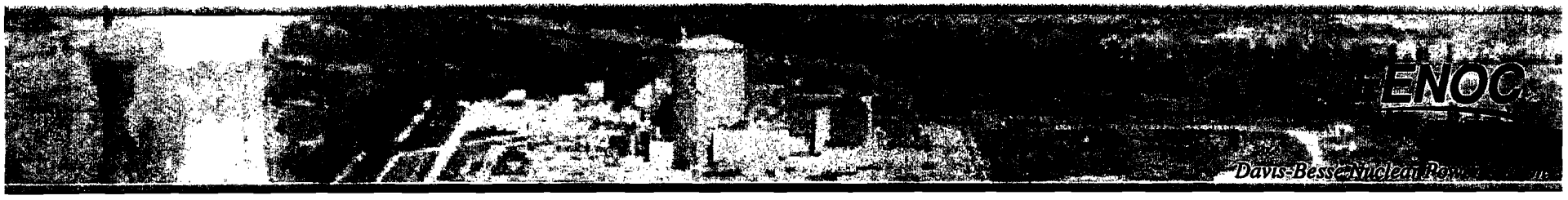
Replacement RPVCH CRD Nozzles

- Midland's Control Rod Drive (CRD) nozzles are similar to Davis-Besse
  - 68 Nozzles: Material Heat M7929
  - 1 Nozzle: Material Heat M6623
- Alignment of control rods to RPVCH nozzles is consistent with original Davis-Besse design

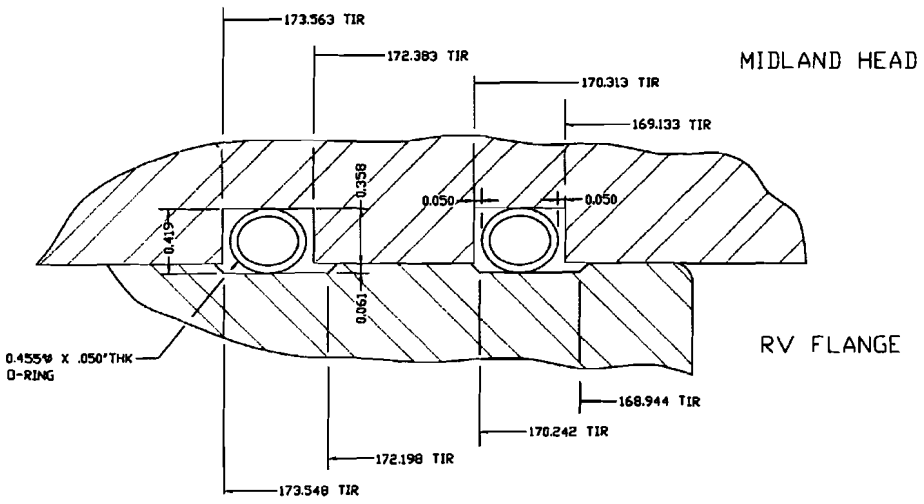
Replacement RPVCH



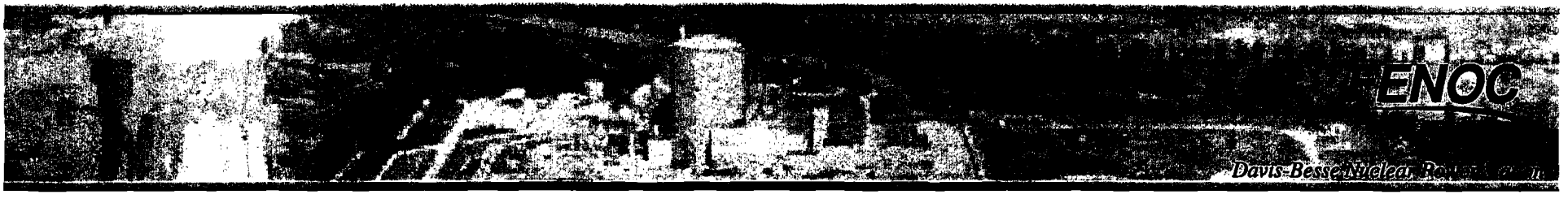
- Minor machining of 4 out of 8 vessel-to-head key-way surfaces is required
- The Midland CRDM flange indexing pin hole locations will be modified to match the proper Davis-Besse azimuth-orientation



Replacement RPVCH



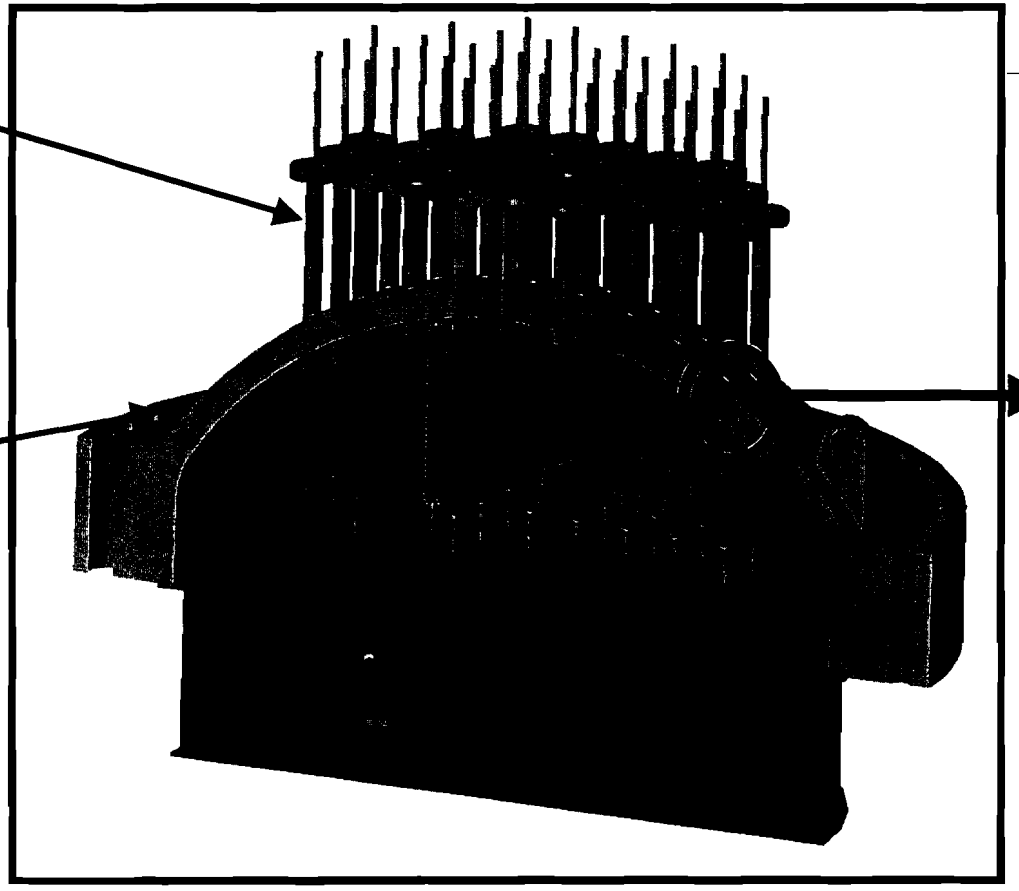
- Minor differences in RPVCH O-ring design
  - O-ring grooves are slightly different requiring the use of smaller diameter O-rings (0.455 in. vs 0.500 in.)
  - New O-rings will be installed

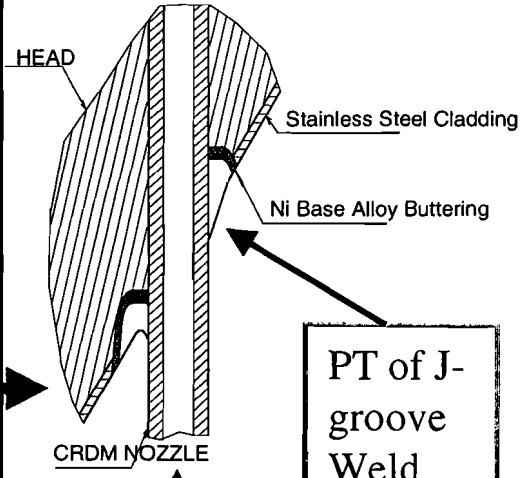


Examinations of Replacement RPVCH

RT and PT of CRD Nozzle to Flange Weld

RT, UT, and MT of Flange to Dome Weld





PT of J-groove Weld

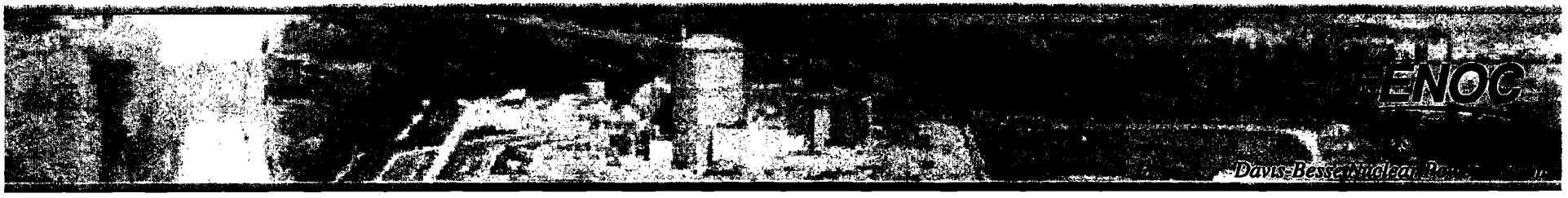
UT and ET of CRD Nozzle

Examinations of Replacement RPVCH

- Examinations to supplement ASME Code Data Package:
  - Visual examinations
  - Radiography (RT) of flange-to-dome weld
    - Lifting attachments prevented full coverage
  - RT of nozzle-to-flange welds
  - PT examination of the CRDM nozzle J-groove welds

Examinations of Replacement RPVCH

- Preservice Inspections
  - Magnetic Particle (MT) examination of flange-to-dome weld
  - Ultrasonic (UT) examination of flange-to-dome weld
  - Liquid Penetrant (PT) examination of peripheral CRDM nozzle-to-flange welds

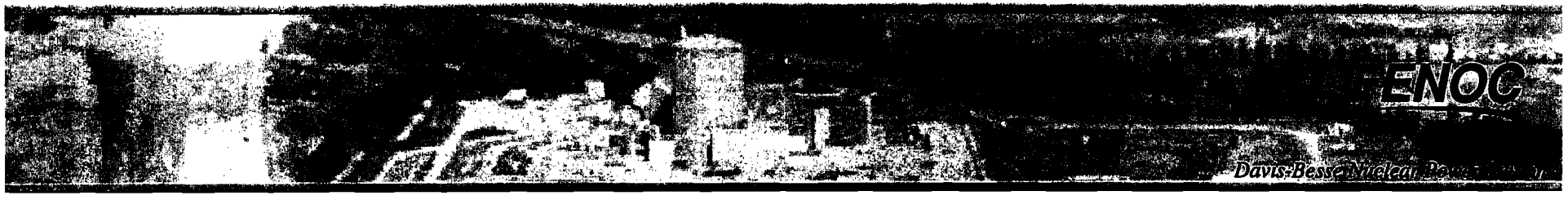


Examinations of Replacement RPVCH

- Additional Non-Destructive Examinations
  - Chemical smears
  - Baseline UT of CRD nozzles
  - Eddy Current Testing (ET) of CRD nozzles

Installation of the Replacement RPVCH at Davis-Besse

- Davis-Besse Containment Building will require temporary access opening
- Original RPVCH will be moved outside Containment Building for storage and/or disposal
- Davis-Besse Service Structure will be used
- Inspection ports will be installed on replacement support skirt



Installation of the Replacement RPVCH at Davis-Besse (continued)

- Original Davis-Besse control rod location and core configuration will be used
  - Existing CRD Mechanisms will be used
  - CRD Mechanisms nozzle flange split nut ring modification will be performed
  - Upgraded gasket design will be incorporated

Root Cause Investigation

Steve Loehlein
Root Cause Investigation Team Leader

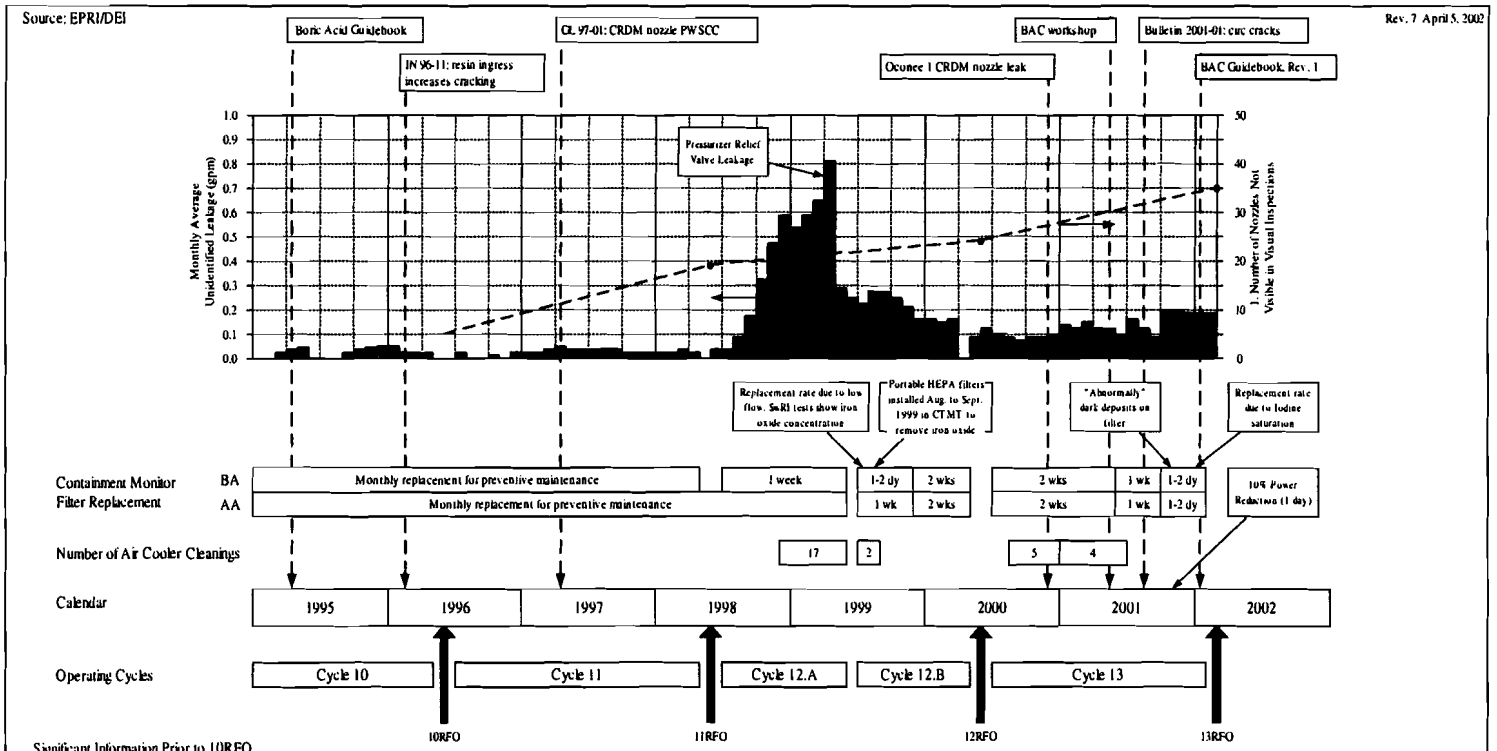
Key Questions

- Was there a new mechanism that caused this degradation?
- Was there adequate guidance/knowledge available to have prevented the degradation to the RPV closure head?

Key Conclusions

- The degradation to the RPV closure head was caused by Primary Water Stress Corrosion Cracking (PWSCC) of the Control Rod Drive (CRD) nozzle which led to leaks that were undetected allowing boric acid corrosion to occur
- The existing guidance/knowledge was adequate for preventing unacceptable RPV closure head degradation from CRD nozzle leaks

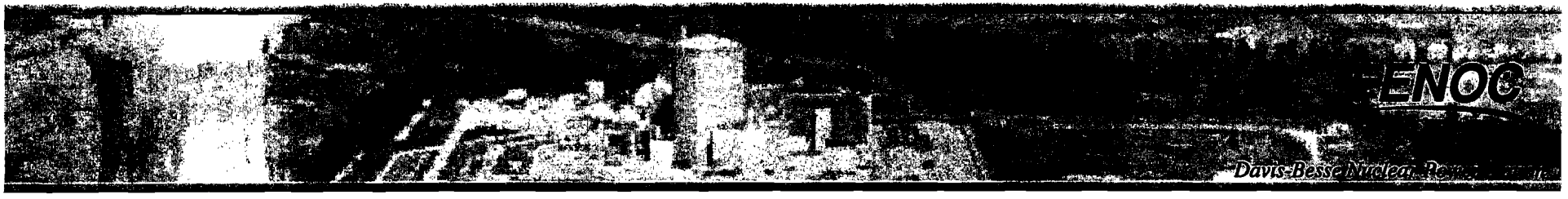
Timeline of Key Events



Significant Information Prior to 10RFO

1) Nozzle 66 gasket replaced at 7RFO and 8 RFO		1) Nozzle 31 gasket leaks but not changed 2) BA is powder and white	1) Replaced gaskets on flanges 3, 5, 6, 11, 31 2) "Red" deposits under nozzle 3 flange 3) Bottom of nozzle 3 flange not inspected due to hard BA	1) No leakage from flange gaskets 2) "Red" deposits under nozzle 3 flange 3) Difficult to get camera in place due to hard, brown, boric acid	CRDM Flange Conditions
1) Head flange clean in 1994	1) BA flowing towards mouseholes in S-E quadrant	First indication of red colored boric acid deposits from mouseholes in quadrant of S-E quadrant	Significant flow of red colored boric acid deposits from mouseholes, piling to 6" deep behind studs in S-E quadrant	1) Significant flow of boric acid deposits from mouseholes in S-E quadrant 2) BA deposits extend around entire flange 1" deep max	RPV Flange Conditions
1) BA deposits dripping through insulation at 8RFO 2) Head "cleaned" at 8RFO 3) Head not inspected during 9RFO	1) Visual inspection 65/69 nozzles 2) Minor BA deposits around outer nozzles of RV Head 3) BA deposits at center of head	1) Visual inspect 50/69 nozzles 2) Loose BA piling up behind outer nozzles 3) BA brittle, breaks easily	1) Visual inspect 45/69 nozzles 2) BA definitely red and hard to break up (needed crowbar) "lava-like"	1) Visual inspect 34?/69 nozzles N-1: leak (Nozzles 5 & 47 cracked) N-2: leak, circ & minor wastage N-3: leak & major wastage	RPV Head Conditions

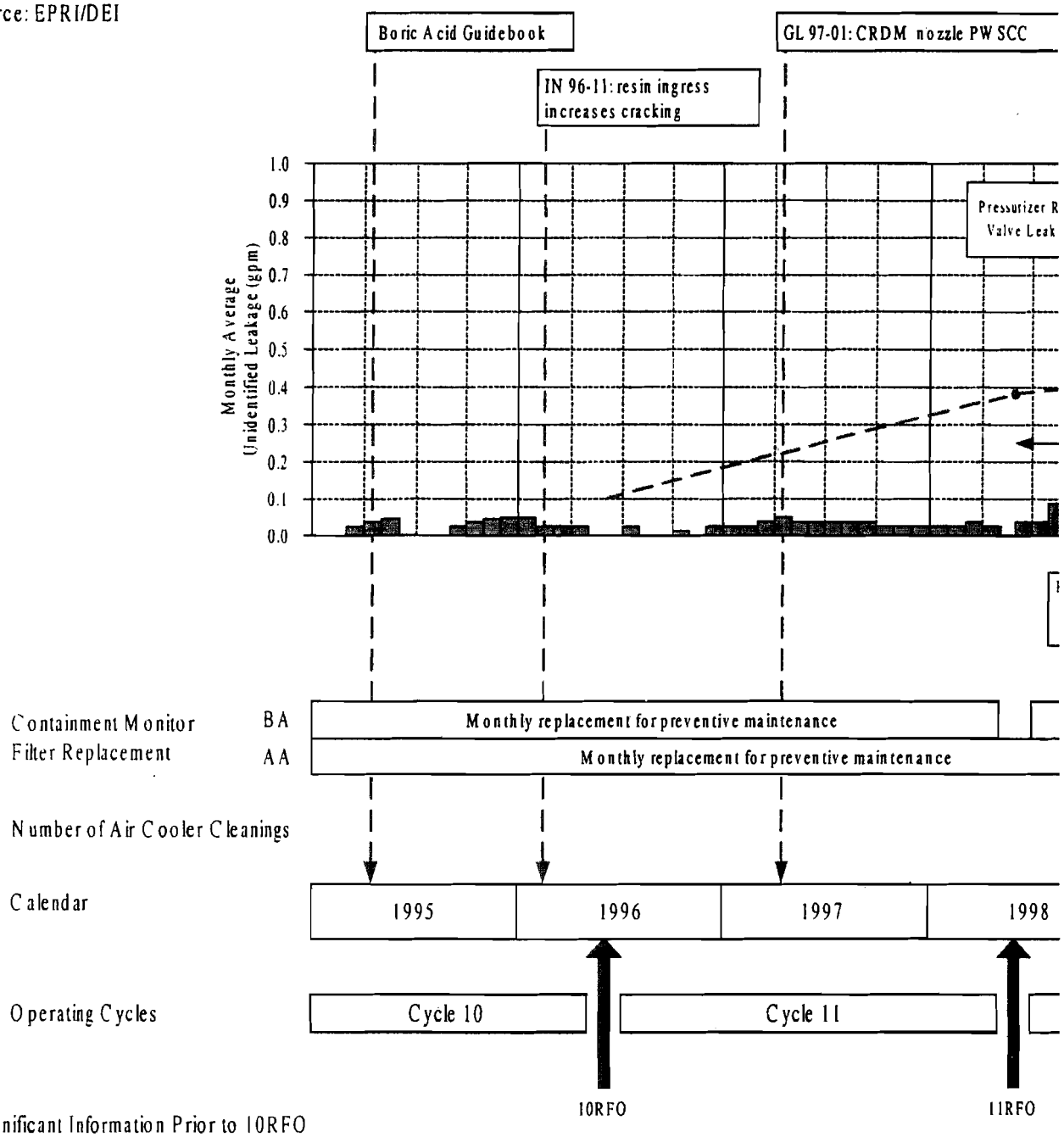
Figure 26. Timeline of Key Events Related to Reactor Vessel Head Boric Acid Corrosion



Concluding Remarks

FENOC

Source: EPRI/DEI



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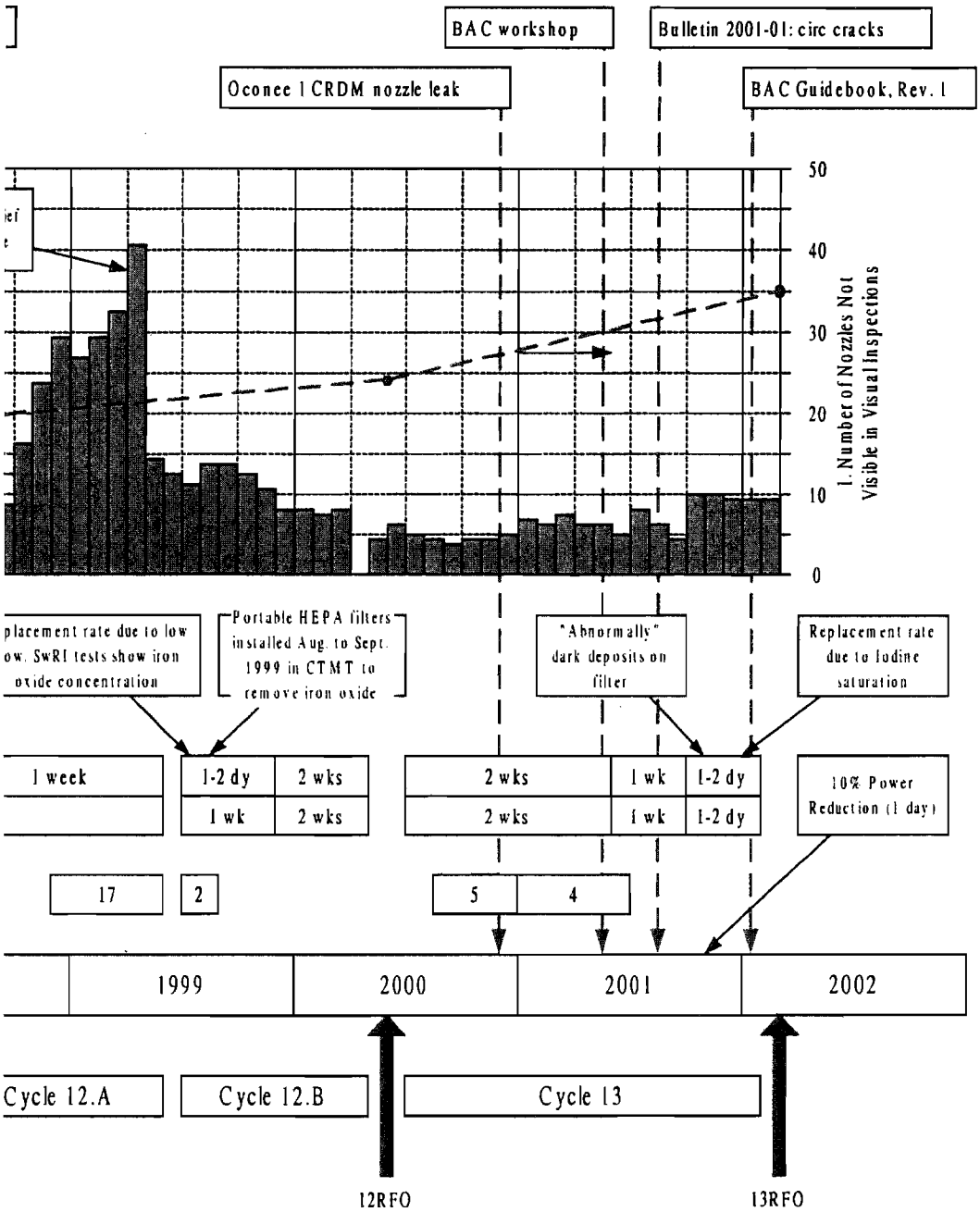

hanged	1) Replaced gaskets on flanges 3, 5, 6, 11, 31 2) "Red" deposits under nozzle 3 flange 3) Bottom of nozzle 3 flange not inspected due to hard BA	1) No leakage from flange gaskets 2) "Red" deposits under nozzle 3 flange 3) Difficult to get camera in place due to hard, brown, boric acid	CRDM Flange Conditions
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Figure 26. Timeline of Key Events Related to Reactor Vessel Head Boric Acid Corrosion

NRC Staff Activities on Vessel Head Penetration Nozzle Cracking and Vessel Head Degradation

Bill Bateman,Branch Chief
Materials & Chemical Engineering Branch
301-415-2795

ACRS Full Committee Presentation
June 6, 2002

Ongoing NRC Regulatory Activities at Davis-Besse

Davis-Besse Nuclear Power Station

(J. Grobe, RIII)

(5)

Ongoing NRC Regulatory Activities at Davis-Besse

Davis-Besse Nuclear Power Station

(J. Grobe, RIII)

Ongoing NRC Regulatory Activities at Davis-Besse

Implementation of IMC 0350 at Davis-Besse

- **Reactor Vessel Head Degradation represents a significant and complex technical and regulatory issue**
- **Plant is in an extended shutdown with a regulatory hold in effect (CAL)**
- **IMC 0350 enhances the agency's focus on clearly defining and addressing plant specific issues prior to restart**
- **IMC 0350 provides focused and coordinated regulatory oversight of Davis-Besse**

Ongoing NRC Regulatory Activities at Davis-Besse

IMC 0350 Panel Goals

- Provide oversight and assessment of licensee performance during the shutdown and through restart
- Assure that restart issues are identified and resolved
- Integrate and prioritize agency resources to maximize agency effectiveness and minimize regulatory burden
- Provide a single focus to ensure consistent and effective communication with external stake holders

Ongoing NRC Regulatory Activities at Davis-Besse

IMC 0350 Panel Goals

- Continue oversight after plant restart until plant is returned to the routine Reactor Oversight Process

- Create a comprehensive public record of agency decisions and actions

Ongoing NRC Regulatory Activities at Davis-Besse

License submitted Return to Service Plan - May 21, 2002

- Reactor Head Resolution Plan
- Containment Extent of Condition Plan
- System Health Assurance Plan
- Program Technical Compliance Plan
- Management and Human Performance Excellence Plan
- Restart and Post-Restart Test Plan

Ongoing NRC Regulatory Activities at Davis-Besse

Current Inspections

- **AIT follow-up (May - June)**
- **Vessel Head Replacement (May - September)**
- **Extent of Condition - Boric Acid (May - August)**

Technical Assessment of GSI-189

Advisory Committee on Reactor Safeguards

JUNE 6, 2002

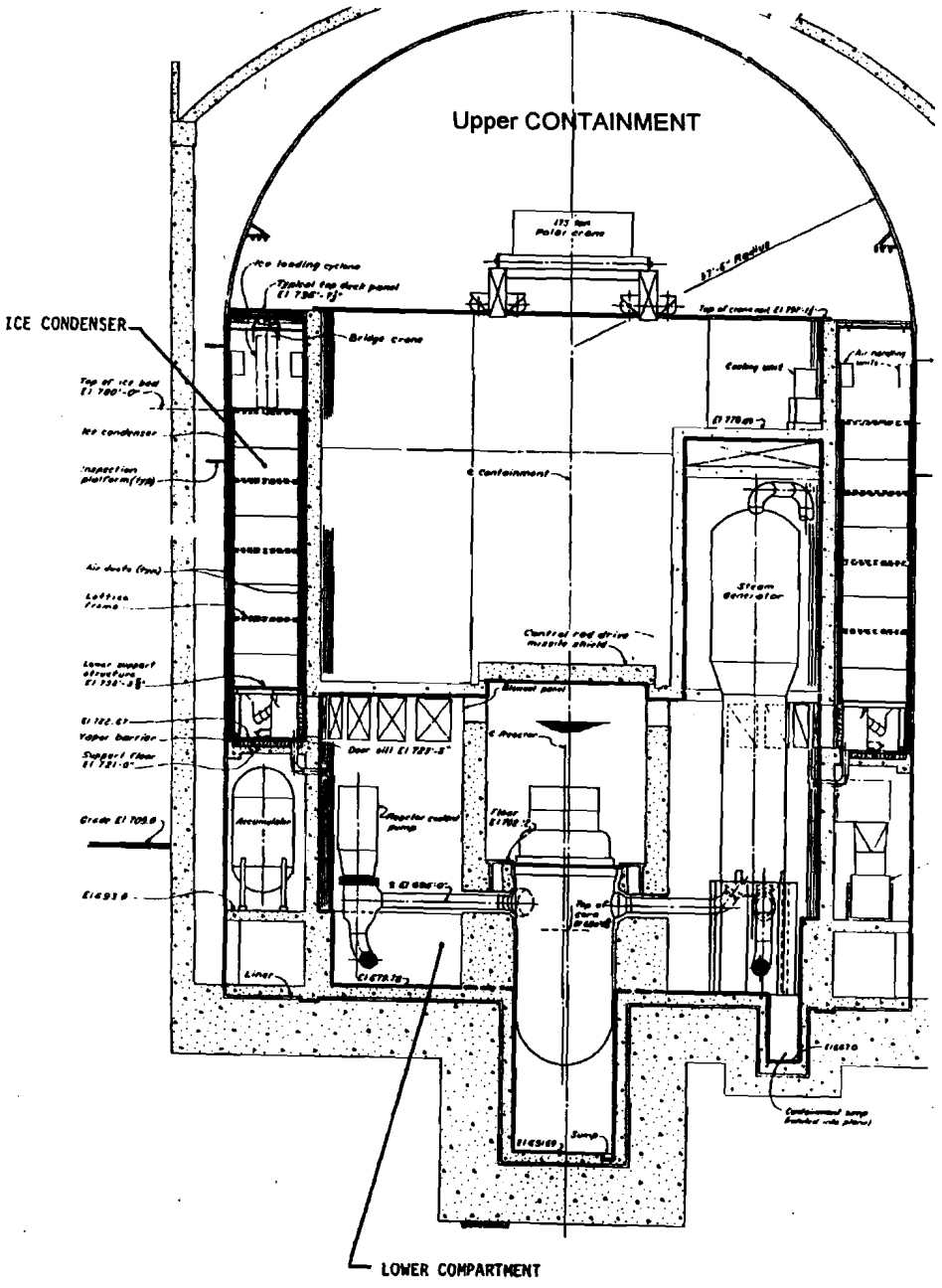
Allen Notafrancesco, Task Manager
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

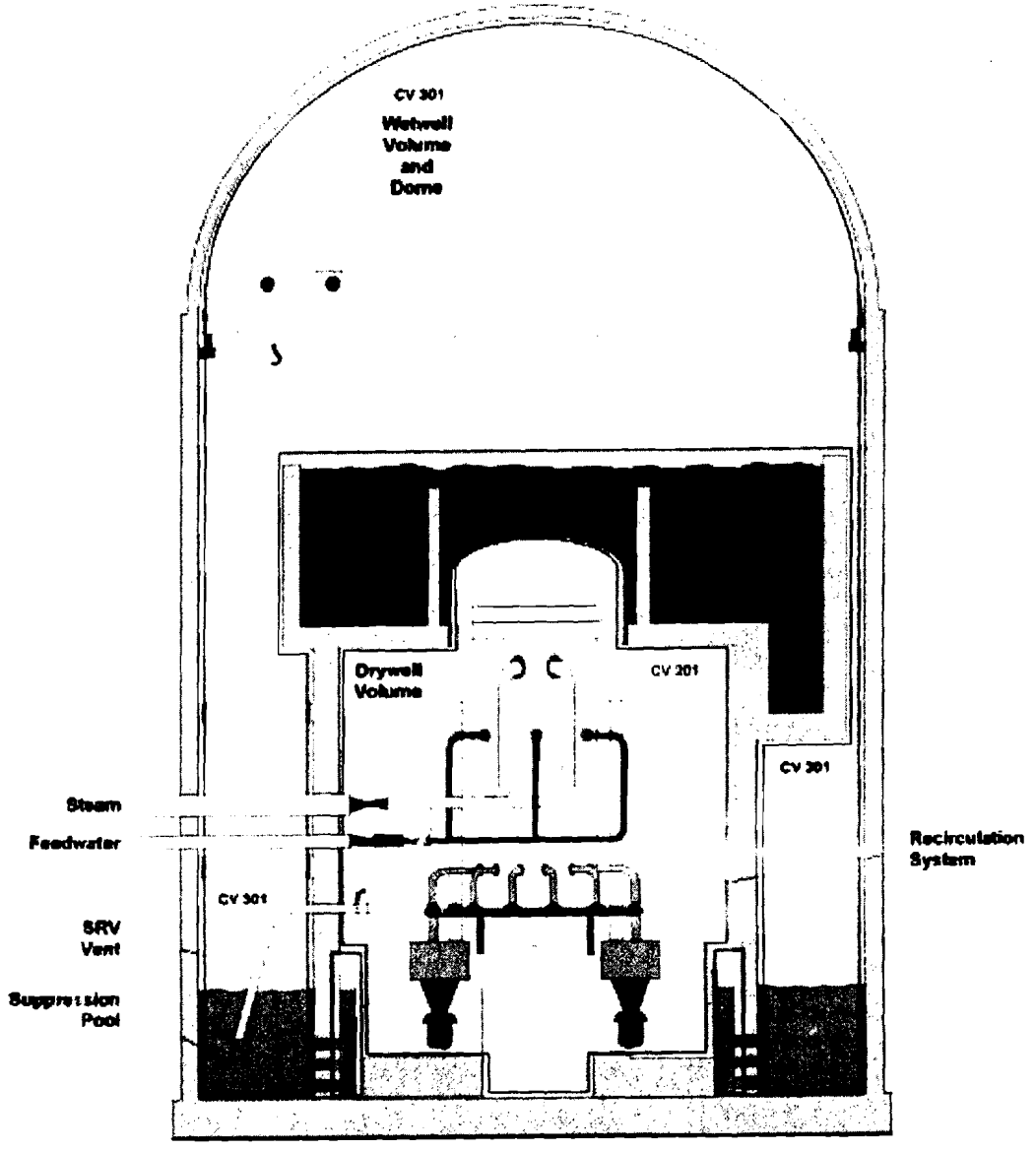
6

GENERIC SAFETY ISSUE 189: "SUSCEPTIBILITY OF ICE CONDENSER AND MARK III CONTAINMENTS TO EARLY FAILURE FROM HYDROGEN COMBUSTION DURING A SEVERE ACCIDENT"

- **Risk-Inform 10 CFR 50.44:**
  - "Status Report" in SECY-00-0198, September 14, 2000
  - "Staff Plans" in SECY-01-0162, August 23, 2001
  - SRM, December 31, 2001.....Resolve GSI-189 Expeditiously

- **GSI-189:**
  - MD 6.4, Passed GI Screening- February 2002
  - Generated Task Action Plan
  - Currently Completing Technical Assessment


Upper CONTAINMENT
ICE CONDENSER
LOWER COMPARTMENT


CV 301
Wetwell Volume and Dome
Drywell Volume
CV 201
CV 301
CV 301
Steam
Feedwater
SRV Vent
Suppression Pool
Recirculation System

GSI-189 TASK ACTION PLAN

- Approach for expeditious resolution:
  - Use Existing Studies
  - Assemble Support Team w/contractor assistance
    - Cost Analysis; REAHFB w/ISL
    - Benefits Analysis; PRAB w/BNL
    - Plant Analysis; SMSAB w/SNL

RESPONSE TO EXTERNAL EVENTS

- PRE-STAGED DESIGN

  -- If designed for external events, cost would double

    -- Expect survivability to be available for a reasonable subset of external events

- LOW COST OPTION

  -- No permanent structure and set-up would occur after the initial impact of the external event

  -- Portable diesel may come from multiple diverse locations

    -- These attributes should allow for increased likelihood of functionability in response to external events

BENEFIT ANALYSIS

ICE CONDENSER CONTAINMENTS:

As part of license renewal, severe accident mitigation alternatives (SAMAs) are to be considered;(from NUREG-1437,Supplements 8 & 9)

- Benefits for Hydrogen Control in SBO events, all early failures are averted:

Plant	Based on most recent plant specific PRA*	Based on conditional containment failure probabilities from NUREG/CR-6427
McGuire	$178 - 248K	$678K
Catawba	$236 - 329K	$387K

* Variation in costs due to sensitivities in discount rate (3-7%) and years of remaining life (20-40 years)

- BNL report confirmed that using NUREG/CR-6427, averted costs are somewhat greater

BENEFIT ANALYSIS

OTHER MARK III CONTAINMENTS:

Plant Name	SBO frequency ratio	2000 population ratio (estimated)	Product of ratios
Grand Gulf	1.00	1.00	1.00
Clinton	1.31	3.14	4.13
Perry	0.30	7.53	2.27
River Bend	1.81	3.14	5.68

Therefore, benefits for Hydrogen Control in SBO events, could exceed $200K (40 yrs @ 7% discount rate)

COST BENEFIT SUMMARY

(1) Focusing on the plant specific averted cost estimates for ice condenser plants, the low cost equipment option is clearly cost beneficial and provides for more overall flexibility in response to external events.

(2) For Mark III plants, the low cost back-up power supply option is marginally cost-beneficial.

(3) Since the proposed mitigative enhancement passes the back-fit cost beneficial test, we recommend that further regulatory action is warranted.

MELCOR SCOPING STUDY CONCLUSIONS

- Igniters alone are effective in controlling Hydrogen build-up
- Marginal improvement if one ARF is included, however fans accelerates time of ice melt-out
- Continuing ice condenser plant response uncertainty study, e.g., H2 source terms, burn propagation, etc.

Response to Letter to ACRS Dated June 3, 2002

Consideration of Adding Back-Up Power to ARF...Basis:

- Arbitrary assumptions of limiting conditions & sequences of events are extreme
- Ease of DDT occurring in the ice chest is not demonstrated

Why Back-Up Power to Igniters Only are Sufficient....

- 30+ igniters distributed through out containment will assure lean ignition and promote diffusive burning

June 6, 2002

Advisory Committee on Reactor Safeguards
Nuclear Regulatory Commission

Chairman George Apostolakis

Members of the Committee

My name is Ann Harris. I have traveled here today by my personal resources without benefit of taxpayer support or government payroll. I appeared before this committee in November, 1995, prior to your support to the Commission for the licensing of TVA's Watts Bar nuclear plant. I moved out of the evacuation zone to a nearby area.

The fact that we are all here again seven years later to hear staff's offering on the Generic Safety Issue 189 and NRC's recommendation is evidence of how things work with Staff and the industry. The Ice Condenser issue may be a generic issue to you, but you should be aware that it is real people's lives you are talking about. This is not a generic issue to me. It is about the nuclear reactors just down the road from where I live and where members of my family and friends live.

I hope that you are as worried about the time factor as I am. I take it as a positive sign that at least something is going to be done even if it is only talk this time. But do we need more talk? I was in this same room seven years ago arguing that Watts Bar was not ready for prime time. That didn't do any good since most of the problems were never fixed, just forgiven. Will we be back talking seven years from now when TVA and Staff admit that safety is still not a prime factor? I think not. TVA will be in the nuclear weapons production business at Watts Bar

and Sequoyah because Staff has never seen an industry license amendment request it did not like.

At the meeting in 1995, one of the subjects I heard about was whether the hydrogen igniters would work. My transcript of that meeting shows that committee member Ivan Catton tried to raise questions about hydrogen igniters and whether the igniters at Watts Bar were adequate to prevent the containment from leaking from hydrogen explosions. In fact, he was asking questions about whether the igniters were located in the right locations in the containment, and now here you are seven years later talking about the same thing. These meetings are like seven year locust visits. They just keep coming.
Committee Members, talking just isn't good enough anymore. Your talking has put lives at stake. It appeared at that 1995 meeting that Mr. Catton was truly interested in whether Watts Bar was safe enough but he was cut off and shut up by the chairman at that time.
What we did not know at that meeting was that the person at Watts Bar responsible for making sure the ice condenser was working correctly before startup had discovered that the screws holding the ice baskets up were defective. TVA devised a scheme to hide Curtis Overall's discovery, then get rid of him, therefore obtaining the Watts Bar license by lying to this Committee and the Commission. After years of investigations and court proceedings the NRC has been forced to levy a fine against TVA. TVA has had so many fines for employee abuse they shed them like water off a duck's back. No big deal!

The most troubling fact is that inspections of the ice baskets that Overall wanted (and was abused for) were never done. We still don't know if they'll stay put if there is an accident at the plant. I have never told anyone that I am an engineer. But I do have common sense. From what I understand, NRC seems to be finally facing up to the fact that ice condensers won't really work------won't protect the public during an accident. Their idea to fix the problem is to get a little portable generator, from Home Depot or Lowe's, put it on a pick up truck, roll it up to containment, and plug it in. I worked in TVA's nuclear program for 16 years, fourteen at Watts Bar. I have seen some crazy, silly, childish and outlandish things done in the name of safety but I believe this one could take the blue ribbon! I keep having this cartoon run through my head of what would be going on if this generator is needed. There's a hurricane, a severe lighting storm, a terrorist attack, a flood, it is dark, no lights, and no backup power. The shift supervisor has just sent someone to the little shed outback containing the Honda generator with a copy of the combination to the padlock. People living down stream are depending upon this person to know the combination without hunting the paper it was written on. The rain is wetting the paper. His glasses are covered with water. The wind blows the paper away and he starts back inside for another copy. When he gets back, he unlocks the shed, rolls the generator to the containment building, plugs it in and proceeds to get it running. I think that our lives and our property values deserve a little more concern than this NRC proposal. Why are you only recommending this blue light special approach?

I feel that the people who live near these plants are getting short-changed, run over and made expendable. The NRC recommendation seems to say the backup power doesn't have to

work if the accident is caused by a flood or an earthquake or a terrorist attack. How do you think this kind of accident is going to happen? Merlin conjuring?

Committee Members, the people living in these communities are real live people whose lives are being talked about here this morning, not just numbers and statistics. Those same people trust the NRC to protect their interests.

I wouldn't be surprised if NRC gets pressure from industry about making changes to the ice condensers to make them actually work. I imagine you will be pushed to pick numbers to re-do your calculations making it impossible to solve the problem that fixes the containment. I am speaking as much to licensing people in the audience as well as this committee and research staff, to keep in mind the interests of the real people living near these plants. Think twice about trying to make industry happy with analyses that say they don't have to fix anything. It is good that NRC has made a start, but so many times good starts end up as dead ends.

I think you should be careful about plans to fix the ice condenser plants depending upon the good will, and good intentions, of the plant owner. Some of the proposed changes, like the cheap portable generator idea, seem to be planning on not having the inspections that you have for other safety equipment. I don't know about other utilities but I know TVA well enough to know that if NRC leaves it all up to them, the generator will not have a motor or a receptacle for the plug. If there's neither inspection nor enforcement, that backup system is not going to be there when it's needed. You see the bigger danger is to have a lot of back and forth talk leading people to think that something's being done to fix the problem. But you and I know that's not

true. And there in lies the problem, misleading is worse than doing nothing.
I would ask that you recommend to the Commission that these ice condensers be fixed to protect the public NOW! You should advise the staff that they should be bending over backwards to protect the public's safety, not bending over to avoid trouble from the industry.

Thank you

Respectfully Submitted

Ann P. Harris
341 Swing Loop
Rockwood, TN 37854
(865) 354-4559

By copy of this statement I request that my statement be made a part of the official record.

Kenneth D. Bergeron, PhD
17 Tierra Monte NE
Albuquerque, NM 87122
e-mail: kenberg@flash.net

June 3, 2002

Advisory Commission on Reactor Safeguards
ATTN: John T. Larkins, Executive Director
U.S. Nuclear Regulatory Commission
Mailstop T-2 E26
Washington, DC 20555

Dear Members of the Committee,

Subject: ACRS Review of NRC/RES Proposal on Resolving Generic Issue 189

I have been involved, in one way or another, in the issue of the vulnerability of ice condenser PWRs to Station Blackout events (SBOs) for almost 20 years, beginning with work on NUREG-1150, followed by my managing Sandia's support to NRC's Containment Performance Improvement program in the late 1980s, and continuing with my involvement in the effort to resolve the Direct Containment Heating (DCH) issue for ice condensers in the late 1990s. In regard to this last effort, I was a co-author of "Assessment of the DCH Issue for Plants with Ice Condenser Containments," NUREG/CR-6427, and I carried out the CONTAIN code calculations for the final version of that report. Shortly thereafter, in 1999, I retired from Sandia (though I imply no cause and effect relationship there). Since then I have followed with interest NRC's efforts to deal with the remarkable vulnerability of ice condenser containments to SBOs.

In his transmittal of NUREG/CR-6427 to NRR, Ashok Thadani, NRC's research director, concluded that the study had, in effect, closed the DCH issue for ice condensers but had also brought to light the high vulnerability of these plants to containment failure (primarily from hydrogen combustion) in SBO sequences. He suggested at the time that this vulnerability be addressed through the ongoing efforts to risk-inform 10 CFR 50.44. I will admit to some skepticism about his suggestion at the time, since the vulnerability in question had been well known for at least fifteen years. But NRC's research effort on Generic Safety Issue 189 (GI-189), recently made public by Farouk Eltawila's May 13 memo to the ACRS, has impressed me and my skepticism has abated somewhat.

It has been less than a year since the NRC's Executive Director for Operations announced to the Commission in SECY-01-0162 the establishment of GI-189 to deal with the vulnerability of ice condensers and BWR Mark IIIs to SBOs, and less than six months since the Commission established a high priority on resolving it. The RES staff has

accomplished a great deal within these compressed timeframes and should be congratulated on their efforts (and those of the tireless contractors at ISL, BNL, and SNL). Their preliminary work has set the stage for an expeditious resolution of GI-189 and, hopefully, a timely implementation of the needed containment improvements to ice condenser PWRs.

I would very much have liked to attend the ACRS meeting at which GI-189 is to be discussed (June 6-8, 2002) and present my views on the ice condenser-related issues, but previous family commitments during that time frame make that impossible. In this letter, I will summarize some of my viewpoints. David Lochbaum of the Union of Concerned Scientists has offered to present some highlights of my assessment at the meeting, for which I am greatly appreciative, though I hasten to take responsibility for any errors in my review (political or otherwise).

My most important comment is that, while the contractor reports provide a basis for supporting a variety of the backfit options that were evaluated, Mr. Eltawila's letter appears to recommend only the low-cost portable electric generator to power only the igniters. I have long believed that a fix to ice condenser vulnerability must include, in addition, power to the air return fans. Nothing in the NRC's recent analyses has changed that belief. I will provide a detailed explanation of my views below, but I would first like to put that discussion in context by stepping back and looking at some overarching issues related to containment improvements.

Why are containment improvements so problematic for the NRC?

It has become apparent to me over the course of my many years of support to NRC that there is an asymmetry between what can be accomplished in the way of improved power plant safety in the 'front end' versus in the 'back end.' There has been no shortage of conflict between the regulator and the industry over improvements to the primary system and associated safety systems, but over time there have been many important safety upgrades as well. Compared to before TMI, we have better pump seals, better system reliability, better steam generators, better water chemistry, and the list can go on and on. By contrast, little has been done to improve containment performance, at least in the past decade or so.

Certainly there have been many changes related to the containment, and some have been based on risk analysis. But almost all such changes are some form or another of regulatory relief. Certainly there have also been innumerable cases of straight regulatory relief for the Nuclear Steam Supply System and its associated support systems, but in addition there have been many 'front end' changes that cost money but improve safety. And those changes continue. Not so for the containment system, in my belief. With very few (and minor) exceptions, risk-based changes to the containment and its operations have been strictly for relief, not improvement. It has become a one way street.

I believe the reason for this difference is simple. The plant owner has much more of a shared interest with the regulator in avoiding conditions that lead to severe accidents.

Doing so not only protects the public, but it also preserves the power plant as a source of revenue to the owner. As with TMI-2, a severe accident almost assuredly ends the life of the reactor as a productive capital asset regardless of whether the containment fails or not. This simple fact gives the licensee a double incentive to work with the regulator to reduce the core damage frequency. There is no such double incentive to reduce the conditional containment failure probability.

I think that at one time the NRC implicitly recognized this difference when it introduced the Containment Performance Improvement program in the 1980s. However, the virulence of industry's reaction to proposed improvements to BWR Mark I containments brought that program to an untimely end around 1990, and since then there has been no significant action, as far as I know, related to improvements in containment performance. That is, until Mr. Eltawila's May 13 memo.

I do not say this to put the industry in a bad light. I say it because it is important for NRC to take account of this asymmetry in its approach to containment improvements. It is inevitable, given the differences in industry's incentives, for it to be more difficult to gain industry consensus on containment improvements than on improvements to the front end. This doesn't mean the NRC should be satisfied with less progress. Defense in depth is important, and the meaning therefore is that NRC should work harder to accomplish such improvements, expect more resistance from licensees, and bring a firmer resolve to the deliberations.

Having now exercised my private citizen right to express my views, soapbox style, I will turn to the RES recommendations on GI-189.

The low-cost backfit option recommended by RES is inadequate and possibly counterproductive.

The Eltawila memo provides in its three attachments a preliminary technical basis for evaluating potential backfits to ice condenser containments (as well as Mark III containments, but I will have no comments on the Mark III issue). Attachment 1 is a brief cost study by Information Systems Laboratories, Inc. (ISL) of four basic options for adding equipment to provide backup power to containment safety systems during SBO conditions. Attachment 2 is an assessment by Brookhaven National Laboratory (BNL) of the averted costs (or benefits) that might accrue from implementation of a containment fix for the SBO vulnerability, which evaluates the dollar value of the benefits for a matrix of cases involving different plants and different analysis assumptions. Finally, Attachment 3 is a very preliminary report from Sandia National Laboratories (SNL) on MELCOR code calculations of the effectiveness of some of the backfit options.

Certainly one thing that is needed is a more integrated presentation of these disparate results, and I would hope that RES plans to prepare such a report in the future. However, given the time pressure for addressing GI-189 I can understand the decision to make available the research results in this fragmented form at this time.

By studying the BNL and ISL reports together, one can see that a number of the backfit options for ice condensers easily pass a cost benefit test for a range of analysis assumptions. For example, in Table 2-12 of Attachment 1 we see that the option involving pre-staged emergency backup power for igniters and air return fans has a total cost of $313,300 for a single unit of a dual unit ice condenser plant. Table 6 of Attachment 2 gives lifetime benefits ranging from $404,000 to $6,730,000, depending on analysis assumptions. That it is cost beneficial to fix the ice condenser containments is not surprising to me, since the inadequacy of these plants in SBO conditions is notorious, and the risk significance of SBOs is high (at least for some plants). What is surprising is that Eltawila's memo, which is the only place any integration of the three reports occurs, seems to recommend only the low-cost option labeled "1b," which involves an off-the-shelf portable generator to provide backup power to the igniters only. Several other options that also have favorable cost/benefit numbers are not mentioned in the recommendations. Knowing that there will be resistance from industry to any of these proposals, I find it hard to understand why RES would choose to endorse only the 'el cheapo' option.

More important, I believe the low-cost option is inadequate to deal with ice condenser vulnerability to SBOs. In this, I agree with Duke Power's decision in their SAMA submittals on McGuire and Catawba to evaluate backup power to igniters only in conjunction with backup power to the air return fans. At a public meeting between the Nuclear Energy Institute and the NRC in September 2000,* industry representatives criticized NRC's assertion that powering igniters only would be sufficient—they said you need to power *both* igniters *and* fans to control the hydrogen burn threat in ice condensers, and they were right.

Far too little importance has been attributed in the RES analysis to the possibility of detonations in the ice chest. What I am worried about is the possibility that in the absence of forced mixing via air return fans, a typical SBO scenario would lead to the following conditions: very high hydrogen concentrations (say, over 20%) throughout much of the ice chest; quite low concentrations (under 5%, say) in the upper plenum and containment dome because there has been little leakage through the upper deck doors until this point in time, and steam inerted conditions in the lower compartment. Then, when the concentration in the upper plenum finally becomes combustible, one or more burns occur there, resulting in upper deck doors opening and closing, perhaps in succession many times, which brings out a plume of much more combustible gas into the upper plenum from the ice chest. This plume would then ignite and carry the flame back to the ice chest, where the deflagration would transition to detonation because of the wide variety of channeling and reflecting surfaces there. A global detonation over most of the ice chest is something I don't even want to think about. Containment failure could occur either through missile generation or dynamic overloading of the containment structure.

* "SUMMARY OF SEPTEMBER 28, 2000, PUBLIC METING WITH NUCLEAR ENERGY INSTITUTE (NEI) AND OTHER INTERESTED STAKEHOLDERS REGARDING RISK-INFORMED CHANGES TO 10 CFR 50.44", Memo from Alan S. Kuritzky, NRC/PRAB to Mark A. Cunningham, chief NRC/PRAB dated February 28, 2001.

I am not the first person to worry about this scenario, of course, but my point is that I don't think today's calculational tools can help NRC develop the confidence it needs to order a mandatory igniters-only backfit. I am glad NRC has commissioned Sandia to carry out a broad matrix of code calculations on hydrogen distribution, because they shed light on the overall issue, but there are some aspects of the problem that I believe defy accurate computational analysis. This means that even an expanded program of calculation will still leave substantial residual uncertainty about the potential for highly destructive detonations. The aspects that are problematic are the following:

1. Control volume codes like MELCOR and CONTAIN are not suitable for predictive modeling of natural convection in open regions that are larger than the characteristic dimensions of the circulation patterns. The governing equations ignore the convection of momentum, for example. Such codes are notoriously incapable of reliably modeling stable stratification, because they artificially diffuse mass in the absence of any driving force (an effect that some people call numerical diffusion). Results are highly sensitive to nodalization, far more so than with true Navier-Stokes solvers. For these reasons I have always been privately skeptical of NRC's use of control volume codes to address hot leg failure due to natural circulation in SBO-induced core meltdowns. It is true that skilled analysts can devise nodalizations that reproduce flow patterns resembling those seen in experiments, but I don't think that leads to *predictive* capabilities for shapes and scales far different from the tests.
2. The unique phenomena occurring in ice chests make such calculations even more uncertain. Hot air/hydrogen/steam mixtures will be affected dramatically and in numerous ways by the ice: first through condensation of steam, which creates a bulk flow towards the ice surface called Stefan flow; second through cooling, which affects buoyancy-related flow; and third through changes to the mean molecular weight of the gas mixture due to condensation of steam, which is intermediate in molecular weight between the lightest of the gases, hydrogen, and the heavier gases that are the principal components of air (nitrogen and oxygen). These various effects will either reinforce each other or oppose each other, depending on conditions. And the processes are occurring over a complex spatial distribution of ice surfaces, not just a simple boundary.
3. One might argue that these effects will serve to increase mixing compared to the corresponding hydrodynamic problem in the absence of ice, but it is also possible that under some conditions the effects might be to stabilize stratification, inhibit the formation of large convective loops, and in general reduce vertical mixing. Similarly, the pressure pulses originating from releases from the primary system are dampened by the effects of the ice chest to the extent that the upper deck doors do not open as often or as far as would be expected in the absence of ice. The result is increased isolation of the upper plenum (where the igniters are) from the ice chest (where most of the hydrogen is if the lower containment is steam inerted).
4. The problem is further complicated by the fact that the ice/gas boundaries are, over time, responsive to the gas flow, resulting in highly uncertain spatial

configurations of unmelted ice. The Finns have reported considerable unevenness of melting in their ice condenser test facilities when mixing processes are weak. The industry has long discussed 'channeling' and 'melt-through' as issues related to the design basis accident—one of the important roles of the air return fans is to insure relatively uniform melting.

These difficulties combine to create very large uncertainties in code predictions of hydrogen concentrations, with or without power to the igniters. With backup power to the air return fans, the NRC has a wonderful opportunity to reduce the uncertainty, and at a cost much cheaper than building a code that can accurately calculate the problem!

When I argue that there is residual uncertainty, I am not saying that there might be a 1% or 10% residual failure probability; I am saying that this fix might be ineffective 50% or 90% of the time. There is little way of knowing what its effectiveness will be *unless you ensure mixing.*

The low-cost option endorsed by RES might even make the accident worse, by causing a detonation-induced containment failure to occur many hours earlier than it might have in the absence of forced ignition. I would guess that the effect of this possibility on the risk picture would be modest, but I don't know. I suspect RES would not feel very comfortable asking one of its contractors to evaluate this downside of a mandated backfit—it could very possibly be used as an excuse to oppose the change. As with medicine, the first rule should be 'do no harm.'

The arguments in favor of providing power to the air return fans are compelling, I believe. The cost benefit numbers look good if reasonable assumptions are made about averted costs. The fans are already there, and they play the role of allowing the ice and the igniters to successfully accomplish their functions. Moreover the potential that the accident is exacerbated by the 'fix' is substantially eliminated.

I strongly encourage the ACRS to endorse the overall approach RES has initiated, but to insist on the additional assurance provided by backup power to the air return fans in ice condenser containments.

<u>The claim in the Eltawila memo that analyses in NUREG/CR-6427 are bounding is false.</u>

Page 3 of the Eltawila includes the statement "Note that in Attachment 2, the ice condenser averted cost estimates used relevant information from NUREG/CR-6427, and it appears to provide upper bound estimates as compared to plant-specific best estimates." This statement is untrue and misleading.

NUREG/CR-6427 was part of a long program (initiated around 1992) intended to resolve the DCH issue at U.S. nuclear power plants. Throughout that program the approach was best estimate and plant specific. Having been intensely involved in the ice condenser

DCH project for several years, and as a co-author of NUREG/CR-6427, I can say that there was a mix of assumptions used for the analyses that ranged from conservative to optimistic, and the end results can't be characterized as occupying any particular point in the spectrum.

For example, the plant-specific containment fragility curves (which dramatically affect the bottom line, of course) were taken from industry IPEs, which have never been audited by the NRC and which are almost certainly optimistic. (I say this because NRC's test program on containment failure has amply demonstrated the sensitivity of code results to nodalization details concerning penetrations, weldments, and other important locations for stress concentrations. It would not be expected that the industry's analysts would intentionally make assumptions that would exaggerate the potential for failure of their containments.)

In addition, certain assumptions about initial conditions in the core were also probably optimistic. Other assumptions, such as our treatment of steam spike, were probably pessimistic (or conservative), having been performed under severe time pressure. While I certainly wish that there had been time and funding to do more thorough analyses for the project, it is simply untrue to characterize the results as bounding or even conservative, regardless of how unpopular some of the results are with industry. The overall picture that emerged from NUREG/CR-6427 about ice condensers' vulnerability to SBO was qualitatively no different from the results of many earlier studies. It is because of that fact that GI-189 was established.

In the September 2000 meeting mentioned earlier, industry representatives complained that NUREG/CR-6427 was only a 'scoping study' and should not therefore be used as the sole basis for deciding on containment backfits. NRC responded that their recommendations were also based on other studies. Both sides were right. But it is inappropriate now for NRC to rewrite history by implying that the NUREG/CR-6427 was bounding in nature.

Final Observation.

While there is a clear need for additional study to support resolution of GI-189 (such as whether backfits that are not qualified for external event-induced SBOs would succeed anyway), I hope that the NRC proceeds into the implementation phase in a timely way. The RES staff has made a good start in establishing the technical basis for resolving this important issue. But I would like them to set aside their pre-conceived notions of what is the right answer and let the scientific facts speak for themselves. In this regard, I reiterate the following Observation from the 1998 ACRS review of the NRC Research program: "The Office of Nuclear Regulatory Research (RES) routinely relies on "assumed" solutions to address technical issues."[1]

[1] Advisory Committee on Reactor Safeguards, USNRC, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program", NUREG-1635 vol. 1, p. 30 (June 1998).

There is an old saying, "if it's not broke, don't fix it." I'd like to propose a new version to the NRC: "if you're going to fix it, then *fix it!*"

I would be glad to discuss any questions the Committee or others at NRC might have in regard to these comments.

Sincerely,

Kenneth D. Bergeron

United States Nuclear Regulatory Commission

Technical Assessment of GSI-168 Environmental Qualification of Low-Voltage I&C Cables

Presentation to Advisory Committee on Reactor Safeguards

Office of Nuclear Regulatory Research
N. Chokshi and S. Aggarwal, Division of Engineering Technology
June 6, 2002

Purpose:

- To presents to the ACRS the technical assessment of GSI- 168 and request a letter from the Committee.

Process

- Technical Assessment is complete.
- RES plans to incorporate the ACRS comments.
- Forward the Technical assessment to Director, NRR by June 30, 2002 (copy to EDO).

Overview of the Technical Assessment:

Technical Assessment Based On:

1. EQ Literature Review
2. LOCA Tests
3. Condition Monitoring Tests
4. Interaction with the Nuclear Industry
5. Risk Insights

Test Results (Items 1- 3) were presented to the ACRS on October 6, 2000.

- The qualification criterion is "zero" failures based on testing of a single prototype.

STEPS IN LOCA TESTS

- Preaging: Thermal & Radiation- to simulate end of life conditions.
- LOCA Exposure: To simulate postulated DBE.
- Post LOCA Simulation Test: To demonstrate an adequate margin of safety by requiring mechanical durability.

Research Results:

- Failures of certain I&C cables in NRC tests.
- Failures of single conductor bonded Okonite cables.
- No single condition monitoring technique is effective to detect degradation- combination of techniques could be used.
- Visual Inspections proven to be useful.

RISK

- Results indicate that benefits from reducing the cable failure probabilities to zero are modest.
- The state- of- the art of incorporating cable failures into PRA is still evolving.
- KEY ASSUMPTION: Operating environments are lower than or equal to those assumed during qualification.
- Uncertainties arise from sparse data, uncertainty of applicability of experiments, and human error probabilities.
- Complete coupling between redundant trains was assumed.

Interaction with Nuclear Industry

- I&C cables have not experienced any significant aging. In limited cases of hot spots, several options are exercised (early replacements, modification of environments, or some kind of condition monitoring).

- Aging evaluation are ongoing throughout the plant life.

60 Years Aging Assessment

- In NRC tests 8 out of 12 cables failed the Post LOCA Simulation Test.
- Some of these cables may not have sufficient margins beyond the 40 years of qualified life.
- If the operating conditions are less severe than the qualification parameters, the margins could be used to extend the life.
- Knowledge of the environment for cables continues to be essential.

Summary of RES Findings:

- Qualification test programs include numerous conservative practices.
- Failures in NRC tests indicate that some cables did not meet qualification criteria and the margins are reduced.
- Knowledge of the operating environment for cables is essential.

Summary of RES Findings (Cont'd):

- Industry good practices, as described by NEI, seems to be adequate.
  - Plant specific practices are not known.
- Walkdowns to look for any visible signs of anomalies attributable to cable degradation have proven to be effective and useful.

RES RECOMMENDATION

- Dissemination of research results and other information consistent with the generic communication process, to include:
  - Results of Tests conducted on cables for service life of 40 & 60 years.
  - Summary of Okonite test results and subsequent NRC actions.
  - Summary of research findings on condition monitoring techniques.
  - Summary of the industry good practices for condition monitoring.
  - Importance of knowledge of operating environments for cables.

Revised Oversight Process
Performance Indicator Pilot Program
Mitigating System Performance Index

Patrick Baranowsky (415-7493)
Office of Nuclear Regulatory Research

June 6, 2002

Purpose/Outline

- The purpose of this presentation is provide an overview of the reliability/availability performance indicator pilot program for the revised reactor oversight process (ROP).
- Presentation outline
  - Background
  - Problems with current ROP PIs and how they are addressed by pilot program PIs
  - Insights from the phase 1 risk-based performance indicator (RBPI) study
  - Overview of technical approach and its limitations
  - Summary of issues raised by ACRS subcommittee on PRA
  - Conclusions/implementation schedule

Background

- SECY 99-007 stated the need to further refine reactor oversight process (ROP) performance indicators (PIs)
- During first two years of initial ROP implementation, staff and industry identified problems with current ROP PIs and made several changes
- It was decided to form a working group to address these problems and develop new/revised PIs
- Reliability and availability performance monitoring of selected mitigating systems is one aspect of the ROP refinement

Problems with Current ROP PIs

- Current ROP PIs include design-basis functions along with risk-significant functions
- ROP PI thresholds are generic
- Demand failures are not properly accounted for
  - Demand failures are treated in unavailability using fault exposure hours
  - Use of fault exposure hours can overestimate the risk significance of equipment performance degradation
- There are no PIs for support systems
  - Unavailabilities of support systems are cascaded onto the unavailabilities of monitored systems
  - Cascading support system unavailabililty results in overestimating the unavailability attributed to the performance of the monitored system

How are These Problems Addressed by PIs in the Pilot Program?

- Pilot Program PIs are based on risk-significant functions
- Pilot Program PIs account for plant-specific design/operating characteristics through use of available risk models and data
- Demand failures are accounted for in unreliability portion of PIs
- New PIs are developed for risk-significant support systems
  - Component cooling water and service water systems (or their equivalent cooling water systems)
- Pilot Program PIs address the recommendations made by ACRS on ROP PIs
  - PI thresholds should be plant-specific
  - Technical basis for choice of sampling intervals should be explained
  - Action levels should be related explicitly to risk metrics such as CDF and LERF, where possible

Insights from Phase 1 RBPI Study

- There were enough risk-significant differences among different plants that necessitated the development of plant-specific thresholds for unavailability and unreliability PIs
  - MSPI accounts for plant-specific differences
- Unavailability and unreliability indicators were found to provide objective and risk-informed indications of plant performance. They also provide broader risk coverage
  - They were tested by evaluating plant-specific data for 44 plants over three-year period (1997-1999)
- Performance indicators for CCW and SSW (or their equivalent support systems) were found to be difficult to develop due to the wide variation of plant-specific design features
  - Based on the technical analyses performed by NRC/industry, an approach has been developed for the Pilot
- Use of Bayesian update for estimating component unreliability was found to minimize the likelihood of false-positive/false-negative indications

Summary of Questions Raised by ACRS Subcommittee on PRA

- Summary of questions raised by ACRS subcommittee on PRA during the presentation on May 30, 2002 are:
  - What are the reasons/justifications for selection of baseline values used for calculating changes in unavailability and unreliability elements of the PIs?
  - Should the pilot program PIs be based on SECY-99-007 thresholds (i.e., 1.0×10^{-6}, 1.0×10^{-5}, 1.0×10^{-4})?
  - Background information on the technical approach/equations should be provided
- We will provide a white paper explaining the technical aspects of the pilot program PIs to ACRS members in time for the next ACRS subcommittee meeting
- We will hold at least two more ACRS subcommittee briefings during the course of the pilot program to discuss these and other technical issues

Conclusions

- The MSPI approach is based on risk insights. It accounts for plant-specific design/operating characteristics through the use of available risk models and data
  - Use of F-V importance measure to account for plant-specific features
  - Treatment of demand failures in unreliability indicators
  - Use of Bayesian update for unreliability indicators
  - Use of risk-significant functions rather that design-basis functions
  - Use of a new indicator for cooling water support systems
- The MSPI approach allows for balancing between component unreliability and unavailability consistent with the Maintenance Rule
- The MSPI provides more objective indication of plant performance and will provide broader risk coverage
- The limitations of the MSPI have been clearly identified and will be covered through inspection/SDP
- The MSPI provides appropriate risk-categorization of performance degradations that are covered by PIs

Implementation Schedule

July 23-25, 2002	Public workshop to prepare for start of the MSPI pilot
August 1, 2002	Start of MSPI pilot
November 2002	Briefing to ACRS subcommittee on pilot progress
February 2003	End of MSPI pilot data collection and start of the analysis period to analyze collected data
March 2003	Briefing to ACRS subcommittee on pilot progress
July 2003	End of pilot program

Plants Participating in the Pilot MSPI

Region I	Limerick 1/2 Millstone 2/3 Hope Creek Salem 1/2
Region II	Surry 1/2
Region III	Braidwood 1/2 Prairie Island 1/2
Region IV	Palo Verde 1/2/3 San Onofre 2/3 South Texas 1/2

Industry Trends Program

Thomas Boyce
Division of Inspection Program Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

June 6, 2002

Industry Trends

- Background
- Communications
- Process
- FY 2001 Results
- Future Development

Background

- NRC performance goal measure
- Purposes
- Relationship to NRC processes

Communications

- Industry indicators are published on the external NRC web site
- Annual report to Commission
- Annual report to Congress in NRC Performance and Accountability Report
- Conferences with Industry

Process

- Identify any statistically significant adverse industry trends
- Evaluate underlying issues and assess safety significance
- Agency response in accordance with existing NRC processes for generic communications
- Review at annual agency review meeting

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FY2001 Results

- No statistically significant adverse industry trends in safety performance
- Insufficient data on ROP indicators (<4 years)

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Future Development

- SRM of 8/2001 – Develop risk-informed thresholds "as soon as practicable"
- Enhanced performance goal measure
- Potential additional indicators

Industry Trend Thresholds

- RES is developing industry trend thresholds for use in a risk-informed regulatory framework.
- Industry thresholds differ from plant-specific thresholds.

Kinds of Thresholds

- Action Threshold
  - Used to measure industry performance, similar to thresholds used in ROP process
  - Report to Congress
- Early-Warning Threshold
  - Used to alert NRC to a change in industry trends that may indicate a change in industry safety performance

Threshold Characteristics

- Rational basis that is well documented
- Practical and simple
- Consistent with the existing regulatory framework
- Reflect risk (including associated uncertainties), safety, and regulatory perspectives

Protocol for Setting Thresholds

- Develop risk and statistical information related to trends for input to an expert panel
- Provide associated safety and regulatory information for expert panel
- Expert panel sets thresholds based upon input and expert judgment

Technical Approaches to Estimating Trend Thresholds

- Prediction Limits
- Bayesian Predictive Distribution
- Percentiles from Industry Distributions
- Insights from PRAs
- Rate-of-Change of Trend
- Expert Panel Input
- Modification of Current ROP PI Thresholds
- Combine Plant-specific Thresholds
- Integrated risk measure concept being developed for the enhanced PIs
- Combination of the above

Technical Questions

- How many years should be included in the estimation of a trend?
- What level is appropriate for reporting to Congress?
- What level is appropriate for agency action to an adverse trend?
- Should some of the PIs be grouped?
- How does the safety goal influence setting thresholds?
- Should concepts in Reg. Guide 1.174 be used in setting thresholds?

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Schedule

- Initial thresholds for initiating events and mitigating systems – July 02
- Thresholds for other cornerstones – Sep. 02
- Change to performance goal measure – Fall 02
- Annual Commission paper – Mar. 03
- Final thresholds – FY03

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United States Nuclear Regulatory Commission

Presentation to the ACRS Full Committee

Potential Policy Issues for Advanced Reactors

Ashok Thadani, Director

Office of Nuclear Regulatory Research

Farouk Eltawila, Director

Division of Systems Analysis and Regulatory Effectiveness

Office of Nuclear Regulatory Research

10

Outline

- Purpose of the Briefing
- Background
- Enhanced Margin of Safety
- Relationship to International Standards
- Five Potential Policy Issues
  - Event Selection
  - Fuel Performance
  - Source Term
  - Containment Performance
  - Emergency Evacuation
- Future Plan

Purpose of the Briefing

- Summarize Technical Issues With Generic Policy Implications For Advanced Reactors Resulting From Technical Portion of Preapplication Reviews
- Summarize Staff Previous Resolution of These Issues
  - SECY-93-092, July 30, 1993, SRM
- Solicit ACRS Feedback On Scope and Nature of Issues:
  - Is the List the of Key Issues Complete?
  - Are They Stated Clearly and Correctly?
  - What Are the Important Considerations in Their Resolution?
- Develop Risk-Informed Performance-Based Criteria for Advanced Reactors
  - Risk-Informed Regulation Implementation Plan Update, June 2002
- ACRS Letter

Background

- Preapplication Reviews For AP-1000 Complete
- PBMR Preapplication Activities in Closeout
- GT-MHR Preapplication Activities Planned
- Other Preapplication Activities Possible (E.G., GE-ESBWR, Framatome SWR1000, NG-CANDU)
- Many Issues Developed in the Course of the Review Have Generic Policy Implications:
  - Legal/Financial (SECY-01-207)
  - Technical (Memorandum to the Commission in Preparation)
- NEI White Paper On Risk-Informed Regulatory Framework For All Reactor Types
- Continued Work On Generic Policy Issues Can Facilitate Future Regulatory Work On Advanced Reactors

Enhanced Margins of Safety

- Advanced Reactor Policy Statement Expectations:
  - Enhanced Margins of Safety
  - Use of Simplified, Inherent, Passive or Other Innovative Means to Accomplish Safety Functions
- The Severe Accident Policy Statement Expectations:
  - Expectation for Advanced Reactors--Should Safer than Current reactors, as a minimum Provide the Same Degree of Protection As Current Plants
  - Completion of PRA and Consideration of Severe Accidents Vulnerabilities
  - Containment Equivalent to a Large, Dry Containment Capable of Mitigating a Core Melt
- Key Fundamental Policy Issues For Commission Consideration (RIRIP)
  - Current Metrics (CDF and LERF) Insufficient
  - Should Additional Cornerstones Be Included?
  - Should Environmental Risk Metrics Be Considered?
  - Should a Higher Level of Safety Be Required For New Plant Designs?
  - How Should Defense-In-Depth Be Applied to New Plant Designs?
  - Should Criteria Apply to Single Unit or Entire Site?

Relationship to International Safety Requirements

- Future Plant R&D Design and Marketing Are Becoming International Efforts
  - Designers
  - Suppliers
- Issue For Commission Consideration:
  - What Should Be the Relationship of NRC Safety Requirements to International Safety Requirements?
- Could Be Useful in Bringing in Expertise in Areas Where NRC Lacks Infrastructure

Event Selection and Safety Classification

- To What Extent Should a Probabilistic Approach Be Used to Establish the Licensing Basis For New Plant Designs, Considering:
  - Less Experience, Data, Methods
  - PRA Quality, Completeness and Documentation
- What Should Be the Criteria For Event Selection and Safety Classification, Considering
  - Role of Engineering Judgment
  - Risk Metrics, Criteria
  - Treatment of Uncertainties
  - Desired Confidence Level
- *In SECY-93-092, the Staff Recommended an Approach That Is Deterministically Based Supplemented With Probabilistic Information. In the July 30, 1993 SRM, the Commission Approved the Staff Recommendation.*

Fuel Performance and Qualification

- Fuel Performance and Qualification
  - Designed to Achieve Burnup of 80,000 MWd/t
  - Withstand High Temperature (1600^0 C) Without Release of Fission Products
- Should Fuel Qualification Testing Be Completed Prior to Granting A COL?
- Should Applicants Be Required to Test Fuel to Beyond Design Basis Event Conditions?
- What Independent Fuel Testing Should NRC Perform?
- Under What Conditions Can Fuel Manufactured Outside the U.S. Be Accepted For Use in a U.S. Plant?
- What Conditions Should NRC Employ to Ensure Fuel Quality Over the Life of the Plant?

Source Term

- Licensing Source Term –TID-14844, NUREG-1465 Are Used to Determine LWR Containment Effectiveness and Site Suitability
- Questions
  - Under What Conditions, If Any, Should the Commission Accept the Use of Scenario Specific Source Terms For Licensing Decisions Regarding Containment and Site Suitability?
  - What Codes and Model Validation Should Be Used?
  - How to Account For Uncertainties?
- *The Commission July 30, 1993, SRM Approved the Staff Recommendation in SECY-93-092, Which Recommended That Scenario Specific (Mechanistic) Source Terms Be Allowed Provided There Was a Sufficient Understanding of Fuel Performance, Fission Product Behavior and Accident Selection to Bound Uncertainties.*

Containment Performance

- Criteria For Containment/ Confinement Performance
  - Limit Fission Product Release
  - Barrier to Release
- Question
  - Under What Conditions, If Any, Should the Commission Accept the Use of Non-pressure Retaining Containment Building?
  - The Role of Containment Vs. Confinement In Protecting the Plant From External Threats, Sabotage, and Maintaining Public Confidence?
- *In SECY-93-092, the Staff Had Proposed an Approach for Containment That Focused on Functional Performance, Rather Than Prescriptive Design Criteria. The Commission July 30, 1993, SRM Approved the Staff Proposal, With the Addition of an Air Ingress Event to the MHTGR Proposed Accidents to Be Considered.*

Emergency Evacuation

- Under What Conditions, If Any, Would the Commission Approve Reducing the EPZ, Including a Reduction to the Site Exclusion Area Boundary?
  - Criteria For Evacuation Planning
    - Timing For Release
  - Source Term and Probability
  - Margin (Defense - In - Depth)
  - Evacuation Planning Zone
  - Infrastructure Requirements
  - Exercises
- *The Commission July 30, 1993 SRM Stated That It Was Premature to Reach a Conclusion on Emergency Planning for Advanced Reactors, but Requested the Staff Remain Open to Suggestions to Simplify EP Requirements for Reactors With Greater Safety Margins.*

Future Plan

- Develop Draft Staff Positions on Each Issue
- Solicit Stakeholder Feedback:
  - Public Workshop
  - ACRS Meetings
- Final Paper Requesting Commission Guidance - Fall 2002

(11)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
493rd ACRS Meeting
PROPOSED RULEMAKING TO ENDORSE NFPA 805
June 7, 2002
ROCKVILLE, MARYLAND

- PROPOSED AGENDA -

	TOPIC	PRESENTER	TIME
I.	Subcommittee Chairman Opening Remarks	S. Rosen, ACRS	8:35-8:40 a.m.
II.	Proposed Revision to 10 CFR 50.48 to permit voluntary adoption of NFPA 805 licensing basis for light water reactor fire protection requirements	E. Weiss, NRR	8:40-9:20 a.m.
III.	Industry Perspective on Proposed Revision to 10CFR50.48	F. Emerson, NEI	9:20-9:50 a.m.
IV.	Committee Comments/Discussion	S. Rosen, ACRS	9:50-10:00 a.m.

Cognizant ACRS Member: S. Rosen
Cognizant ACRS Staff Engineer: S. Duraiswamy/R. Elliott

NOTE:

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

Number of copies of the presentation materials to be provided to the ACRS - 50.

NFPA 805 RULEMAKING

Briefing for the ACRS

Eric Weiss, Chief
Fire Protection Section, NRR
415-3264

June 7, 2002

NFPA 805 - Performance-Based Standard for Fire Protection for LWRs

- Background
- Advantages of Endorsing NFPA 805
- NFPA 805 Structure
- Rule Structure
- Major Issues
- Status of Rulemaking
- What this means

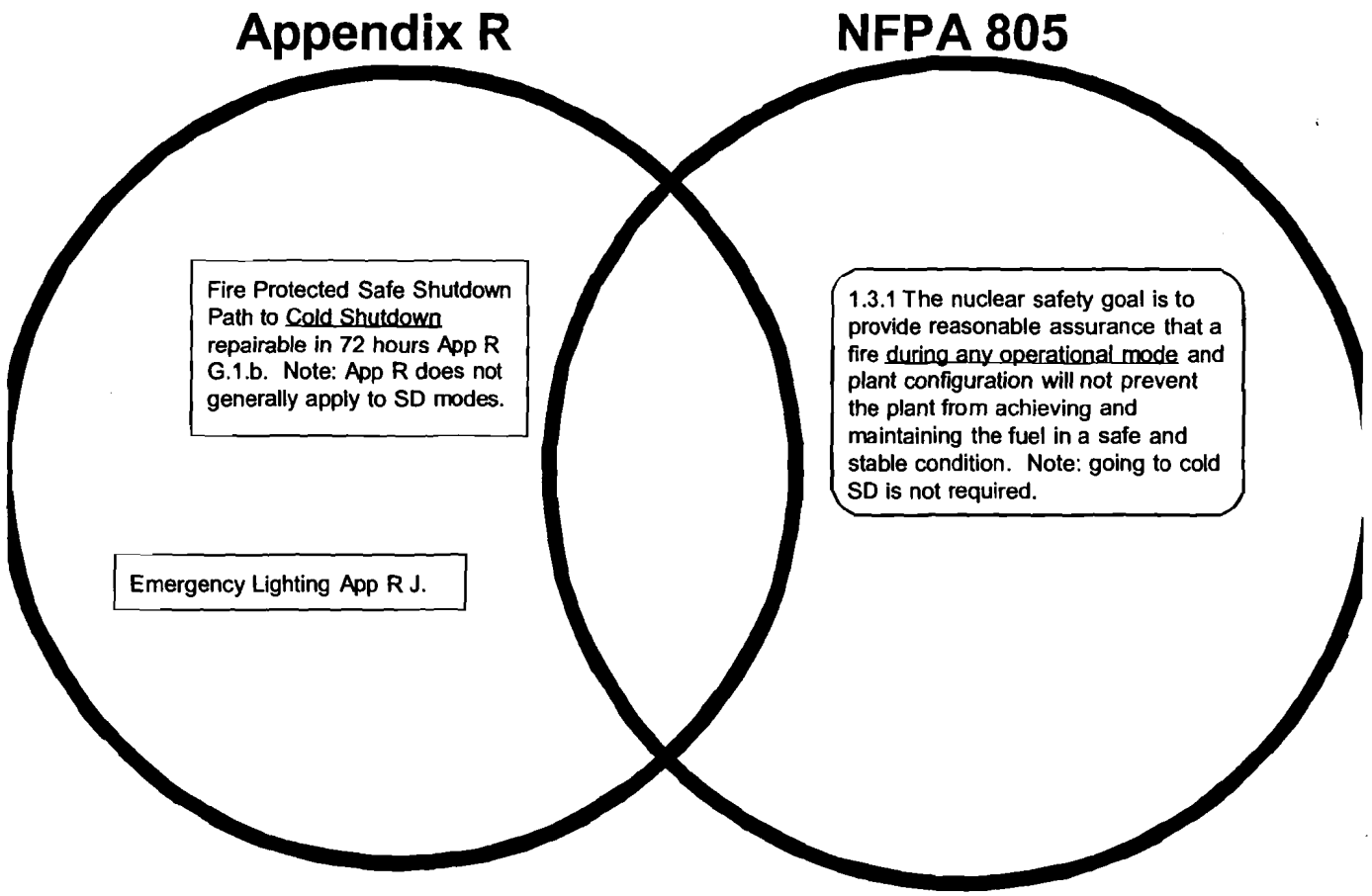
NFPA 805 - Background

- 10CFR 50 Appendix R – deterministic regulation
- ACRS Letter Feb 18, 1999
- Comprehensive Reg Guide 1.189 recently issued
- RI/PB National Consensus Standard Published Feb 2001
- Approved by the ANSI

NFPA 805 – Background (Continued)

- Rule to endorse consistent with National Technology Transfer and Advancement Act (PL 104-113) and OMB Circular A119
- Takes advantages of advances in PRA and Fire Science since Appendix R was issued over 20 years ago
- Some Methods of Achieving Fire Safety are different than Appendix R

NFPA 805 – Background (Continued)



Advantages

- Allows licensees to maintain safety through more flexible, efficient, and rational processes
- Reduces exemptions, submittals, and reviews
- Allows use of risk insights and fire modeling, science, and engineering
- Consistent with NRC's Outcome Goals
- Allows Licensees to focus FP program on most safety significant issues

NFPA 805 Structure

- Allows transition of existing Appendix R licensing basis including existing exemptions and GL86-10 equivalencies
- Allows future changes to plant and licensing basis to be either Deterministic or RI/PB
- Incorporates a change control process to monitor risk
- New RI/PB methods to be approved by the NRR

NFPA 805 Structure (Continued)

- NFPA 805 allows either a deterministic or a RI/PB approach (see NFPA 805 figure 2.2 "Methodology")
- Deterministic Requirements similar to Appendix R (see NFPA 805 figure 4.2.2)
  - 3-hour encapsulation of one success path
  - 1-hour encapsulation of one success path with suppression and detection
  - 20ft of separation without intervening combustibles and suppression and detection throughout the area

NFPA 805 Structure (Continued)

- NFPA 805 Chapter 3 "Fundamental Fire Protection Elements"
  - Fire Protection Plan
  - Prevention (e.g. control of combustibles)
  - Fire Brigade
  - Water Supply
  - Standpipes and Hose Stations
  - Fire Extinguishers
  - Fire Alarm and Detection Systems
  - Automatic and Manual Water-based Fire Suppression Systems
  - Gaseous Fire Suppression Systems
  - Passive Fire Suppression (e.g. Building separation, Fire Barriers, Penetrations)

Rule Structure

- Amendment of 50.48 to allow use of NFPA 805
- Adoption is voluntary
- Existing Licensing Basis, Configuration and Procedures convey to new RI/PB environment
- Licensees document and retain records on site
- ROP monitors future changes
- Allows NRC to approve new RI/PB methods in the future
- Licensees may use NFPA 805 appendices

Major Points

- One of the NRC's first RI/PB rules
- NEI endorsed rulemaking in September 2001
- Key to successful implementation is appropriate regulatory guidance
- NEI agreed to develop guidance document by 2003 (first draft June 2002)
- NFPA 805 addresses existing LWRs
- NFPA 804 addresses Advanced LWRs
- Future NFPA std to address Advanced LWR and Gas Reactors in RI/PB manner

Schedule

- Proposed Rule to ACRS and CRGR June 2002
- Proposed Rule to Commission July 2002
- Proposed Rule Published in FR for Comment one month after SRM
- Final Rule to Commission 15 months after close of public comments on Proposed Rule
- Final Rule Published in FR one month after SRM

What This Means

- This moves Fire Protection into Risk-Informed Performance Based Arena
- This represents an opportunity to improve the efficiency and effectiveness of NRC's regulatory environment

The following is the current draft proposed rule language which updates the draft proposed rule wording published in the *Federal Register* on April 2, 2002. This language is preliminary and may be incomplete in one or more aspects. NRC may post additional updates periodically on the rulemaking website.

Questions regarding this language can be directed to Leon Whitney, NFPA 805 Task Manager, (301) 415-3081, lew1@nrc.gov.

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§ 50.48. Fire protection.

* * * * *

(c) *National Fire Protection Standard NFPA 805.*

(1) *Approval of incorporation by reference.* National Fire Protection Association (NFPA) Standard 805, "Performance-Based for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (NFPA 805), which is referenced in this section, was approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of NFPA 805 may be purchased from the NFPA Customer Service Department, 1 Batterymarch Park, P.O. Box 9101, Quincy, MA 02269-9101 and in PDF format through the NFPA Online Catalog (www.nfpa.org) or by calling 1-800-344-3555 or 617-770-3000. Copies are also available for inspection at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852-2738, and at the NRC Public Document Room, Building One White Flint North, Room O1-F15, 11555 Rockville Pike, Rockville, Maryland 20852-2738. Copies are also available at the Office of the Federal Register, 800 N. Capitol Street, Suite 700, Washington, DC.

(2) *Exceptions, modifications, and supplementation of NFPA 805.* As used in this section, references to NFPA 805 are to the 2001 Edition, with the following exceptions, modifications, and supplementations:

(i) Life Safety Goal. The Life Safety Goal of Section 1.3.3 is not endorsed.

(ii) Plant Damage/Business Interruption Objectives. The Plant Damage/Business Interruption Objectives of Section 1.3.4 of NFPA 805 are not endorsed.

(iii) Use of Feed-and-Bleed. In demonstrating compliance with the performance criteria of Sections 1.5.1(b) and (c) of NFPA 805, a high pressure charging/injection pump coupled with the pressurizer power-operated relief valves (PORVs) as the sole fire-protected safe shutdown path for maintaining reactor coolant inventory, pressure control, and decay heat removal capability (i.e., feed-and-bleed) for pressurized-water reactors (PWRs) is not permitted.

(iv) Uncertainty Analysis. An uncertainty analysis performed in accordance with Section 2.7.3.5 is not required to support deterministic approach calculations.

(v) Existing Cables. In lieu of installing cables meeting flame propagation tests as required by Section 3.3.5.3 of the standard, a flame retardant coating may be applied to the electric cables, or an automatic fixed fire suppression system may be installed to provide an equivalent level of protection." In addition, the italicized exception to Section 3.3.5.3 is not endorsed.

(vi) Water Supply and Distribution. The italicized exception to Section 3.6.4 is not endorsed.

(3) *Compliance with NFPA 805.*

(i) A licensee may maintain a fire protection program that complies with NFPA 805 as an alternative to complying with paragraph (b) of this section for plants licensed to operate

before January 1, 1979; the fire protection license conditions for plants licensed to operate after January 1, 1979; or paragraph (f) of this section for plants for which licensees have submitted the certifications required under 10 CFR 50.82(a)(1). The licensee shall submit a request to comply with NFPA 805 in the form of an application for license amendment under § 50.90. The application must identify any orders and license conditions that must be revised or superseded, and contain any necessary revisions to the plant's technical specifications and the bases therefore. The Director of the NRC's Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the licensee has identified orders, license conditions, and the technical specifications that must be revised or superseded, and that any necessary revisions are adequate. The Director or designee of the Director shall issue a license amendment approving the use of NFPA 805 and any necessary revisions to the technical specifications.

(ii) The licensee shall complete its implementation of the methodology in Chapter 2 of NFPA 805 (including all required evaluations and analyses) and, upon completion, modify the fire protection plan required by paragraph (a) of this section to reflect the licensee's decision to comply with NFPA 805, before changing its fire protection program or nuclear power plant as permitted by NFPA 805.

(4) *Alternative Methods and Analytical Approaches.* A licensee may submit a request to use alternative methods and analytical approaches in lieu of those specified in NFPA 805. The request must be in the form of an application for license amendment under § 50.90. The Director of the NRC's Office of Nuclear Reactor Regulation, or designee of the Director, may approve the application if the Director or designee determines that the alternative methods and analytical approaches meet the following criteria:

(i) Satisfy the goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release.

(ii) Maintain safety margins.

(iii) Maintain fire protection defense-in-depth (fire prevention, fire suppression, and post-fire safe shutdown capability).

(14)

Risk-Informed, Performance-Based Fire Protection Implementing Guidance

Fred Emerson, NEI

Advisory Committee on Reactor Safeguards
June 7, 2002

Topics

- Background
- Industry positions
- Current rule language
- Implementing guidance
- Moving forward

Background

- Industry agreed to develop implementing guidance for rulemaking
  - Vehicle for resolving some of the open issues
  - Others to be resolved in the rule language (exceptions to the standard)
- NRC will utilize in Regulatory Guide
- Multi-discipline contractor team developing
  - NEI Rulemaking ITF oversight
  - Frequent interaction with NRC
- Rule language an issue
  - Developing guidance in parallel with rule language

Industry Positions

- Increased use of risk information in fire protection regulation
  - Licensees should be able to use tools whether or not they transition to NFPA 805
- Rulemaking optional
- Uncomplicated transition
- Thorough understanding of licensing basis during transition

Uncomplicated Transition

- "Safe today = safe tomorrow"
- Clear guidance and well-understood process necessary
- Areas to be addressed
  - Document submittal requirements vs. retained documentation
  - License amendment submittal vs "50.59" supporting evaluation
  - Use of docketed licensing basis material to address Chapter 3 elements

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Licensing Basis

- Thorough understanding of licensing basis throughout transition is essential
- Areas to be addressed in rule or guidance
  - Analysis/definition of licensing basis
  - Use of currently docketed licensing basis
  - Approval of RI/PB methods
  - Inspection and enforcement during transition

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Implementing Guidance

- Appendices provide one or more methods for interpreting/using information in NFPA 805, such as
  - Establishing fire protection program fundamentals
  - Identification of performance criteria, fire hazards, and applicable SSCs
  - Evaluation against performance criteria
- Completion in parallel with issuing rule in late 2003
  - Partial first draft to be provided to NRC later this month

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Potential Barriers

- Convergence of rule language, implementing guidance, and inspection guidance and training
- License amendment for use of analytical methods (SER preferred)
- Definition/use of current licensing basis
- Technical issues
  - Seismic 1E fire pumps
  - Seismic standpipes

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Moving Forward

- One scenario for utility use
  - First, better methods for fire PSAs
    - ANS fire PSA standard development
    - EPRI/RES fire PSA requantification project completion
  - Second, a few plants willing to utilize the NFPA 805 rulemaking
    - Plants with established PRAs and successful use of risk information in the past
    - Successful use of 805 tools in a few evaluations, then more widespread

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Moving Forward

- One scenario
  - Third, with successful plant experience, more plants willing to use it
    - Improve plant fire PSA tools
    - Integrate fire risk with overall plant risk metric
    - Willing to apply risk information generally and for fire protection
- Eventually expect to see most plants using this to some degree, **IF....**

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The following is the current draft proposed rule language which updates the draft proposed rule wording published in the *Federal Register* on April 2, 2002. This language is preliminary and may be incomplete in one or more aspects. NRC may post additional updates periodically on the rulemaking website.

Questions regarding this language can be directed to Leon Whitney, NFPA 805 Task Manager, (301) 415-3081, lew1@nrc.gov.

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§ 50.48. Fire protection.

* * * * *

(c) *National Fire Protection Standard NFPA 805.*

(1) *Approval of incorporation by reference.* National Fire Protection Association (NFPA) Standard 805, "Performance-Based for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (NFPA 805), which is referenced in this section, was approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of NFPA 805 may be purchased from the NFPA Customer Service Department, 1 Batterymarch Park, P.O. Box 9101, Quincy, MA 02269-9101 and in PDF format through the NFPA Online Catalog (www.nfpa.org) or by calling 1-800-344-3555 or 617-770-3000. Copies are also available for inspection at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852-2738, and at the NRC Public Document Room, Building One White Flint North, Room O1-F15, 11555 Rockville Pike, Rockville, Maryland 20852-2738. Copies are also available at the Office of the Federal Register, 800 N. Capitol Street, Suite 700, Washington, DC.

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(ii) Maintain safety margins.

(iii) Maintain fire protection defense-in-depth (fire prevention, fire suppression, and post-fire safe shutdown capability).

PRESENTATION TO ACRS

ON

POTENTIAL UNDETECTED VOIDS IN CONCRETE STRUCTURES

OF NUCLEAR POWER PLANTS

NRR/DE/EMEB

Gene Imbro
Kamal Manoly

I REGULATORY MEASURES TO MINIMIZE CONSTRUCTION DEFECTS

- Licensee Implementation of NRC Approved 10 CFR 50, Appendix B, "Quality Assurance Criteria For Nuclear Power Plants"

- Compliance with 10 CFR 50.55(e)

- NRC Inspection Manual Chapter 2512, "LIGHT WATER REACTOR INSPECTION PROGRAM"

II DETECTION OF SIGNIFICANT VOIDS IN CONCRETE STRUCTURES

- Search of Available NUDOCS Records on Concrete Voids Dating Back to 1968
- Identified Nine 10CFR 50. 55(e) Correspondence From Licensees on Cases Involving Containment and Other Concrete Structures
- Records Show That QA & QC Typically Identifies Significant Voids in Concrete Structures
- Most Voids Are Located in Congested Reinforcing Steel Areas (i.e., around penetrations)
- Most Voids Are Created Due to Improper Concrete Placement and Inadequate Concrete Vibration

III STRUCTURAL INTEGRITY TEST (SIT)

- After Completion of Construction and Prior to Operation, Every Containment Structure is Subjected to an SIT, at a Test Pressure at Least 1.15 Times Containment Design Pressure

- Instrumentation of Containment Rebar at Several Critical Joints (Near Mat and Around Largest Penetration), Provides Assurance That Containment Behavior During SIT in Close Agreement With Predicted Results From Analysis

- Performance of SIT Provides Assurance That Potential Undetected Voids in Concrete Would Not Significantly Affect The Load Carrying Capacity of Containment

VI VOIDS IN TURKEY POINT UNITS 3 & 4 CONCRETE CONTAINMENTS

- Voids Identified in Licensee's LERs for both Units During Steam Generator Replacement

- Size of Voids Found on Turkey Point Units 3 & 4 Containments Were Generally Small and Did Not significantly Affect Containment Load Carrying Capacity

- NRC Reviewed Licensee's LERs as Part of The Normal LER Follow-up Process, Verified The Adequacy of Licensee's Corrective Action and Determined That The Finding Had No Generic Applicability

V CONCLUSION

- Staff Revisited The Issue of Whether The Voids Found in The Concrete Containment at Turkey Point Units 3&4 Had Generic Applicability.

- NRC Inspection Manual Chapter 2512 Program Verifies That Licensee's QA & QC Program Adequately Implemented

- Based on NRC and Licensees Programmatic Activities, There is Reasonable Assurance That Significant Voids in Concrete Structures Are Detected and Corrective Actions Are Implemented

CONCLUSION (Cont.)

- While Minor Construction Deficiencies, Including Voids in Concrete Structures, May Be Present, Their Presence is Not Expected to Affect The Load Carrying Capacity of These Structures

- Structural Integrity Test Provides Additional Assurance That Voids That Could Affect The Load Carrying Capacity of Containment Are Detected

- Treatment of This Concern as a Generic Issue is Not Warranted

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 6, 2002

MEMORANDUM TO: Pao-Tsin Kuo, Program Director
License Renewal and Environmental Impacts
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

FROM: Eugene V. Imbro, Chief ***/RA/***
Mechanical & Civil Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

SUBJECT: ADVISORY COMMITTEE FOR REACTOR SAFETY (ACRS) INQUIRY REGARDING VOIDS IN CONCRETE STRUCTURES IN NUCLEAR POWER PLANTS

During the staff's presentation to the ACRS regarding the license renewal application for Turkey Point, an issue was raised by the ACRS that related to the potential for the existence of undetected voids in concrete structures at nuclear power plants and their safety significance. This memorandum responds to the ACRS's concern regarding the staff's treatment of any potential generic implications of the voids in the concrete containment structures found at Turkey Point Units 3 and 4. This memorandum provides information in support of the staff's conclusion that potential undetected voids in concrete structures do not have any generic safety implications.

The NRC was informed of the concrete voids found in the Turkey Point Units 3 and 4 containment penetrations as a result of an LER (No. 251-81-14) submitted by the licensee upon identification of this construction deficiency during the steam generator replacement activity. The issuance of an LER as required by 10 CFR 50.72, 50.73, or 10 CFR 73.71, satisfies the reporting requirements of 10 CFR 21. The NRC performed an inspection and assessment of the LER findings, licensee's corrective action plan and physical repairs as part of the routine follow-up and closure of LERs. The closure of these LERs is documented in NRC Inspection Report 50-250/81-33 & 50-251/81-33. The NRC inspection report indicated that NRC's assessment of the concrete voids issue reviewed the issue for generic applicability, however, no specifics were included in the report. The fact that the identified voids in Turkey Point Units 3 and 4 concrete containments were relatively small, and an analysis by Bechtel indicated that all stresses remained within the allowable stress range, may have been the reason for not treating the finding as a generic safety issue requiring resolution.

The staff has revisited the issue of whether the existence of voids in the concrete containment has generic safety implications. As discussed below, our evaluation of the concern involving potential existence of voids in concrete containments that could degrade containment performance and its load carrying capacity, supports the original determination, made by the NRC upon identification of the voids in the Turkey Point containments, that the treatment of such findings as a generic issue was not warranted.

The staff relies on several layers of protection to prevent, detect and repair defects discovered during construction, including voids in concrete structures during the construction of the facility. These are: 1) the implementation by the licensee of their NRC approved 10 CFR 50, Appendix B, QA program and the licensee's QC program; 2) the requirements of 10 CFR 50.55(e) that

holders of Construction Permits identify, evaluate and report defects and failures to comply with NRC requirements associated with substantial safety hazards to the NRC in a timely manner, generally within 60 days; and 3) the verification by NRC inspectors as defined by the NRC's construction inspection program contained in NRC Inspection Manual Chapter 2512, "LIGHT WATER REACTOR INSPECTION PROGRAM - CONSTRUCTION PHASE" that the construction is in accordance with approved design documents, the licensee is properly and effectively implementing their QA/QC program, construction defects are reported to NRC as required by 10 CFR 50.55(e), and that appropriate corrective actions are taken by the licensee. These layers are described below.

1) Appendix B to 10 CFR 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," requires an applicant to establish a quality assurance program which complies with the requirements of this appendix. This program is documented by written policies, procedures, or instructions and is carried out throughout plant life, including construction, in accordance with those policies, procedures, or instructions. Quality assurance comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. An important part of any Q/A program is the identification of conditions adverse to quality and corrective actions.

2) 10 CFR 50.55(e) requires an applicant holding a facility construction permit to adopt procedures to identify and correct and report any deviations or failures to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, regulation, order, or license of the Commission relating to a substantial safety hazard found during the construction of nuclear power plants. A substantial safety hazard is defined in 10 CFR 50.2 as a loss of safety function to the extent that there is a major reduction in the degree of protection provided to the public health and safety for any facility or activity authorized by the construction permit. A construction permit holder must also report any significant breakdowns in any portion of their Appendix B, Quality Assurance program which could have produced a defect in a basic component. A concrete containment or other concrete structures housing safe shutdown or accident mitigation equipment would meet the 10 CFR 50.2 definition of a basic component. Reports made under 50.55(e) and corrective actions were reviewed on a sample basis by NRC inspectors during the construction inspections performed as a part of Inspection Manual Chapter 2512.

3) The primary objective of the construction inspection program contained in NRC Inspection Manual Chapter 2512, "LIGHT WATER REACTOR INSPECTION PROGRAM - CONSTRUCTION PHASE" is to ensure public health and safety through the evaluation of the adequacy of licensee performance during construction and major plant modifications. This is accomplished by determining licensee effectiveness in identifying conditions that may adversely affect operational safety and in achieving compliance with NRC requirements and licensee commitments. Information for the above is typically obtained by direct observation of activities, including concrete placement, personnel interviews, review of procedures and records, and by evaluation of licensee and contractor performance, including licensee involvement and control over licensed activities. The NRC's Manual Chanter 2512 program requires the inspection of a licensee's construction activities to verify the implementation of their QA/QC program throughout the construction of nuclear structures. Therefore, the staff's determination

that a facility is constructed in substantial conformance with its design requirements relies in large part on the assurance provide by the Inspection Manual Chapter 2512 program that the licensee's QA/QC program has been adequately implemented throughout the construction phase. Minor construction deficiencies could be expected in any type of construction as massive as that encountered in nuclear power plants. However, it is unlikely that any deficiency that would significantly affect the structural adequacy of major concrete structures and the performance of their safety function would not be identified and corrected.

Documentation demonstrating that these controls have been effective in detecting and repairing voids in concrete structures is provided in Attachment 1. The staff searched the records in NUDOCS for examples of identified concrete voids in nuclear concrete structures. Attachment 1 provides a summary of the identified instances of concrete voids and how each finding was resolved. The samples show that significant voids in concrete structures that are detected during construction were typically identified by QC, and the defects were corrected. The samples also show that voids in concrete were generated mainly due to improper concrete placement, and/or inadequate and insufficient vibration to consolidate the wet concrete. The samples identified areas of the structures where voids were likely to occur, such as those detected in concrete containment buildings near the areas where the rebar is most congested, e.g., the equipment hatch, and penetrations, and placement and vibration of concrete become difficult.

Despite the layered approach described above to detect and repair construction defects there is a small likelihood that undetected voids exist in concrete nuclear power plant structures. In general, the licensee's Q/A program pursuant to 10 CFR 50, Appendix B, licensee's Q/C program supplemented by the reporting requirements of 10 CFR 50.55(e), and NRC's inspection oversight during construction that verified effective implementation of these programs as well as direct observation of construction activities, provide reasonable assurance that concrete structures will perform their intended design function. In addition to the above, for concrete containments, an additional level of assurance that undetected voids in concrete do not represent a safety concern is provided by the conduct of the Structural Integrity Test (SIT) which is performed prior to the issuance of an operating license.

The ASME *Boiler and Pressure Vessel Code*, Section III, Division 2, Article CC-6000, "Structural Integrity Test of Concrete containments," contains the requirements for structural integrity testing of concrete containments (CC). Prior to the incorporation of SIT requirements in ASME Section III Code, concrete containments in nuclear power plants were tested in accordance with the provision in Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments." The SIT is performed at a test pressure of at least 1.15 times the containment design pressure to demonstrate the quality of construction and to verify the acceptable performance of new design features

During the design phase of concrete containments, and prior to the performance of an SIT, the design organization performs an analysis of the concrete containment when subjected to the SIT test using the same structural model utilized in the evaluation of other design loading conditions. Radial deflections are tabulated at several points along six meridians spaced around the containment, including locations with varying stiffness characteristics such as buttresses, walls, large openings and connections with other structures, i.e., areas of structural discontinuity. Vertical deflections of the containment are obtained at the apex, and the springline of the dome. Radial and tangential deflections of the containment wall adjacent to

the largest opening are determined from the analysis at twelve points along the 3 o'clock, 6 o'clock, 9 o'clock and the 12 o'clock orientations from the opening. If the second largest opening is loaded in a different manner from the largest opening, similar deflection values are determinated. Results from prototype and non-prototype containments include determination of strain values near the inside face, at the mid-point and near the outside face at the designated locations. When the SIT is performed, strain and displacement measurements are obtained at the same locations identified above and compared against the analysis results. The acceptability of the SIT is based on close agreement between calculated and tested values of strains and displacements.

The staff concludes that, in addition to the programs described above, the SIT provides an additional level of assurance, that the containment structure is constructed in substantial conformance with the design requirements and that the containment will perform its specified function. The existence of undetected voids in the concrete that would significantly affect the load carrying capacity of a containment, will likely affect the measured structural displacements and would result in a significant deviation from calculated displacements, indicative of a problem. Furthermore, voids in concrete containments that could significantly affect the load carrying capacity of the containment will be self-revealing by the development of cracking on the external containment surface that would be substantially different from expected concrete cracking that normally results from the SIT. Following the SIT, detailed walkdowns are performed of the concrete containment structures. Cracking resulting from voids in the concrete that could affect the load carrying capacity of the containment would be easily detectable on external concrete surfaces.

Quality assurance programs established by licenses to satisfy the requirements of 10 CFR 50, Appendix B, licensee programs required by 10 CFR 50.55(e) and the NRC's Construction Inspection Program defined in Inspection Manual Chapter 2512 provide the staff reasonable assurance that major construction deficiencies, including voids in concrete, are detected and corrected. In addition, the performance of the SIT provides an added assurance regarding the overall structural capability of a concrete containment. The staff concludes that adequate measures were in place during construction and prior to operation that would have identified and corrected any voids in concrete structures, including concrete containments, that would have significantly affected their structural integrity. Therefore, no generic action was necessary as a result of the voids found in the concrete containment at Turkey Point Units 3 and 4 during the replacement of their steam generators.

If you need further clarification regarding the information we obtained, please do not hesitate to contact me or my staff.

Attachment: As stated

CONTACT: Kamal Manoly, NRR/EMEB
415-2765

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Quality assurance programs established by licenses to satisfy the requirements of 10 CFR 50, Appendix B, licensee programs required by 10 CFR 50.55(e), and the NRC's Construction Inspection Program defined in Inspection Manual Chapter 2512, provide the staff reasonable assurance that major construction deficiencies, including voids in concrete, are detected and corrected. In addition, the performance of the SIT provides an added assurance regarding the overall structural capability of a concrete containment. The staff concludes that adequate measures were in place during construction and prior to operation that would have identified and corrected any voids in concrete structures, including concrete containments, that would have significantly affected their structural integrity. Therefore, no generic action was necessary as a result of the voids found in the concrete containment at Turkey Point Units 3 and 4 during the replacement of their steam generators.

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NAME	J.S. Ma*	K.A. Manoly*	E.V. Imbro
DATE	06/04/02	06/04/02	06/06/02

*see previous concurrence **OFFICIAL RECORD COPY**

Attachment

SUMMARY OF FINDINGS RELATING TO VOIDS IN CONCRETE STRUCTURES IN NUCLEAR POWER PLANTS

The following findings were identified during the records search on the subject of voids in concrete structures in nuclear power plants. The examples confirm that, for the most part, when deficiencies involving concrete voids took place during the construction phase, they were typically identified as a result of QA/QC program implementation, and were resolved in accordance with established programs for resolution of non-conformances.

(1) Voids at the Concrete/Buttress Plate Joint of Zion Containment Buildings

On December 31, 1970, Commonwealth Edison company reported to the Atomic Energy commission (AEC) information relating to the discovery, evaluation, corrective actions, and resolution of a void at the concrete/buttress plate joint of the Zion Containment Building.

It was concluded the observed void had been generated between poured concrete and a vertical steel buttress plate after concrete form ties were released. Corrective action was instituted to require use of concrete anchors on the balance of concrete pours and leaving form ties in place. All buttress joints involved in concrete pours previous to the discovery of the void were to be grouted.

(2) Voids at the Ring Girder of the Three Mile Island Unit 1 Containment Building

In a correspondence dated January 5, 1972, from AEC Region I to Metropolitan Edison company, AEC discussed its inspection of construction records and repair in the North 180 degree portion of the Unit No. 1 ring girder, which had identified voids that were discussed in a summary report, dated December 3, 1971, and titled "Report on Containment Building Ring Girder Construction and Repair." In a follow-up inspection, on January 27, 1972, AEC discussed its finding that the repair of the ring girder concrete voids was continuing in accordance with the proposal submitted by the applicant. The inspection report indicated that the ring girder appeared to have been excavated to sound concrete before its repair.

(3) Voids at the Reactor Building Exterior Wall of V.C. Summer

On January 16, 1976, South Carolina Electric and Gas company notified NRC Region II that a situation at V.C. Summer Nuclear Station constituted a probable significant deficiency under the provisions of 10 CFR 50.55(e). The deficiency report indicated that voids in concrete had been detected behind the liner plate of the Reactor Building exterior wall. In its report on concrete placement, the licensee's expressed dissatisfaction with the method of placing and consolidating the concrete. An investigation was made by striking on the liner with a hammer which yielded approximately fifty places with "hollow" sound indicating voids behind the liner plate. Honeycombed and unconsolidated concrete was removed by chipping. The void areas were grouted with new concrete by pumps. South Carolina Electric and Gas company interim and final reports related to the construction deficiency provided a description of

the deficiency, analysis of safety implication, repair, and corrective action.

(4) Delamination at Crystal River Containment Dome

On April 21, 1976, Florida Power Corporation (FPC) sent a letter to the NRC stating that a separation in concrete of the Crystal River Containment Building dome had occurred. FPC submitted a report to the NRC that described, in detail, the structure of the Containment Building, the delamination problem, and the proposed corrective actions. The report stated that, on April 14, 1976, electricians were attempting to secure drilled-in anchors to the top surface of the dome and found that certain anchors would not hold. Further investigation revealed an area of the dome surface which sounded hollow when hit with a hammer. The size of the area was approximately 30 feet long by 3 feet wide. The area was curved slightly as it appeared to follow the circumferential construction joint between pours G and H. The delaminated cap was removed, starting at the apex and continued toward the ring girder until the concrete above the main delamination was totally removed and lower sound concrete was exposed. A new concrete cap was then poured and was wet cured for fourteen days.

(5) Voids at the Unit 1 Reactor Containment Building Wall of South Texas

On June 18, 1979, pursuant to 10 CFR 50.55(e), Houston Lighting & Power Company notified the NRC by telephone of voids in concrete behind the liner plate in the eighth lift of the Reactor Building exterior wall, and submitted an interim report on the deficiency on July 16, 1979. On August 15, 1979, Houston Lighting & Power Company submitted a second interim report on the deficiency. At that time, a significant amount of investigation had been completed and recommendations for corrective actions were proposed. The voids were found in areas beneath the shell penetrations and/or beneath the 8 inch channel and plate stiffeners where additional rebars were located. A program was established to identify all significant voids in the Reactor Containment Building for both Units 1 and 2. Larger voids were found in areas of large penetrations where concrete placement was more difficult due to the added congestion of rebars. Voids were grouted with concrete by pumps and the repair was witnessed by a Brown & Root QA/QC inspector. NRC Region IV staff inspected the void areas and indicated that 156 exploratory holes had been drilled in the liner plate. Twenty-six (26) holes were associated with voids areas and the rest of the holes had shown solid concrete. On February 18, 1980, Houston Lighting & Power Company submitted a final report to the NRC that described the deficiency, the cause, the extent of the problem, and the corrective action that had been taken.

(6) Void Beneath the Equipment Hatch of Wolf Creek

On December 14, 1978, Kansas Gas and Electric company reported to NRC that a void in concrete was revealed when the forms for the 6th lift of the Wolf creek Reactor Containment Building were lifted the day before. The void extended about 2 feet downward from the bottom of the equipment hatch centerline. The width of the void was approximately six feet and the depth extended through the thickness of the wall, which is approximately eight feet. An interim report on the void was submitted to the NRC on January 12, 1979. A 5-1/2' by 8' section of the equipment hatch liner was removed to facilitate the removal of unsound concrete and the placement of the new concrete. New concrete was placed into the void area and later the cut-out section of the hatch liner

was welded back into position. A final report, which presented details of the repairs, was sent to the NRC on September 25, 1979.

(7) Void in the Auxiliary Building Base Mat of Marble Hill

On June 4, 1979, Public Service Company of Indiana, Inc. (PSI) notified the NRC by telephone of a potentially reportable item as required by 10 CFR 50.55(e). PSI discovered a void in the Auxiliary Building base mat on March 29, 1979. The void was visually inspected on April 6, 1979, and ultrasonically tested on May 7 and 8, 1979. The results of these inspections revealed that the void extended 25' in the north-south direction, up to 9' in the east-west direction and vertically from the bottom layer of rebars to the mud mat 12" below. PSI stated that a void of this size could degrade the structural integrity of the Auxiliary Building and, therefore, make it unable to meet its seismic design requirements. All unsound concrete was removed using chipping hammers. The intended configuration of the bottom of the slab was restored with shotcrete. The shotcrete repair was tested ultrasonically and found to be acceptable. Voids were filled with concrete of the same compressive strength as the original base mat. PSI submitted its final report on the void and its repairs on April 28, 1980.

(8) Honeycombs at the Callaway Containment Building Base Mat

On June 17, 1977, Union Electric (UE) notified the NRC by telephone that voids (honeycombs) were found in the base mat concrete at the trumplates. On July 15, 1977, UE submitted a 30-day interim report stating that 33 of the 172 trumplates required removal of honeycomb concrete to evaluate the significance of the deficiency. Voids (honeycombs) were grouted with high strength concrete. An NRC staff inspected the concrete deficiency in the base mat. The NRC staff observed the locations where removal of unsound concrete extended behind nineteen bearing plates and exposed bottom layers of the base mat rebars. The detailed inspection results were documented in inspection reports.

(9) Voids in Unit 2 Containment Building Exterior Walls of Palo Verde

In prestressed concrete containments, after each tendon in the containment building wall is installed and tensioned, the specification requires the sheath containing the tendon to be pumped full of grease to protect the tendon from corrosion. As part of the greasing procedure, a preliminary air test of each sheath is routinely performed prior to greasing to assure that the sheath will hold the grease and will not permit uncontrolled escape of grease. At Palo Verde Unit 2, upon conducting the preliminary air tests on tendons H32-009 and H32-001 prior to greasing, the licensee noted that air was escaping to the outside through a crack in the exterior surface of the concrete wall. After an investigation, the licensee determined that the total volume of the leak was conservatively estimated to be about 5-3/4 cubic feet. Approximately five cubic feet of grout was injected into the wall and effectively sealed the leak pathway. Arizona Public Service company submitted its final report relating to the voids in concrete in Unit 2 Containment Building exterior wall to the NRC on June 30, 1983.

GRS Company Profile

Visit of German Delegation of BMU and GRS to NRC

June 04 - 05, 2002

Organisation Chart

Strategic Group

[redacted]

Eastern Europe

Moscow Technical Office

Kiev Technical Office

Technical Divisions

Thermal Hydraulics	Operating Experience	Waste Management	Final Repository Safety Research
Containment	PWR Systems	Nuclear Fuel Cycle	Long Term Safety Analyses
Cooling Circuit	BWR Systems	Radiological and Environmental Protection	Geochemistry
Incident Analyses	Operational Safety	Final Storage	Geotechnics
Reactor Dynamics	Component Integrity		
Accident Analyses	Probabilistics		
Simulation Technology	Special Issues		

Central Divisions

Projects and International Programmes	Data Processing	Administration	Research Management
Analyses Project Controlling	DP-Applications	Finances and Controlling	Programmes and Facilities
Research Project Controlling	DP-Systems	Personnel and Legal Matters	Incidents and Components
International Programmes	Networks	Administration	Central Activities
Communication		Köln Garching Berlin Braunschweig	

*) jointly with IPSN / Riskaudit

9707204

Tasks, Objectives and Competence of GRS

Central technical and scientific expert organisation for nuclear safety and waste management in Germany

Task

Assess and improve the safety of technical facilities

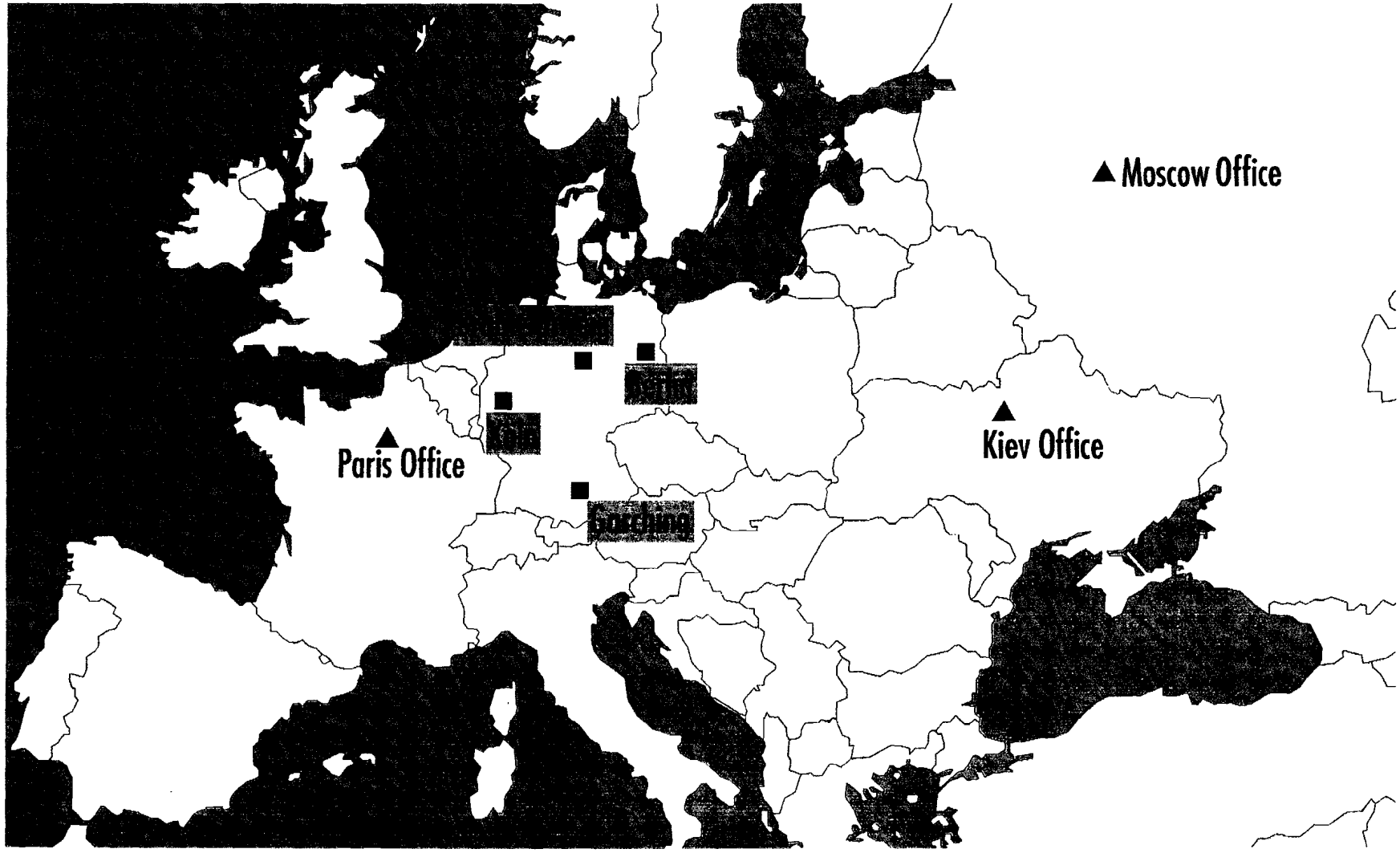
Objective

Protect man and the environment from the hazards of technology

Competence

- Interdisciplinary knowledge
- Advanced methods
- Qualified data

Company Locations and Technical Branch Offices



9707202

Governing Bodies of GRS

Shareholders' meeting

The shareholders are:

- the Federal Republic of Germany (46.1%)
- the Free State of Bavaria (3.85%)
- the *Land* of North Rhine-Westphalia (3.85%)
- the technical inspection agencies (TÜVs) and the Germanischer Lloyd (3.85 each, together 46.2%).

Supervisory board (12 members)

Chairman: Staatssekretär Rainer Baake

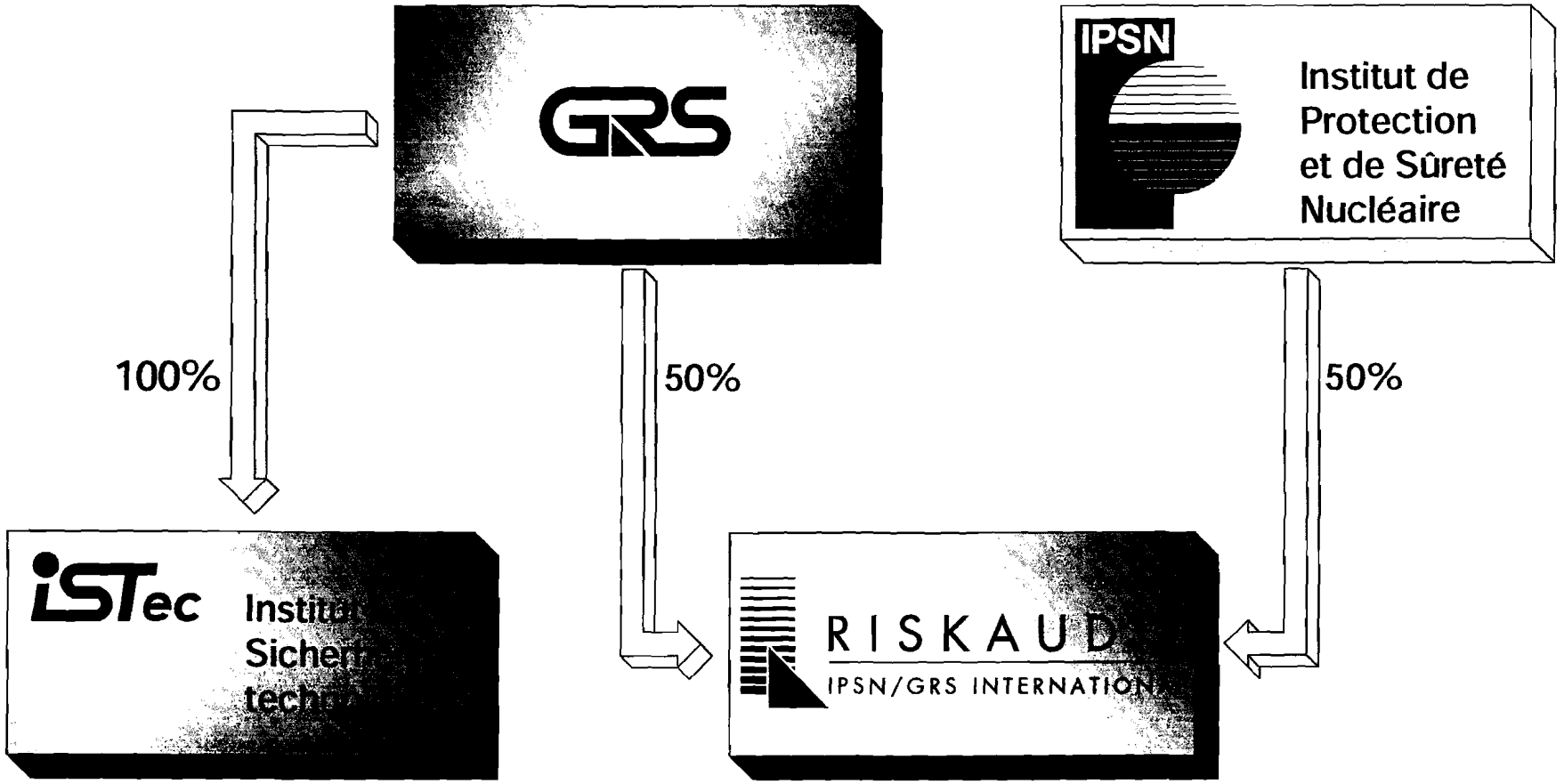
Vice-chairman: Prof. Dr.-Ing. Bruno O. Braun

Directors General

Dipl. Phys. Lothar Hahn

Dr. Walter Leder

Subsidiaries of GRS



9707210

Customers 2001

GRS is exclusively financed through contracts.

BMU/BfS 51%

Analyses of fundamental nuclear safety issues and ad-hoc assessments of current issues of power plant operation.

BMBF/ BMWi 33%

Research and development in the field of nuclear facilities and waste management, including the final storage of radioactive waste.

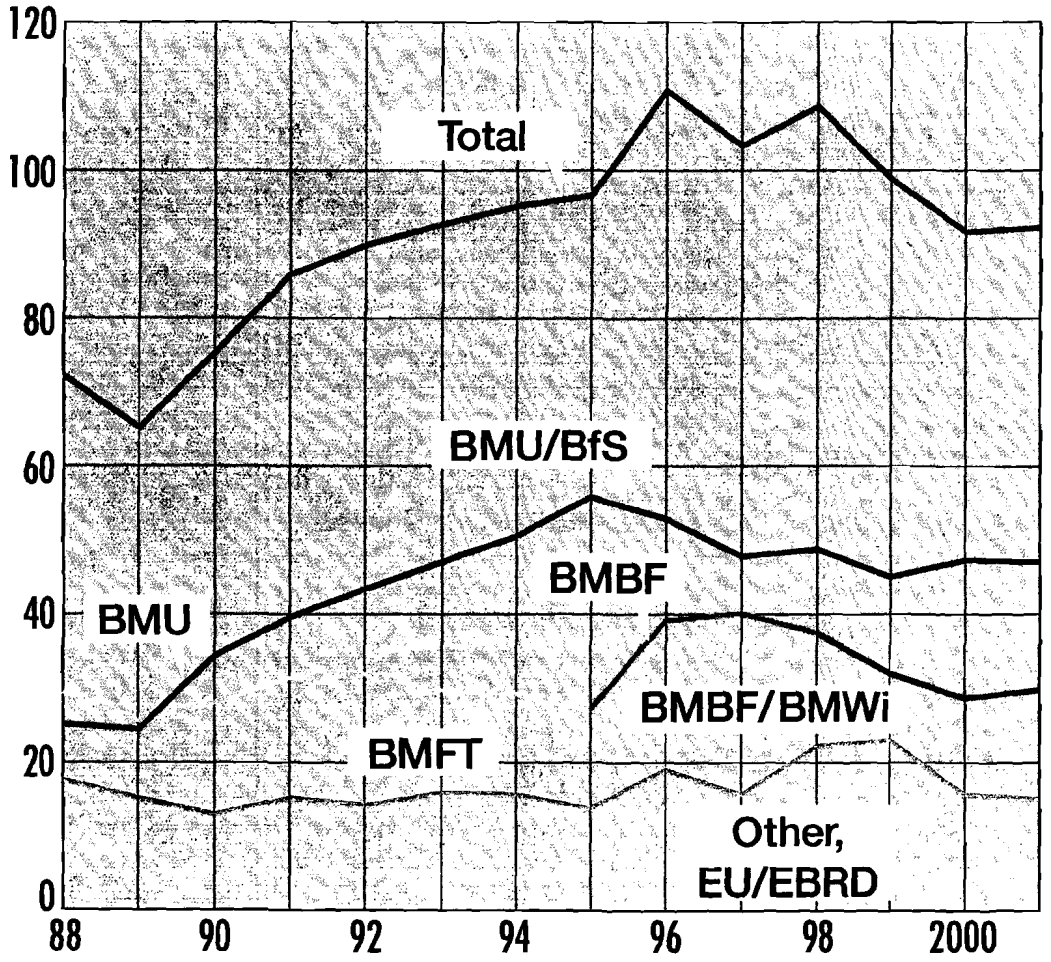
Other, EU/EBRD 16%

Safety analyses and assessments of nuclear facilities in Germany and abroad, providing expert advice to foreign supervisory and licensing authorities.

9707207

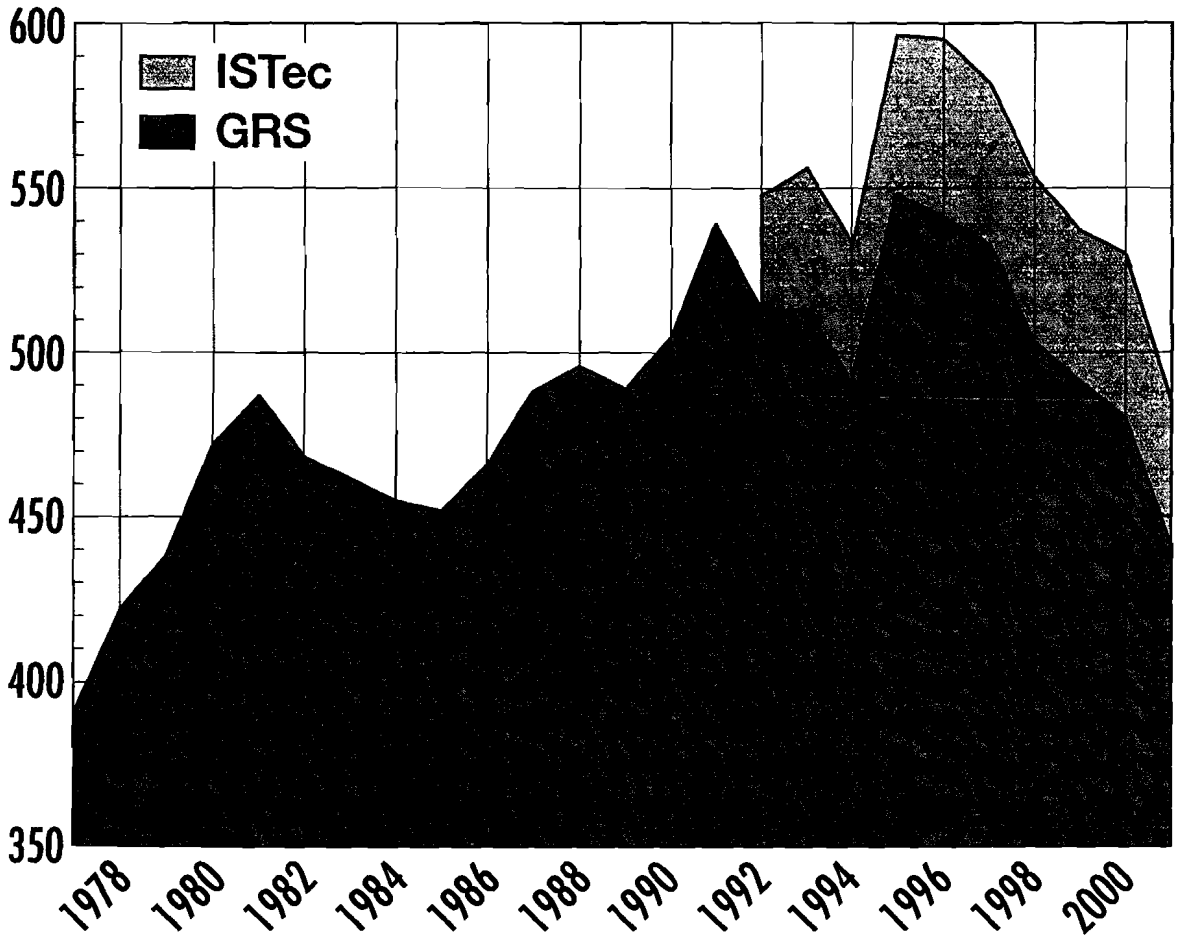
Volume of Contracts, Customers and Staff

In 2001, contracts to the amount of approx. DM 92m were awarded to GRS for its scientific and technical work. This work was performed by about 440 staff members, of which about 260 are scientists and engineers of such disciplines as:



- mechanical engineering
- electrical engineering
- physics
- nuclear engineering
- process engineering
- safety engineering
- civil engineering
- chemistry
- geochemistry
- geophysics
- mathematics
- informatics
- biology
- jurisprudence
- meteorology

Number of Staff Over the Years



In 2001, GRS and ISTec had around 480 staff, of which about 260 were scientists and engineers of such disciplines as:

- mechanical engineering
- electrical engineering
- physics
- nuclear engineering
- process engineering
- safety engineering
- civil engineering
- chemistry
- geochemistry
- geophysics
- mathematics
- informatics
- biology
- jurisprudence
- meteorology

Major Activities
Research and Development

- Development and verification of scientific software for the simulation of nuclear power plant behaviour under accident conditions
- Development of advanced methods for probabilistic risk assessment
- Development of simulators for investigations into the behaviour of complex technical systems and their man-machine interfaces
- Methods for the early diagnosis of mechanical failures (e.g. vibration analysis, loose-part monitoring)
- Development of methods for the assessment of the uncertainties of computer predictions
- Methods to assure and assess the quality of safety-relevant software
- Computer models for the performance assessment of final repositories
- Experiments concerning geological and geo-technical influences final repository safety
- Development of advanced safety concepts
- Development of information and documentation systems

Major Activities
Analyses, Assessments and Expert Opinions

- Safety analyses and assessments of nuclear facilities
  (e. g. nuclear power plants, waste repositories)
- Probabilistic risk analyses of complex technical systems
- Analyses and assessments of transport safety
- Safety analyses of facilities for the disposal of chemotoxic waste
- Analyses and assessments of specific technical safety issues
  (e. g. incident control, reactor physics, material issues, fire protection, safety of digital I&C, software reliability)
- Analyses and assessments of nature conservation issues
  (e.g. impact of mining-related radioactive contamination, restoration of contaminated industrial sites, questions regarding occupational radiation exposure)
- Monitoring and evaluation of national and international operating experience from nuclear and other technical facilities
  (e. g. common-mode effects, human factor analyses, precursor studies)

Joint R&D Activities of GRS within the German Reactor Safety Research Programme

Universities

R&D contributions to resolve individual phenomena

GRS

Basic and application-oriented R&D as a basis for sound scientific-technical safety statements

- Development and supply of codes and methods for safety evaluations of incidents and accidents
- Integral view and evaluation of R&D results

MPA, BAM, IzfP

Basic and application-oriented R&D in the area of component safety and material testing

Research Centres

Basic and developemental research for particular subjects, e.g.

FZK: Severe Accidents

FZJ: Passive Safety Systems

FZR: Reactor Dynamics

Industry

Development and operation-oriented research for design and optimisation of components and systems of reactor plants

9902501

GRS and its Partners in Western Europe

United Kingdom
- UK Atomic Energy Authority (AEA Technology)

Netherlands
- Kernfysischer Dienst (KFD)

Belgium
- AIB-Vincotte Nuclear (AVN)

France
- Institut de Protection et de Sûreté Nucléaire (IPSN)
- Agence Nationale pour la Gestion des Déchets Radioactifs (ANDRA)

Spain
- Empresa Nacional des Residuos Radioactivos SA (ENRESA)
- Consejo de Seguridad Nuclear (CSN)

Finland
- Imatran Voima (IVO)
- Finnish Centre for Radiation and Nuclear Safety (STUK)
- Technical Research Centre of Finland (VTT)

Sweden
- Swedish Nuclear Power Inspectorate (SKI)

Switzerland
- National Co-operative for the Storage of Radioactive Waste (NAGRA)
- Swiss Federal Nuclear Safety Inspectorate (HSK)

Italy
- Agenzia Nazionale per la Protezione dell' Ambiente (ANPA)

Turkey
- Türkiye Atom Enerjisi Kurumu (TAEK)

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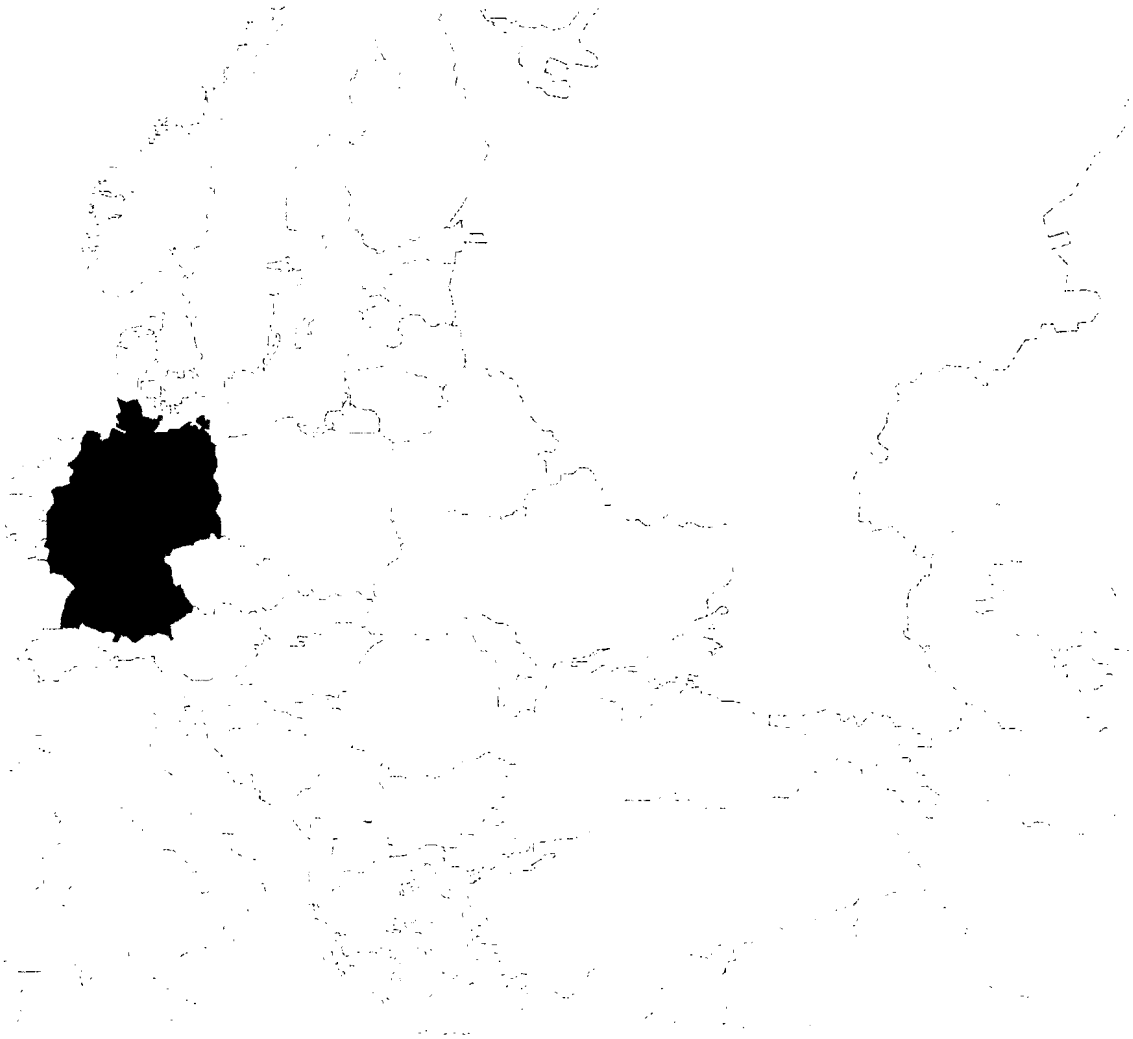
GRS and its Partners in Eastern Europe (1)



Russia

- Russian Research Centre Kurchatov Institute, Moscow
- Rosenergoatom (REA)
- Balakovskaya AES
- The Russian State Committee on Nuclear Safety and Radiation Protection
- Institute for Power Engineering NIKIET
- Experimental Design Bureau Gidropress (OKB GP), Podolsk
- Atomenergoproject AEP, Moscow
- Russian Academy of Sciences, Nuclear Safety Institute (IBRAE), Moscow
- Gosatomnadzor of Russia
- Scientific and Engineering Centre on Nuclear and Radiation Safety (SEC NRS, expert organisation of Gosatomnadzor of Russia)

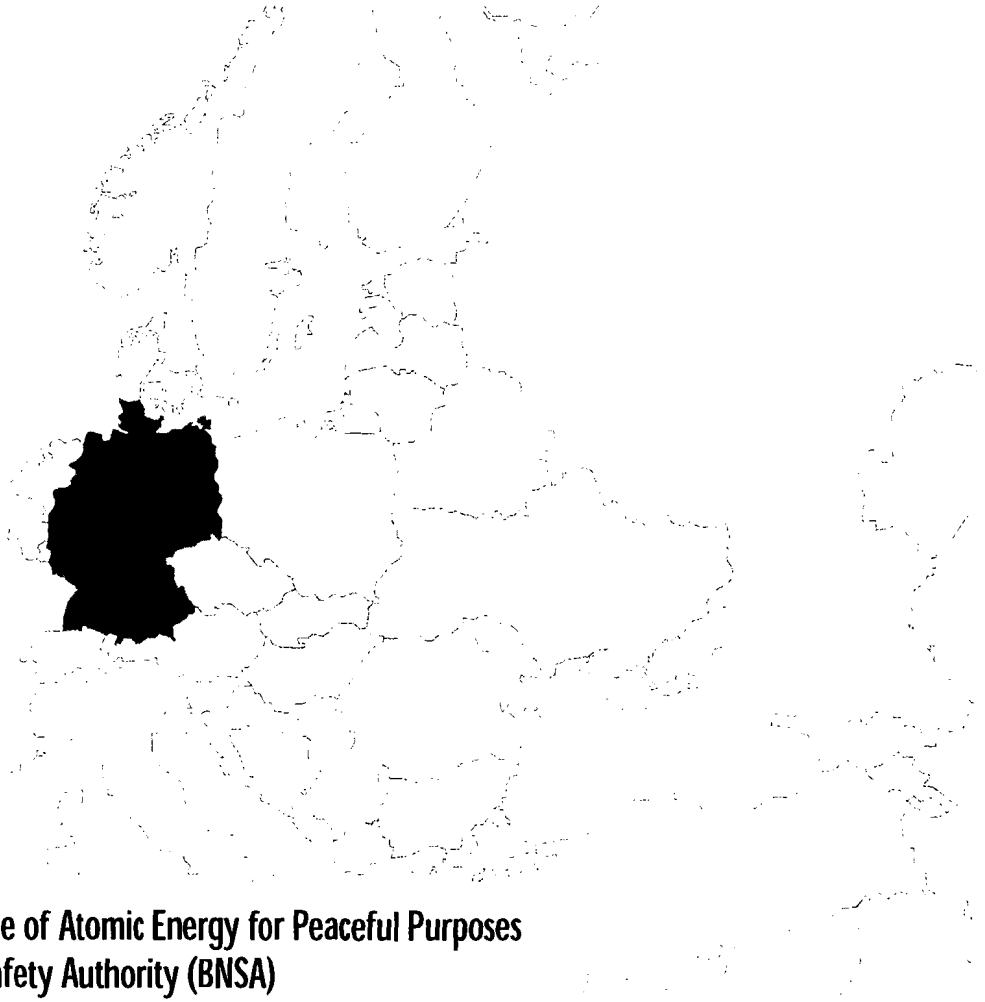
GRS and its Partners in Eastern Europe (2)

Czech Republic
- State Office for Nuclear Safety (SONS)
- Nuclear Research Institute Rez (NRI)

Slovak Republic
- Nuclear Regulatory Authority of the Slovak Republic

Hungary
- Hungarian Atomic Energy Commission
- Atomic Energy Research Institute (AERI)

Bulgaria
- Committee on the Use of Atomic Energy for Peaceful Purposes
- Bulgarian Nuclear Safety Authority (BNSA)



Lithuania
- Lithuanian Nuclear Power Safety Inspectorate (VATESI)
- Lithuanian Energy Institute (LEI)
- Nuclear Regulatory Authority (NRAUI)

Ukraine
- State Scientific-Technical Centre (SSTC, expert organisation of the NRA Ukraine)

Armenia
- Ministry of Energy and Fuel, Department "Armatomenergo"
- Armenian Nuclear Regulatory Authority

International Co-Operation of GRS in the Area of Reactor Safety Research

Bilateral research projects with CEEC&NIS within the frame of STC of BMWi

Russia,
Ukraine,
Czech Republic,
Slowak Republic,
Hungary, Bulgaria

Contracts for bilateral co-operation between

France,
Great Britain,
USA,
Japan,
Republic of Korea

Bilateral contracts for usage of GRS-Codes

with a total of 19 countries in Europe, Asia, America, South Africa

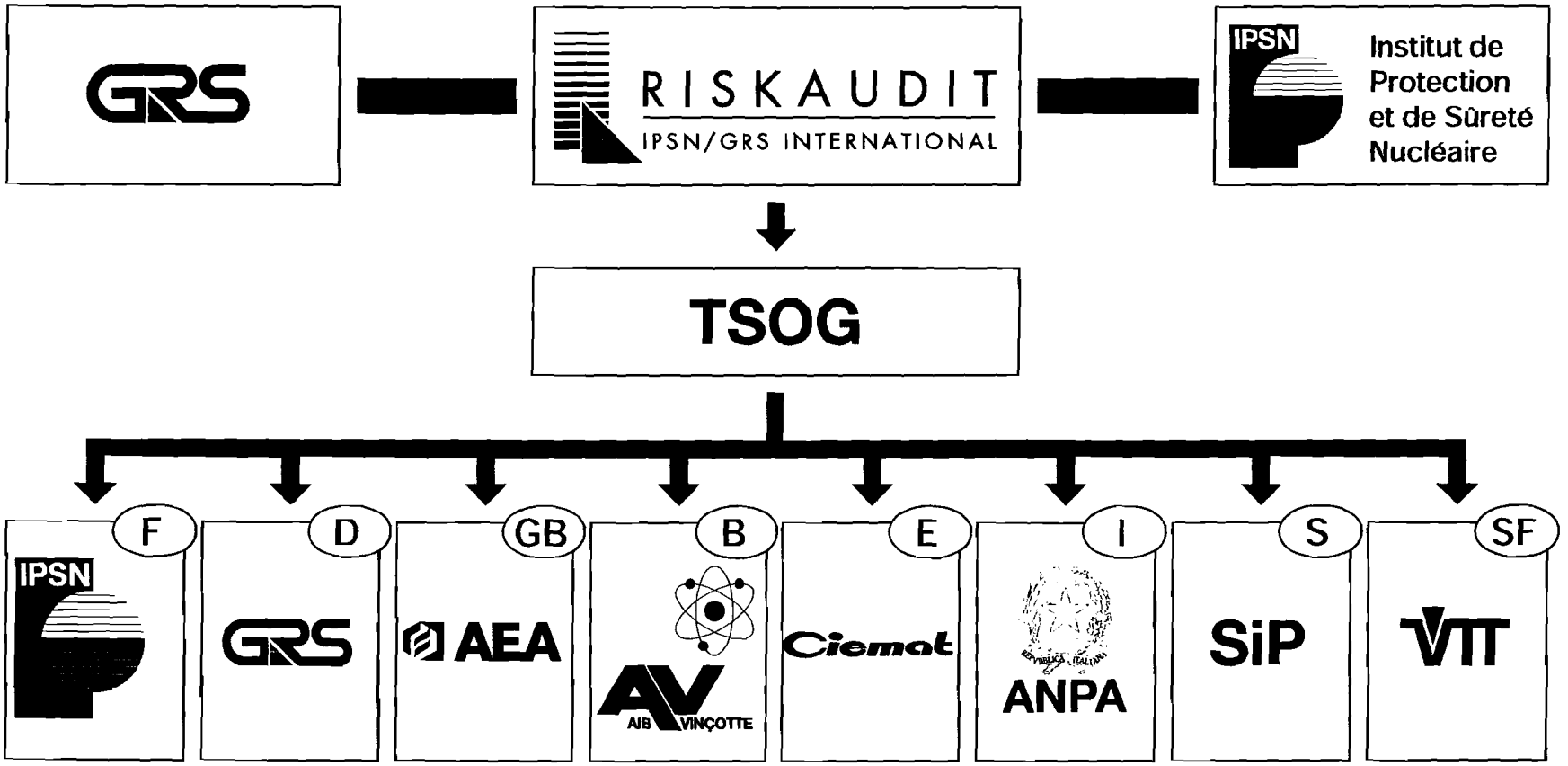
Multilateral contracts related to R&D projects of the 4th EU framework programme

11 EU member countries

Multilateral scientific technical co-operation within the frame of OECD-NEA

26 member countries in Europe,
North America,
Asia,
Australia

Relationship between GRS-IPSN and other European Technical Safety Organisations:



9717401

AGENDA ITEM STATEMENT

Meeting No.	Agenda Item
493	11

PLANNING PROCESS FUTURE ACTIVITIES

Subject:

Action to be taken by Commission

Brief Description of Action:

Recommendation Prepared by:

1. Program Manager
2. Program Staff
3. Planning Staff

Date Prepared:

Staff Contact:

G:PlanPro(ACRS):ppmins.493
June 4, 2002

SUMMARY/MINUTES OF THE
ACRS PLANNING AND PROCEDURES MEETING
THURSDAY, MAY 30, 2002

The ACRS Subcommittee on Planning and Procedures held a meeting on Thursday, May 30, 2002, in Room T 2 B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 10:00 a.m. and adjourned at 12:00 noon

ATTENDEES

MEMBERS

G. Apostolakis, Chairman
M. Bonaca
T. Kress
J. Sieber
S. Rosen

ACRS STAFF

J. T. Larkins
S. Bahadur
H. Larson
S. Duraiswamy
S. Meador
J. Gallo

NRC STAFF

I. Schoenfeld

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

Member assignments and priorities for ACRS reports and letters for the June ACRS meeting are attached (pp. 10-12). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the June 2002 ACRS meeting be as shown in the attachment (pp. 10-12).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through September 2002 is attached (pp. 10-12). 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 discussed and developed recommendations on the items that require Committee decision, which are included in Section II of the Future Activities list (p. 13).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate. The Committee should decide on the Subcommittee's recommendations on items in Section II of the Future Activities List.

In addition, the Subcommittee recommends that Dr. Powers review the major ACRS comments and recommendations included in previous ACRS reports to the Commission (NUREG-1635, Vols. 1-4) on the NRC Safety Research Program and determine the significance of ACRS impact on the NRC Safety Research Program.

3) Quadripartite Meeting Update

The Quadripartite meeting is scheduled to be held on October 23-25, 2002, in Berlin, Germany. On May 16, 2002, RSK sent the latest agenda, and we have written in some suggested changes (pp. 15-24) for this meeting.

As confirmed at the April meeting, the following technical papers have been prepared by cognizant members for discussion at the Quadripartite meeting:

- Safety Culture and Safety Management (Bonaca/Powers)
- Risk-Informed Regulation (Apostolakis/Kress)
- Thermal-Hydraulic Analysis and Code Issues (Wallis/Ransom)
- Stress Corrosion Cracks in Pressure Retaining Components in Nuclear Power Plants (Ford/Shack)
- Risk Analysis of Spent Fuel Storage (Kress/Powers)

Dr. Larkins has informed the Commission on the Quadripartite meeting, including forwarding the draft agenda. We are coordinating this trip with the Office of International Programs.

In connection with the Quadripartite meeting trip, several members agreed to visit a MOX facility in France prior to the meeting in Berlin, Germany.

RECOMMENDATION

The Subcommittee recommends the following:

- The Committee should discuss and approve the technical papers prepared by cognizant members during the July 2002 ACRS meeting. Cognizant members should complete the papers and send them to other members for review and comment by June 17, 2002. drafts by June 21

- Dr. Larkins should keep the Committee informed of the travel arrangements to the MOX facility in France and to the Quadripartite meeting in Germany.

- Dr. Apostolakis and Dr. Larkins should get feedback from the NRC Chairman on whether or not there is a concern with the level of participation in the Quadripartite meeting.

- Dr. Larkins should inform the RSK that Dr. Apostolakis would like to Chair Session 2, "Safety Culture and Safety Management," instead of Session 3, "PSA/PSR/Risk-Informed Regulation," and that Dr. Kress would like to Chair Session 3.

4) Staff Requirements Memorandum

In a Staff Requirements Memorandum (SRM) (p. 25-26) dated December 20, 2001, resulting from the ACRS meeting with the Commission on December 5, 2001, the Commission requested the following:

- The ACRS should continue to review staff efforts on risk-based PIs and improvements to the significance determination process.

- The staff, with ACRS input, should provide recommendations for resolving, in a transparent manner, apparent conflicts and discrepancies between aspects of the revised reactor oversight process that are risk-informed (e.g., significance determination process) and those that are performance-based (e.g., performance indicators).

- The ACRS should continue its efforts to ascertain regulatory challenges for future reactor designs. The Committee should also ensure that it is prepared to review NRC staff efforts on advanced reactors in the near term, including issues related to Westinghouse's AP1000, General Atomics' GTMHR, and Exelon's pebble bed modular reactor.

During the February 2002 ACRS meeting, it was agreed that cognizant members should develop a plan for addressing the ROP issues in the SRM after the staff has provided its

proposal. In May 2002, the NRC staff has developed SDP improvement strategies and associated SDP improvement Task Action Plan.

During the May meeting, Dr. Kress provided a draft plan to address the SRM issues on future plant designs. He is in the process of revising this plan with assistance from the cognizant staff engineer, Dr. El-Zeftawy. In the revised plan, he will include ACRS review of early site permit applications.

request CDROMS

As suggested by the Committee during its May 2002 meeting, Dr. Apostolakis and Mr. Sieber met with representatives of the NRC staff on May 30, 2002, to discuss staff plans to address the ROP issues specified in the SRM.

RECOMMENDATION

The Subcommittee recommends that Dr. Apostolakis and Mr. Sieber provide a report to the Committee regarding their meeting with the staff on May 30, 2002, and also develop a plan for addressing the ROP issues in the SRM.

5) ACRS Meeting with the NRC Commissioners

The ACRS is scheduled to meet with the NRC Commissioners on Wednesday, July 10, 2002, between 2:00 and 4:00 p.m. The topics have been approved by the Commission (p. 27):

(1) Overview by the ACRS Chairman (GEA/JTL)

- Status of ACRS activities on power uprates and license renewal

(2) Advanced reactor designs (TSK/MME)

(3) Risk-Informing Special Treatment Requirements of 10 CFR Part 50 (GEA/AWC)

(4) PTS reevaluation project (FPF/RBE)

As a result of delays in NRC staff briefings on some of these issues (PTS reevaluation project risk acceptance criteria and, Human Reliability Research Plan), the Committee at the May ACRS meeting discussed whether changes should be made to the proposed list of topics. The Committee agreed to drop only the topic on Human Reliability Research Plan, and all other topics remain the same subject to approval by the Commission.

On May 28, 2002, the ACRS/ACNW Office was informed that the Commission has agreed to the topics proposed by the ACRS, except it needed additional information on the MOX topic to better understand the scope of the presentation. Additional

information was provided to the Commission on May 29, 2002, and the Commissioners subsequently decided to drop this item from the list of meeting topics .

During the May meeting, the Committee recommended that cognizant members prepare proposed slides and send them to Dr. Sher Bahadur by May 17, 2002. Proposed slides provided by the members are attached (pp 28-52).

RECOMMENDATION

The Subcommittee recommends that the Committee review and approve the slides during the June 2002 meeting.

4) Celebration of the 500th ACRS Meeting

As agreed to by the members, invitations were sent to the NRC Commissioners to participate at the 500th ACRS meeting ceremony, which is scheduled for March 4-5, 2003. (This is also coincidental with the Committee's 50th Anniversary.) So far, NRC Chairman Meserve, Commissioner Dicus, and Commissioner McGaffigan have agreed to participate. Invitations have also been sent to those who are expected to serve as panel members.

Commissioner McGaffigan has expressed his appreciation for changing the agenda to permit him to chair the Panel on "Significant Contributions of ACRS and its Future Directions" instead of the Session on "Proactive Initiatives."

Dr. Hal Lewis, Dr. Robert Seale, Dr. Bill Stratton, and Mr. Dave Ward have agreed to participate in the celebration. The Committee previously agreed to invite foreign advisory committee members to participate in this program.

RECOMMENDATION

The Subcommittee recommends that Mr. Herb Kouts be invited to participate in this celebration and that the Executive Director keep the Committee informed of further developments. The Committee was unable to decide who should be invited from a foreign advisory committee and decided not to invite a foreign dignitary. The ACRS Executive Director was requested to invite the NRC's EDO. The Committee needs to decide who to invite from industry.

5) ACRS Senior Fellow Position

On behalf of the full Committee, the Planning and Procedures Subcommittee members interviewed one candidate in March and another one in April 2002. Subsequently, one candidate accepted a position outside NRC. We have offered the job to Dr. Hossein Nourbaksh who has a Ph.D. in chemical engineering. He is the president and founder of Energy and Environmental Science, Inc. consulting company.

Dr. Nourbaksh has accepted the job offer and will report to work subsequent to completion of all necessary paper work.

Bertram Wolf
Draper
Poindexter
O'Hanlan
Luckman
Beedle
Pate/Edmonds
McIntosh
Counsel
Hinze

Gene Hughes
Dennis Bley
Karl Fleming

RECOMMENDATION

The Subcommittee recommends that ACRS management keep the Committee informed of further development regarding Dr. Nourbaksh's status. Also, the members should identify issues on which the Committee can make major contributions to the Commission to enhance the regulatory programs. These issues will be discussed and a consolidated list of issues will be prepared by the Planning and Procedures Subcommittee. Subsequently, the ACRS management should get a qualified person(s) under contract to work on the issue(s) and develop a report(s) to the Committee. The Committee should discuss the report(s) and develop comments and recommendations to the Commission, as warranted.

8) Arrangements for the Visit to Watts Bar Nuclear Plant and Region II

Members are scheduled to tour the Watts Bar Nuclear Power Plant on June 18 and hold a meeting with Region II personnel on June 19, 2002. Arrangements for the Watts Bar tour and meeting with Region II personnel have been finalized (p. 53). Mag Weston will provide an itinerary for this trip along with draft agendas for the meetings at Watts Bar and Region II.

RECOMMENDATION

The Subcommittee recommends that the members take note of the arrangements.

9) Revised ACRS Subcommittee Structure

A revised Subcommittee structure is attached (pp. 54-70). This revision reflects creation of a new Subcommittee on Safeguards and Security, establishment of two Plant License Renewal Subcommittees (approved by the Committee in 2001), and member/staff reassignments. This revision was provided to the ACRS Chairman, members, and ACRS staff for review and comment. Comments received were incorporated, as appropriate. The ACRS Chairman has approved this revision. It will become effective on June 10, 2002.

10) Workshop on Nuclear Regulatory Decisionmaking Process in Switzerland

The Swiss Federal Safety Inspectorate (HSK), in coordination with the International Atomic Energy Agency (IAEA), is organizing a Workshop on Nuclear Regulatory Decisionmaking Process (pp. 71-77) to be held in Switzerland on October 13-16, 2002. As a result of the Planning and Procedures Subcommittee discussion in April, the Committee asked the Executive Director to contact HSK to see if this workshop could be held on October 16-18 instead of October 13-16, 2002. The HSK has accommodated our request. Additionally, they have included ACRS participation in 2 places in the agenda. One would be a presentation on ACRS Perspective on Regulatory Decisionmaking and the second is a Panel Discussion. Both are on Wednesday, October 16, 2002. At the May meeting, the Committee had approved Drs. Bonaca, Kress, and Ransom to participate at this Workshop. In addition, the Committee stated that Dr. Kress should prepare a paper, for Committee review and presentation at the

workshop, on the ACRS Perspectives on Regulatory Decisionmaking and that Dr. Bonaca should represent the ACRS on the Panel discussion on Issues and Opportunities for the Evaluation and Improvement of Nuclear Regulatory Decisionmaking Process.

RECOMMENDATION

The Subcommittee recommends that Dr. Kress with assistance from Dr. Apostolakis provide a paper on ACRS perspectives on Regulatory Decisionmaking process for Committee discussion and approval at the July 2002 ACRS meeting and that Dr. Bonaca provide an outline of the points that he plans to make at the Panel discussion.

11) Member Compensation Reports

The agency is operating under a new time and labor reporting system, Starfire. Sooner rather than later, the time by labor category reported in that system will be used to make management decisions regarding each office's contribution to the agency's strategic goals. Members' time by labor category must be reported in the agency's Starfire system so that the true ACRS/ACNW office contribution is realized. To accomplish this, a data entry clerk from the Payroll Office will use the members' compensation sheets to enter each members' time by labor category into the Starfire system. Since the system has standardized reporting categories referred to as "activity codes," it is very important that the members' time be captured by the same activity codes when reporting their time. Each member will receive a laminated card with the labor category title and corresponding activity code as well as new compensation forms. The information on the laminated card (also on the backside of the compensation report) must be used when filling out the member compensation report. In addition, time must be attributed to an activity code for all Committee meetings and for preparation time. The nature of the work such as travel, meeting preparation, type of meeting must also continue to be identified.

RECOMMENDATION

The Subcommittee recommends that effective immediately, members use the activity codes and categories when reporting their time. Time claimed for Committee meetings and preparation time must be reported by subject matter rather than listed as "prep time" or "subcommittee/full Committee meeting." See attachment (pp. 78-79) for the listing of categories and activity codes.

12) Member Issues

Role and Use of PRA in the ACRS Review Process

Mr. Sieber provided several comments on the Committee's use of PRA results in its review of regulatory issues (e.g., power uprates, license renewal), including the following:

- The Committee tries to use risk information in making decisions on regulatory matters. He continues to believe that risk information is one of the tools we can use to decide what is correct.

- The Commission's Safety Goal Policy defines "what is safe enough." The issue of the use of PRA in relation to the Safety Goal Policy seems to continually arise. In his view, the ACRS has addressed this issue only in a fragmented way.

- Adding individual opinions on PRA and Safety Goal Policy to our reports on licensee applications is not the appropriate avenue to transmit our concerns to the staff or the Commission.

Mr. Sieber recommends that a "White Paper" be prepared by using an ACRS fellow to address the adequacy of PRAs and issues associated with the Safety Goal Policy. The Committee should discuss this matter during its 2003 retreat and subsequently develop comments and recommendations to the Commission as warranted.

In addition, Dr. Powers discussed several issues (p. 80) about how PRAs are used in License Renewal, Power Uprates and in the future. Basically, Dr. Powers suggests that because of enhanced computational capability that the ACRS could propose that PRAs could be expanded to cover issues and phenomena that are either excluded or treated in a less than a complete manner.

RECOMMENDATION

The Subcommittee recommends the following:

- The Subcommittee recommends getting a contractor to prepare a draft "White Paper" addressing issues raised by Sieber, Apostolakis, and Powers. The draft paper should be subsequently finalized by the Committee.

- This paper could be prepared in late fall of this year and discussed at the November/December 2002 meeting.

13) Travel Request

Dr. Ford has requested Committee approval and support to attend the Fifth International Symposium on "Contribution of Materials Investigation to the Resolution of Problems Encountered in Pressurized Water Reactors," which is scheduled to be held in Fontenraud, France on September 23-27, 2002.

RECOMMENDATION

The Subcommittee recommends that the Committee defer approving Dr. Ford's travel request until after the Commission has approved the ACRS/ACNW trip to attend the Quadripartite meeting in Berlin, Germany, that is scheduled for October 23-25, 2002.

ANTICIPATED WORKLOAD
June 6-8, 2002

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	All Members	Larkins	Preparation for meeting with the NRC Commissioners on July 10, 2002.	--	--	--
Bonaca	--	Cronenberg/Weston	Development of Reliability/Availability Performance Indicators and Industry Trends.	B	--	--
Ford	--	Weston	CRDM cracking of vessel head penetrations and vessel head degradation	A	To provide timely advice to the Commission	--
		El-Zeftawy	Format and content for the 2003 ACRS report to the Commission on the NRC Safety Research Program	--	--	--
Kress	--	El-Zeftawy	Technical and Policy Issues Related to Advanced Reactors	B	To provide early feedback	--
		Elliott/Larson	Proposed Resolution of GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident"	A	To meet the CTM schedule	1
Leitch	--	Kobetz/Duraiswamy	Technical Assessment of GSI-168, "Environmental Qualification of Low-Voltage I&C Cables"	A	To meet the CTM schedule	1
Rosen	--	Elliott/Duraiswamy	Proposed Rulemaking to endorse NFPA 805 Fire Protection Standard	A	To meet the CTM schedule	--

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ANTICIPATED WORKLOAD
July 10-12, 2002

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	All Members	Larkins	Meeting with the NRC Commissioners (7/10, 2:00-4:00 p.m.)	--	--	--
		Cronenberg/Larson	Risk-Informed Regulation Implementation Plan	B	To provide feedback	--
Ford		Cronenberg/ El-Zeftawy	Application of Fracture Mechanics Methods to Reactor Vessel Integrity Assessment	--	--	--
		Savio	Format and content for the 2003 ACRS report on the NRC Safety Research Program **[If not completed in June]**	--	--	--
Kress	Ford	Weston	PTS Reevaluation Project: Risk Acceptance Criteria	A	To meet CTM schedule	--
		El-Zeftawy	Advanced Reactors Research Plan	B	To provide feedback	
Powers	--	Boehnert	Proposed Resolution of GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs	A	To meet the CTM schedule	
		Kobetz/Duraiswamy	Overview of NRC Research Activities in the Seismic Area	--	--	--
Wallis	Sieber	Boehnert	Development of Review Standard for Reviewing Core Power Uprate Applications **[Information Briefing]**	--	--	--

ANTICIPATED WORKLOAD
September 12-14, 2002

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Cronenberg/Larson	Risk-Informed 10 CFR Part 50 Pilot Program (Option 2)	A	To meet the CTM schedule	
		Cronenberg/ Duraiswamy	Draft Final ANS Standard ion External Events PRA Methodology	A	To meet the CTM schedule	
Leitch	Bonaca	Kobetzt/Duraiswamy	North Anna and Surry License Renewal Application	A	To meet the CTM schedule	
Powers	--	El-Zeftawy	Human Reliability Analysis Research Plan	B	To provide feedback	
		Kobetz/Larson	Draft ANS Standard on Low-Power and Shutdown Operations PRA	B	To provide feedback	
Sieber	--	Boehnert	Virginia Class Nuclear Propulsion Plant Submarine Design **[CLOSED]**	A	To meet the CTM schedule	
Wallis	--	Boehnert	Framatone ANP Richland, Inc. S-RELAP5 Realistic Large-Break LOCA Code	A	To support licensing review	
		Boehnert/Weston	Draft Final Regulatory Guide (DG-1096) and SRP Section Associated with NRC Code Reviews	A	To meet the CTM schedule	

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II. ITEM REQUIRING COMMITTEE ACTION

1. Draft Final Revision-1 to Regulatory Guide 1.174, "An Approach to Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (GEA/AWC) ESTIMATED TIME: 1 hour.

   Purpose: Determine course of action.

   During the 483rd meeting of the ACRS (June 6-8, 2001) the Committee considered the proposed revision to Regulatory Guide 1.174 and associated revisions to SRP Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific Risk-Informed Decision Making". In the Larkinsgram, dated June 12,2001, it was stated that the Committee plans to review the proposed final version of RG 1.174 and SRP Chapter 19 following the reconciliation of public comments. During the February 7-9, 2002 ACRS meeting, the staff was scheduled to brief the Committee on the draft final version of the RG 1.174, Rev. 1. However, the staff only briefed the Committee on a modified approach for endorsing the industry standards on PRA quality. The staff did not provide a final version of RG 1.174, Rev. 1 for ACRS review. Additionally, the Committee has not considered SECY-02-0070, dated April 24, 2002 (Policy Issue Memorandum from Travers to the Commission, "Publication of Revisions-1 to Regulatory Guide 1.174 and SRP Chapter 19 and Notice of Staff Plan for Endorsing Consensus Probabilistic Risk Assessment Standards and Industry Peer Review Programs". The staff would like to know whether ACRS has any problems with issuing the draft final version of Regulatory Guide 1.174, Revision 1 and the associated SRP Chapter 19.

   The Planning & Procedures Subcommittee recommends that Dr. Apostolakis propose a course of action.

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From: Guenter Weimer <gweimer@bfs.de>
To: <corentin.ledoare@ipsn.fr>, <JTL@nrc.gov>, <ynagano@op.cao.go.jp>
Date: 5/24/02 6:42AM
Subject: quadripartite meeting

Dear Sirs,

on behalf of RSK I send you a summarised and actualised information on the meeting.

For individual details already exchanged please see my previous e-mails to you, respectively.

Thanks in advance for your ongoing engagement.

Kind regards

Guenter Weimer

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RSK-Geschäftsstelle
c/o Bundesamt für Strahlenschutz
Hermann-Ehlers-Straße 10
53113 Bonn
Tel.: +49-228-305-37 20
Fax: +49-228-67 03 88
e-mail: gweimer@bfs.de

Groupe Permanent „Reacteurs" (GPR),
Groupe Permanent „Dechets" (GPD),
Groupe Permanent „Transports" (GPT),
Groupe Permanent „Usines" (GPU),
attn. **Mr Corentin Le Doaré**

Nuclear Safety Commission (NSC)
attn. **Mr. Yuko Nagano**

Advisory Committee on Reactor Safeguards
(ACRS),
Advisory Committee on Nuclear Waste (ACNW)
attn. **Mr. John Larkins**

23.05.2002

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Quadripartite Meeting of the National Advisory Committees of France, Germany, Japan and the USA
October 23 – 25, 2002, Berlin (Germany)

Dear Sirs,

on behalf of Mr. Michael Sailer, chairman of the German Commission on Reactor Safety (RSK), the secretariat of RSK would like to present you some actual information on the meeting.

The actualized program of the meeting includes your comments given so far (attachment).
For some of the presentations envisaged by RSK-members scope and structure have been prepared (attachments). The corresponding papers will be developed until July/August and forwarded to the participants.
To allow sufficient discussion in each session the oral presentations should possibly take 10 - 15 minutes.

The RSK suggests that the meeting should provide a forum for discussions and exchange of experience in particular among the members of the partner committees or commissions. But, each committee or commission should decide on participation of members of competent national authorities and their contribution at the meeting. We planned for up to 20 participants from each country.
Please note that requests for changes in chairman and co-chairmanship are welcomed - e. g. to take some of the co-chairmanships of RSK-members off - and in particular RSK appreciates proposals for chairmanship of session 1 and session 2.

Arrangements for the meeting, e. g. accommodations will be prepared by the secretariat of RSK. A contingent of single and double rooms has been reserved at the conference hotel

Maritim pro Arte Hotel Berlin
Friedrichstraße 151
Tel.: +49-30-20335
Fax: +49-30-20334209
e-mail: info.bpa@maritim.de
internet: www.maritim.de.

Please give us a short term notice of time period, type of rooms and number of persons respectively – expenditure will be about 250 EURO per night and person in a single room including breakfast and – for meeting participants – including lunch and coffee breaks.

Information on travelling to Berlin and to the Hotel and on city transport is available (in English) at www.bfs.de/presse/va/decomm02.

We will appreciate your interest and we hope the information given will facilitate your further preparations and decisions.

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Kind regards

Guenter Weimer

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RSK-Geschäftsstelle
beim Bundesamt für Strahlenschutz

16.05.2002

DRAFT

Quadripartite Meeting of the National Advisory Committees of France, Germany, Japan and USA

France: Groupe Permanent "Reacteurs" (GPR),
Groupe Permanent "Dechets" (GPD),
Groupe Permanent "Transports" (GPT),
Groupe Permanent "USINES" (GPU)

Germany: Reaktor-Sicherheitskommission (RSK)

Japan: Nuclear Safety Commission (NSC)

USA: Advisory Committee on Reactor Safeguards (ACRS),
Advisory Committee on Nuclear Waste (ACNW)

and invited representatives of Swiss and Swedish advisory committees

October 23 – 25, 2002, Berlin (Germany)

Maritim pro Arte Hotel Berlin

Wednesday, October 23

09:00 **Opening Remarks** — Sailer(RSK-Chairman), Renneberg(BMU)[1]

09:15 Session 1
Response to terrorist attack
Main topic: generic issues of commercial airplane crash
Chairman: N. N. (N. N.)
Co-Chairman: Schneider (RSK)
announced papers: Schneider
member GPR depending on briefing June 6, 2002

[1] Federal Ministry for the Environment. Nature Conservation and Nuclear Safety, Germany

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10.30 Coffee Break

11:00 Session 2
Safety Culture and Safety Management
Chairman: ~~N. N. (N. N.)~~ George Apostolakis (ACRS)
Co-Chairman: Wieland (RSK)
announced papers: Wieland (Safety Culture: Requirements, Indicators, Assessment and Reporting)
~~member~~ ACRS Mario Bonaca/Dana Powers, "Safty Culture and Safety Management"
member GPR depending on briefing June 6, 2002

12:30 Lunch

14:00 Session 3:
PSA/PSR/Risk informed Regulation
Chairman: ~~Apostolakis~~ (ACRS) Thomas Kress
Co-Chairman: Hahn (RSK)
announced papers: Hahn (Role of PSA, Periodic Safety Review, Risk Considerations in Regulatory Process, Potential and Limitations of Risk Informed Regulation)
Apostolakis/Kress (ACRS), "Risk-Informed Regulation"
member GPR depending on briefing June 6, 2002

16:00 Guided tour of Berlin or tour to Castle Sanssouci (Potsdam)

Thursday, October 24

09:00 Session 4
Thermal Hydraulic Analysis and Code Issues
Chairman: Wallis (ACRS)
Co-Chairman: Kersting (RSK)
announced papers: Kersting (Challenges to Development, Assessment and Application of Thermo-Hydraulic Codes in Germany)
Wallis/Ransom (ACRS), " Thermal-Hydraulic Analysis and Code"
member GPR depending on briefing June 6, 2002

10:30 Coffee Break

11:00 Session 5
Stress Corrosion Cracks in Pressure Retaining Components in NPP
Chairman: Speidel (RSK)
Co-Chairman: Schuler(GPR)
announced papers: Speidel
~~member~~(ACRS) Ford/Shack, " Stress Corrosion Cracks in Pressure Retaining Components in Nuclear Power Plants"
member GPR depending on briefing June 6, 2002

12:30 Lunch

14:00 **Actual issues** (incidents Brunsbuettel, Hamaoka, Davis Besse; fuel element damage at Cattenom; ...)

15:30 Coffee Break

16:00 **Summary and Conclusion of sessions 1 – 5, preparation for plenary session** ← Possibly ACNW Members will Join the meeting

19:00 Dinner on invitation of RSK

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Friday, October 25

09:00 Session 6
Safety of Spent Fuel Storage
Chairman: Guillaumont(GPD)
Co-Chairman: Thomas (RSK)
announced papers: Guillaumont
Sailer (Dry Interim Storage of Spent Fuel Elements in Storage Casks)
~~member~~(ACRS)(Risk Analysis of Spent Fuel Storage)
Kress/Powers

10:30 Coffee Break

11:00 Session 7
Waste Disposal Concepts; Performance Assessment for the Disposal;
Safety Assessment of Final Repositories
Chairman: Thomas (RSK)
Co-Chairman: Devillers(GPD)
announced papers: Storck/RSK-subcommittee (Status of German Research for Final Disposal)
member GPD
~~probably member~~(ACNW), "Total System Performance Assessment"
Garrick

12:30 Lunch

14:00 Session 8
Transport of Spent Fuel and Waste
Hornberger would like to chair Session 7, if possible.
Chairman: Hornberger(ACNW)
Co-Chairman: Sailer (RSK)
announced papers: Drotleff/RSK-subcommittee (amended regulations, experience)
~~probably member~~ ACNW (TBD)
Levenson

15:30 Coffee Break

16:00 Plenary Session
- **Summary and Conclusion**
- **Final Remarks**

Chairman: Sailer(RSK)
Co-Chairman: N. N. (N. N.)

17:00 **End of Meeting**

Quadrilateriales Meeting

Session 3: **PSA/PSR, Risk informed Regulation**

Chairman: Apostolakis

Co-Chairman: Hahn

Struktur und Inhalte:

1. Einführung (Apostolakis) 10'
2. Vorgehen und Entwicklungen in den Ländern 45'
   - Deutschland: Potential and Limitations of Risk informed Regulation
   - Frankreich
   - USA
   - Japan
   - Schweiz
   - Schweden
3. Diskussion 25'
4. Zusammenfassung (Hahn) 10'

Structure of German Contribution

PSA/PSR/Risk informed Regulation, German Position

Role of PSA,
PSAs performed

Periodic Safety Review

Risk considerations in the Regulatory Process

Potential and Limitations of Risk informed Regulation

IN RESPONSE, PLEASE
REFER TO: M011205B

December 20, 2001

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

William D. Travers
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), 1:30 P.M., WEDNESDAY, DECEMBER 5, 2001, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by members of the ACRS on the following topics:

1. Reactor Oversight Process
2. Regulatory Challenges for Future Plant Designs
3. ACRS Core Power Uprate Activities
4. ACRS License Renewal Activities

The NRC staff should review ACRS recommendations concerning the white/yellow and yellow/red thresholds for performance indicators (PIs), particularly as regards implementation of risk-based PIs.

The ACRS should continue to review staff efforts on risk-based PIs and improvements to the significance determination process.

The staff, with ACRS input, should provide recommendations for resolving, in a transparent manner, apparent conflicts and discrepancies between aspects of the revised reactor oversight process that are risk-informed (e.g., significance determination process) and those that are performance based (e.g., performance indicators).

The ACRS should continue its efforts to ascertain regulatory challenges for future reactor designs. The Committee should also ensure that it is prepared to review NRC staff efforts on advanced reactors in the near term, including issues related to Westinghouse's AP1000, General Atomics' gas turbine modular helium reactor, and Exelon's pebble bed modular reactor. The NRC staff should review the ACRS recommendation to develop a standard review plan to improve the effectiveness of power uprate reviews. The staff should inform the Commission of

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the results of the staff review.
(EDO) (SECY Suspense: 2/15/02)

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

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6/4/2002

SCHEDULING NOTES

Title: Meeting with ACRS

Scheduled: 2:00 p.m. - Wednesday, July 10, 2001 (PUBLIC)

Duration: Approx. 2 hours

Topics:		Presentation
1.	Overview – Power Uprates & License Renewal – Human Reliability Analysis Research Plan George Apostolakis	xx minutes*
2.	Advanced Reactor Designs Thomas Kress	xx minutes*
3.	Risk-Informing Special Treatment Reqts. of Part 50 George Apostolakis	xx minutes*
4.	Pressurized Thermal Shock Technical Basis Reevaluation Proj. Peter Ford	xx minutes*

Documents: - ACRS viewgraphs due 7/2.

* For presentation only and does not include time for Commission Q & A's.

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PROPOSED SLIDES FOR ACRS MEETING WITH THE COMMISSIONERS

JULY 10, 2002

			PAGE
1.	OVERVIEW BY THE ACRS CHAIRMAN • License Renewal • Power Uprates	– G.E. APOSTOLAKIS	1
2.	Advanced Reactor Designs	– T.S. Kress	4
3.	Mixed Oxide Fuel Fabrication Facility	– D.A. Powers	8
4.	Risk-Informing Special Treatment Requirements of 10CFR Part 50	– G.E. APOSTOLAKIS	13
5.	PTS Reevaluation Project	– F.P. Ford	15

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OVERVIEW BY THE ACRS CHAIRMAN

STATUS OF ACRS ACTIVITIES ON LICENSE RENEWAL

- COMPLETED REVIEW OF TURKEY POINT LICENSE RENEWAL APPLICATION. REPORT ISSUED APRIL 19, 2002.

- COMPLETED REVIEWS OF AT LEAST ONE PLANT FROM EACH VENDOR (COMBUSTION ENGINEERING/WESTINGHOUSE/BABCOCK & WILCOX/GENERAL ELECTRIC).

- INTERIM LETTER AT THE DRAFT SAFETY EVALUATION REPORT STAGE WILL BE PROVIDED AS NEEDED.

- UPCOMING REVIEWS
  - MCGUIRE AND CATAWBA (FIRST ICE CONDENSER CONTAINMENTS)
  - NORTH ANNA AND SURRY
  - PEACH BOTTOM
  - FORT CALHOUN (FIRST APPLICATION USING APPROVED GENERIC GUIDANCE DOCUMENTS)
  - ST. LUCIE
  - THE ACRS HAS ESTABLISHED TWO SUBCOMMITTEES TO SUPPORT THE LICENSE RENEWAL WORKLOAD

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ACRS REVIEWS OF CORE POWER UPRATES

- ACRS HAS REVIEWED AND RECOMMENDED APPROVAL OF 5 POWER UPRATES SINCE 10/01:

  - o DUANE ARNOLD ENERGY CENTER (15.3%)
  - o DRESDEN UNITS 2&3/QUAD CITIES, UNITS 1&2 (17%/17.8%)
  - o ARKANSAS NUCLEAR ONE, UNIT 2 (7.5%)
  - o CLINTON POWER STATION, UNIT 1 (20%)
  - o BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1&2 (14.3%)

- ACRS REVIEWED GE TOPICAL REPORT "CONSTANT PRESSURE POWER UPRATE" (CPPU) (4/02)

  - o CPPU METHODOLOGY APPLIED TO BWR UPRATES UP TO 20% NOMINAL POWER

  - o COMMITTEE FOUND CPPU METHODOLOGY ACCEPTABLE

- COMMITTEE REVIEW ISSUES

  - o LACK OF ADEQUATE DOCUMENTATION IN STAFF SAFETY EVALUATION REPORTS - ISSUE IS BEING ADDRESSED VIA STEADILY IMPROVED DOCUMENTS

  - o NEED FOR STAFF GUIDANCE DOCUMENT ON FUTURE UPRATE REVIEWS - STAFF ADDRESSING THIS ISSUE PURSUANT TO COMMISSION DIRECTION

  - o CORE RELOAD SAFETY ANALYSES - NRR PERFORMING AUDITS TO CONFIRM USE OF APPROVED METHODOLOGY

ADVANCED REACTORS: A STATUS REPORT

T. S. Kress
ACRS

Presented at
Meeting with Commissioners
July 10, 2002

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ACRS ACTIVITIES

- ESTABLISHED AN ADVANCED REACTORS SUBCOMMITTEE
- TWO MEMBERS PARTICIPATED IN THE NRC WORKSHOP ON HIGH TEMP GAS-COOLED REACTOR SAFETY AND RESEARCH ISSUES
- MAIN TOPIC AT OUR RETREAT
- ACRS SPONSORED A WORKSHOP ON FUTURE REACTORS
- DEVELOPING A PLAN FOR COORDINATING WITH STAFF
- FOCUS OF OUR NEXT RESEARCH REPORT

WE ARE PREPARED TO ADVISE COMMISSION AND STAFF ON ISSUES

- PRIORITY IS AP1000
- FAMILIARIZED OURSELVES WITH THE VARIOUS DESIGNS
- IDENTIFIED POTENTIAL IMPEDIMENTS
- DEVELOPED "STRAWMAN" POSITIONS ON VARIOUS ISSUES

MAJOR IMPEDIMENTS COULD BE:

- HIGH-LEVEL RISK ACCEPTANCE CRITERIA OTHER THAN CDF & LERF
- CRITERIA FOR SELECTING DESIGN BASIS ACCIDENTS
- DEFENSE IN DEPTH

D.A. POWERS

Mixed Oxide Fuel (MOX) Fabrication Facility

- to make MOX from weapons plutonium
- located on Savannah River Site
  - security & emergency response advantages
  - co-located worker issue

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Mixed Oxide Fuel (MOX) Fabrication Facility

- receive PuO_2
- crush PuO_2
- dissolve PuO_2
- purify Pu solution by PUREX process
- convert to PuO_2-UO_2 solid solutions
- fabricate fuel

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Facility Design

- modeled after a French facility
- conventional 'nested' ventilation regions
- glove box technology
- HEPA filters versus sand filters
- novel digital permissive system

Principal Hazards

- Fire
  - large kerosene inventory
  - "red oil"
  - zirconium alloy cladding for fuel
- Criticality
- Seismic issues of Charleston vicinity

Status

- mission recently expanded and made more complicated with four types of feeds
- application for construction permit delayed about 1 year
- staff has prepared a draft safety evaluation report on earlier design
  - integrated safety analysis versus PRA

Risk-Informing Special Treatment Requirements of 10 CFR Part 50

George E. Apostolakis

Presented at

Meeting with Commissioners

July 10, 2002

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Risk-Informed Categorization Scheme for Systems, Structures, & Components (SSC)

	Safety Related	Non-Safety Related
Safety Significant	RISC-1	RISC-2
Not Categorized as Safety Significant	RISC-3	RISC-4

Report Dated October 12, 1999

- Agreed that terminology of "safety-related" SSCs should be preserved.
- Pointed out the significance of importance measures and their limitations
- Recommended that staff should clarify the proper roles of:
  - ✓ importance measures
  - ✓ sensitivity and uncertainty analysis
  - ✓ baseline CDF and LERF
  - ✓ ΔCDF and ΔLERF
- Recommended that the guidance to the expert panel should include insights from implementing the preceding recommendation.

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Report Dated March 19, 2002

Reviewed proposed industry guidance in NEI 00-04/Rev. B *Option 2 Implementation Guideline*

Recommendations

- The criteria used by the Integrated Decision-making Panel (IDP) for categorization of SSCs should be made explicit and should include consideration of risk metrics that supplement CDF and LERF, such as late containment failure probability and inadvertent release of radioactive material

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Report Dated March 19, 2002
Recommendations Continued

- The use of a more complete set of risk metrics may allow the elimination of additional special treatment requirements for SSCs in class RISC-3.
  - ✓ Difficulty of treatment of RISC-3 stems from recognition that risk concerns cannot be completely addressed by CDF and LERF
  - ✓ Specific guidance to the IDP could include:
    1. Does the SSC support a system that acts as a barrier to fission product release during severe accidents?
    2. Is the SSC relied upon in EOPs or SAM guidelines?
    3. Would failure of the SSC result in inadvertent release even in the absence of severe accident conditions?
  - ✓ Materials degradation should be considered by IDP

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Report Dated March 19, 2002
Recommendations Continued

- The rigor in the treatment of uncertainties in PRA results should be made consistent with the current capabilities of PRA software and data. When simplified methods are used, comparison with more rigorous analyses should be available to demonstrate the adequacy of these methods
  - ✓ Using risk information in regulations is still viewed with skepticism by some groups
  - ✓ Rigor could contribute to building confidence
  - ✓ Substituting "sensitivity" analysis for uncertainty analysis does not contribute to confidence building
  - ✓ Assessing the impact on CDF and LERF of changing the failure rates by factors ranging from 2 to 5 (in lieu of the STP factor of 10) needs better justification

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F.P. FORD

PRESSURIZED THERMAL SHOCK (PTS) RULE RE-EVALUATION (10CFR50.61) PROJECT

- Need For Re-evaluation
  - Transients that generate risk of PTS
    - Occur less frequently than originally thought
    - Operator performance better than previously thought
    - Reactor vessel is tougher than originally thought
    - Reactor vessel contain smaller cracks than previously thought
  - As a result, original PTS criterion may be overly conservative

- Staff Approach
  - Development of integrated analytical process using models in three technical areas
    - Probabilistic Risk Assessment (PRA)
    - Thermal Hydraulics (T-H)
    - Probabilistic Fracture Mechanics (PFM)
  - Multi-Organization collaboration

- ACRS was Briefed in October 2000, February 2002, and July 2002

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PRESSURIZED THERMAL SHOCK (PTS) RULE RE-EVALUATION (10CFR50.61) PROJECT

- Project applies the integrated analytical process to four plants reflecting a range of pressurized water reactor (PWR) designs: Oconee Unit 1, Beaver Valley Unit 1, Palisades, and Calvert Cliffs Unit 1

- Current integrated analytical process versus 1980's analysis
  - Use of latest PRA/human reliability data
  - More refined binning
  - Operator action credited
  - Acts of commission considered
  - External events considered
  - Many more T-H sequences modeled
  - Significant conservative bias in toughness model removed
  - Spatial variation in fluence recognized
  - Most flaws embedded rather than on surface, flaws are smaller
  - Non-conservatisms removed in arrest and embrittlement models

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PRESSURIZED THERMAL SHOCK (PTS) RULE RE-EVALUATION (10CFR50.61) PROJECT

- Observations
  - Dominant scenarios are all primary system LOCAs
  - Realistic accounting for operator action significantly mitigates the influence of secondary system events on total failure probability
  - Main steamline breaks and steam generator tube rupture no longer dominant
  - Time of safety relief valve closure (and thus re-pressurization) has significant influence on event severity

- Ongoing work
  - Complete internals events analysis for Oconee, Palisades, BeaverValley & Calvert Cliffs
  - External Events
  - Containment Integrity
  - Source Terms

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PRESSURIZED THERMAL SHOCK (PTS) RULE RE-EVALUATION (10CFR50.61) PROJECT

- ACRS Conclusions
  - PTS Re-evaluation Project is extensive and appears technically sound.
  - The preliminary analysis results of the Oconee Unit 1 reactor pressure vessel (RPV) indicate that when the current PTS screening criterion is reached, the frequency of throughwall cracking of the vessel would be approximately two orders of magnitude below the acceptance criteria for vessel failure given in Regulatory Guide 1.154. If the such results are characteristic of the fleet of PWRs, then the current PTS screening criterion may be overly conservative.

PRESSURIZED THERMAL SHOCK (PTS) RULE RE-EVALUATION (10CFR50.61) PROJECT

- Current Technical Issues

- Factors identified as having large impacts on the failure frequency of the RPV should be scrutinized appropriately.

- For example:
  - Documentation and validation of input codes and models, i.e.
    - Relap-5 Thermal Hydraulic code,
    - Flaw distribution model
    - Re-examination of human error model
    - Varience narrowing associated with histogram sampling
    - Material composition/fluence/embrittlement model
    - Effects of reactor operating power level and fuel burnup

- All such issues are being addressed in a timely manner

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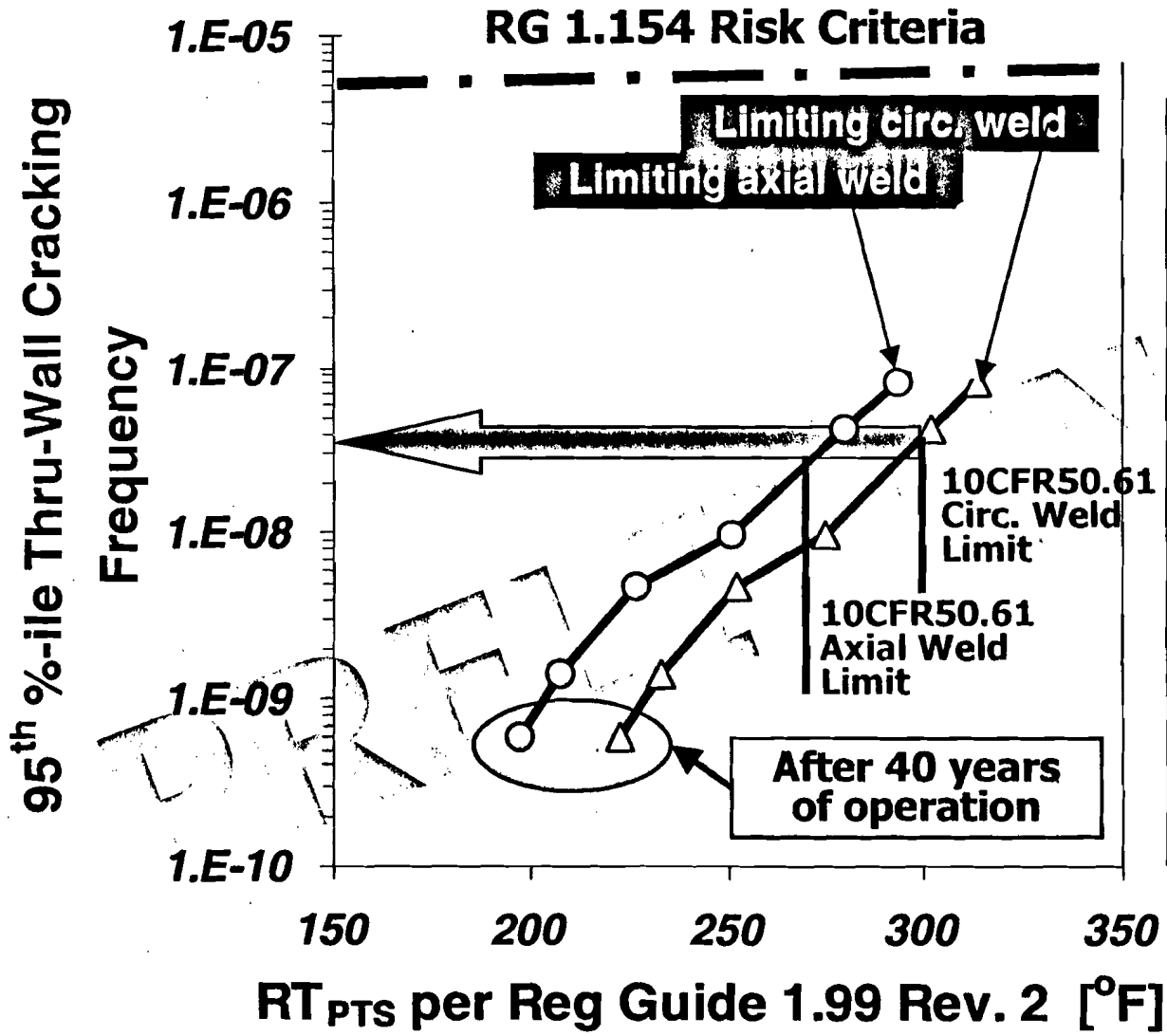
Preview of Results for Oconee 1



Relative to the RG 1.154 risk criteria, current results are

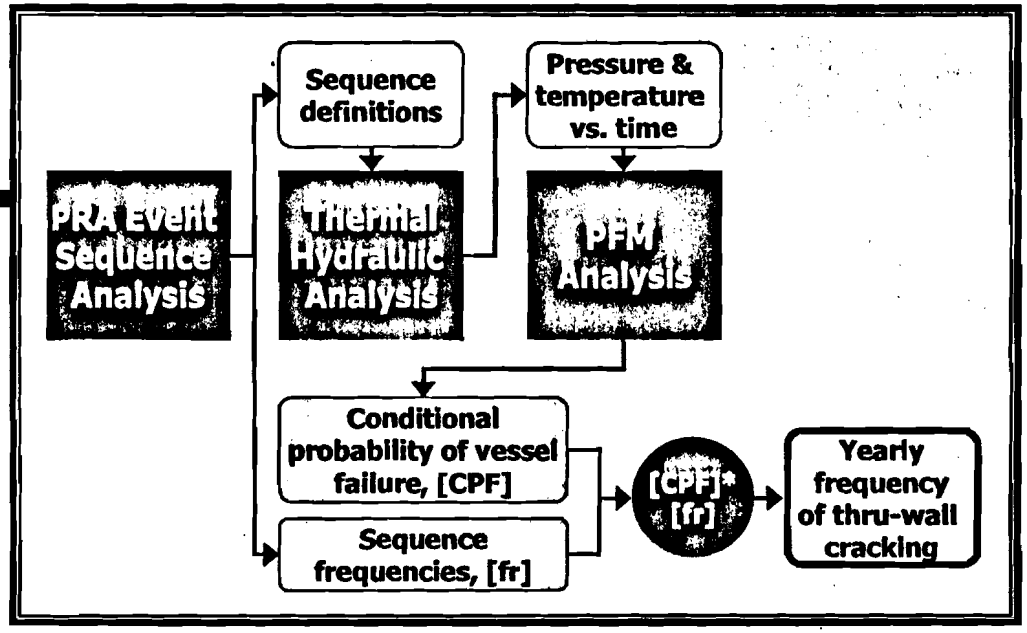
- ✓ ≈ 4 orders of magnitude lower after 40 years of operation
- ✓ ≈ 2 orders of magnitude lower at the current screening limits

DRAFT

VG 6

Summary of Changes (*vs.* 1980s Analysis)

- **PRA**
  - **Use of latest PRA/HRA data**
  - **More refined binning**
  - **Operator action credited**
  - **Acts of commission considered**
  - **External events considered**
- **TH**
  - **Many more T/H sequences modeled**



- **PFM**
  - **Significant conservative bias in toughness model removed**
  - **Spatial variation in fluence recognized**
  - **Most flaws now embedded rather than on the surface, also smaller**
  - **Non-conservatisms removed in arrest and embrittlement models removed**

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From: Maggalean Weston
To: Carol Rowe
Date: 5/23/02 9:45AM
Subject: Plant/Region Visit

Carol Ann,

ACRS members and staff will fly to Knoxville, TN on June 17, 2002, Visit Watts Bar Nuclear Power Plant on June 18, 2002, travel by motor coach on the evening of June 18, 2002, to Atlanta. We will stop for dinner en route. We will visit the Region II office on June 19, 2002 and return to Washington, DC on June 20, 2002.

I hope this helps with you advance P&P information.

Maggalean

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May 22, 2002

MEMORANDUM TO: ACRS Members

FROM: Sam Duraiswamy, Technical Assistant

SUBJECT: REVISED ACRS SUBCOMMITTEE STRUCTURE

Attached is the revised ACRS Subcommittee structure. The proposed revision to the Subcommittee structure was sent to the members and the ACRS staff. Comments received wre incorporated, as appropriate. The ACRS Chairman has approved the revised Subcommittee structure and it will become effective on June 10, 2002. A summary of the significant changes is provided below:

- **Creation of New Subcommittees**

  - A new Subcommittee "Safeguards and Security" has been created to review NRC activities associated with Safeguards and Security.

    Dr. Apostolakis will Chair this Subcommittee. Members of this Subcommittee are: Bonaca, Leitch, Powers, Rosen, Shack, and Sieber.

  - Two Plant License Renewal Subcommittees have been established as previously committed to the Commission. This was approved by the Committee in 2001.

    - Plant License Renewal Subcommittee 1. This will be Chaired by Dr. Bonaca. Members of this Subcommittee are: Apostolakis, Kress, Leitch, Shack, Sieber, and Ransom.

    - Plant License Renewal Subcommittee 2. This will be Chaired by Mr. Leitch. Members of this Subcommittee are: Bonaca, Ford, Powers, Rosen, and Wallis.

- **Chairmanship Change**

  - Safety Research Program - Dr. Ford, who has the lead responsibility for preparing the 2003 report to the Commission on the NRC Safety Research Program, will become the Chairman of this Subcommittee.

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Attachment: As stated

cc: ACRS Staff
John T. Larkins
Sher Bahadur
Howard J. Larson
Richard P. Savio

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Cognizant ACRS Staff:

PAB - Paul A. Boehnert
SD - Sam Duraiswamy
RBE - Robert B. Elliott
MME - Medhat M. El-Zeftawy
TJK - Tim J. Kobetz
JTL - John T. Larkins
MTM - Michael T. Markley
RPS - Richard P. Savio
MWW - Maggalean W. Weston

ACRS Members:

GEA - George E. Apostolakis
MVB - Mario V. Bonaca
FPF - F. Peter Ford
TSK - Thomas S. Kress
GML - Graham M. Leitch
DAP - Dana A. Powers
VHR - Victor H. Ransom
SLR - Steve L. Rosen
JDS - Jack D. Sieber
WJS - William J. Shack
GBW - Graham B. Wallis

TOPICAL SUBCOMMITTEES

FIRE PROTECTION (RBE) . . . . . **ROSEN**, Apostolakis, Kress, Powers, Sieber, Wallis

- Review adequacy of fire protection requirements for operating plants as well as for future plants.
- Review fire risk assessment methodology in coordination with the Reliability and Probabilistic Risk Assessment Subcommittee.
- Review the rulemaking to endorse National Fire Protection Association, NFPA-805 performance-based fire protection standard.
- Review the Boiling Water Reactor Owners Group (BWROG)/Nuclear Energy Institute (NEI) post-fire safe shutdown circuit analysis and associated NRC staff evaluation.
- Review insights from the Individual Plant Examination of External Events (IPEEE) Program associated with fire protection.
- Review the results and findings of the pilot baseline reactor oversight program in the area of fire protection.
- Review the Significance Determination Process for findings of inspections dealing with fire protection
- Review fire risk assessment research program, including the models for circuit failure mode analysis and IEEE-383 rated cable fire frequency analysis.
- Review fire protection aspects of the proposed mixed oxide (MOX) Fuel Fabrication Facility, in coordination with the Reactor Fuels Subcommittee.

FUTURE PLANT DESIGNS (MME) . . . . . **KRESS**, Apostolakis, Bonaca, Leitch, Powers, Ransom, Rosen, Shack, Wallis

- Review Westinghouse application for certification of the AP1000 design and the associated NRC staff's SER.
- Review regulatory challenges associated with advanced reactor designs.
- Review technical and policy issues for the advanced reactor designs.
- Review draft final revisions to 10 CFR Part 52.
- Review advanced reactors research plan.
- Review NRC staff activities associated with early site permit and pre-application review of

advanced reactor designs.

- Consider application of Severe Accident Policy to future plants.

HUMAN FACTORS (MME) ..... **POWERS**, Apostolakis, Bonaca, Leitch, Rosen

- Consider man-machine interaction, including design and arrangement of the control room and operator response under stress.
- Review control room habitability issues.
- Review NRC Human Performance Program, including its implementation.
- Review Human Reliability Research Plan.
- Review regulatory requirements for dealing with human factors issues.
- Review human/organizational factors issues associated with significant operating events in coordination with the Plant Operations Subcommittee.
- Review qualifications and training of personnel at nuclear facilities in coordination with the Plant Operations Subcommittee.
- Review proposed revisions to Standard Review Plan, Chapter 18, Human Factors Engineering.
- Assess the importance of Safety Culture in the error forcing context and assess the need to include modeling of Safety Culture in PRAs.

MATERIALS AND METALLURGY (MWW) ..... **FORD**, Shack (Vice Chair.), Bonaca, Powers

- Review NRC program to evaluate plant aging of metal components (e.g., pressure vessel embrittlement, steam generator tube degradation, and thermal aging of cast stainless steel piping and components).
- Review the adequacy of nondestructive examination techniques in detecting and sizing flaws in metal components, piping systems, and steam generator tubes.
- Keep abreast of new developments associated with ASME Section III for pipe design criteria and Section XI, inservice inspection/pressure boundary integrity of operating nuclear plants.
- Review 1994 Addenda for Class 1, 2, and 3 piping systems to the ASME Code, Section III, and the resolution of the differences between the staff and the ASME.
- Review regulatory approach associated with the steam generator tube integrity, and the staff's safety evaluation on industry proposed technical specifications for addressing steam generator

tube integrity.

- Review causes, consequences, and preventive measures associated with the control rod drive mechanism penetration nozzle cracking and degradation of reactor pressure vessel head.

- Review NRC staff's Steam Generator Action Plan for resolving steam generator issues as well as the issues raised by the ACRS in NUREG-1740 associated with the Differing Professional Opinion (DPO) on steam generator tube integrity
(Dr. Powers lead member on DPO issues)

- Review proposed revisions to 10 CFR 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock (PTS), in coordination with the Subcommittees on Thermal-Hydraulic Phenomena and on Reliability and Probabilistic Risk Assessment.

- Review proposed amendment to 10 CFR 50.55a regarding volumetric examination of the small-bore piping of the high pressure safety injection (HPSI) system.

- Review proposed revisions to Regulatory Guide 1.99 (irradiation embrittlement) and Regulatory Guide 1.154 (PTS).

- Review proposed regulatory guidance on reactor coolant chemistry effects on flaw growth estimates.

- Consider plant water chemistry and quality control as related to corrosion control.

- Consider integrity of nuclear plant piping under earthquake-induced conditions.

- Consider integrity of degraded piping systems and components caused by erosion/corrosion phenomenon.

- Consider issues associated with license renewal for Independent Spent Fuel Storage Installations and the structural integrity of fuel shipping casks and adequacy of spent fuel encapsulators, as requested by the Advisory Committee on Nuclear Waste (ACNW).

NATURAL PHENOMENA (SD) . . . . . **POWERS**, Apostolakis, Kress, Rosen, Shack

- Review the bases for the assumptions used in Probabilistic Seismic Hazard Analysis (PSHA) and its use in nuclear plant regulation.

- Review the seismic requirements for site permit applications for advanced reactors.

- Review the vulnerabilities of the AP1000 passive plant design to seismic events in coordination with the Future Plant Designs Subcommittee.

- Review proposed final revisions to Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response."

- Review NRC research and information needs in the seismic area.

- Review NRC line organization capabilities for quantitative assessment of seismic and natural phenomena risk.

- Review the cooperative research with Japan and others, especially the experimental investigations of structural responses to earthquakes.

- Review the staff activities to address the uncertainties in the treatment of soil liquefaction, which were identified by the IPEEE Program.

NAVAL REACTORS (PAB) . . . . . **SIEBER**, Ford, Kress, Leitch, Rosen, Wallis

- Consider safety issues associated with the U.S. Naval reactor designs.

- Review proposed Naval reactor design (e.g., Virginia class attack submarine).

- Visit U.S. Naval reactor facilities, as needed, to gather information for use in reviewing proposed Naval reactor designs, training programs, etc.

PLANNING AND PROCEDURES (SD) . . . . . **APOSTOLAKIS**, Bonaca, Kress

(NOTE: This Subcommittee includes the ACRS Chairman, Vice Chairman, and one member-at-large elected each year at the same time as the Chairman and Vice Chairman, and is chaired by the ACRS Chairman.)

- Prioritize items proposed for each ACRS meeting for consideration by the Full Committee.

- Organize ACRS retreats to discuss and exchange views on safety philosophy and policy.

- Develop proposals for significant changes in ACRS policies, practices, and bylaws for consideration by the Full Committee. Consider especially changes mandated by revisions to the Federal Advisory Committee Act.

- Implement policies of ACRS in planning Full Committee activities, articulating priorities, allocating and controlling resources, and scheduling and monitoring activities of the Subcommittees.

- Perform annual review of the Subcommittee structure, tasks, and workload of members and recommend changes, as needed, for Full Committee consideration.

- Assume lead responsibility for the coordination of ACRS meeting with international

organizations.

- Monitor the adequacy of implementation of the memorandum of understanding (MOU) between the ACRS and the EDO.

- Follow-up on the resolution and implementation of the commitments made at the annual ACRS retreats.

- Review self assessment of ACRS performance and provide to the Full Committee for consideration.

PLANT LICENSE RENEWAL -1 (RBE) . . . . . **BONACA**, Leitch (Vice Chair.), Apostolakis, Kress, Shack, Sieber, Ransom

- Review regulatory requirements related to plant license renewal.

- Review license renewal applications and associated NRC staff's Safety Evaluation Reports for:

  - Catawba/McGuire
  - Fort Calhoun
  - Robinson 2
  - Summer

- Consider the NRC program to evaluate plant aging except for those aspects being considered by the Materials and Metallurgy Subcommittee (e.g., pressure vessel embrittlement, stream generator tube degradation, and thermal aging of cast stainless steel piping and components).

- Review selected industry topical reports associated with license renewal of above plants.

- Review the resolution of differences between the staff and industry on generic license renewal issues.

- Review updates made to the Standard Review Plan, Generic Aging Lessons Learned (GALL) Report, and Regulatory Guidance associated with license renewal to reflect lessons learned from the review of license renewal applications.

PLANT LICENSE RENEWAL -2 (TJK) . . . . . **LEITCH**, Bonaca (Vice Chair.), Ford, Powers, Rosen, Wallis

- Review license renewal applications and the associated NRC staff's Safety Evaluation Reports for:

  - North Anna and Surry
  - Peach Bottom
  - St. Lucie
  - Ginna

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Dresden and Quad Cities

- Review selected industry topical reports and the staff's safety evaluations associated with the license renewal of the above plants.

PLANT OPERATIONS (MWW) ..... **SIEBER**, Leitch (Vice Chair.), Bonaca, Rosen

- Assume lead responsibility for activities related to plant operations and provide oversight and coordination of the review of plant operating experience by ensuring that events and reports needing review are covered by other ACRS Subcommittees, or assume responsibility for the review as appropriate.

- Consider regulatory guidance and issues associated with PWR containment sump strainer blockage.

- Review Augmented Inspection Team (AIT) and Special Inspection Team (SIT) reports associated with operating events, as appropriate, and recommend specific events for consideration by the Full Committee.

- Review the implementation of the NRC Performance Indicator Program and industry programs to provide voluntary performance information.

- Consider the need for new technical training in areas of increasing importance to the NRC mission related to nuclear power plants and the nuclear fuel cycle facilities.

- Review restart of plants that have been shut down for an extended period (more than one year) and make recommendation to the Full Committee.

- Consider the Licensee Event Report (LER) and incident investigation processes, operational QA/QC, and operator performance.

- Review, in coordination with the Reliability and PRA Subcommittee, initiatives related to risk-informed Technical Specifications.

- Review issues associated with the operation and maintenance of fuel cycle facilities and the adequacy of associated regulatory requirements.

- Review information on accidents and unusual events at foreign nuclear plants and consider potential implications at U.S. plants.

- Monitor the activities of the NRC Regional Offices by holding annual meetings at different Regional Offices. Visit one plant per year in different Regions and obtain information on industry issues.

- Review, in coordination with the Reliability and PRA Subcommittee, the use of performance

indicators (PIs) in the Revised Reactor Oversight Process to ensure that the PIs provide meaningful insight into aspects of plant operation that are important to safety.

- Review enhancement to the Significance Determination Process in coordination with the Reliability and PRA Subcommittee.

- Review NRC staff and industry activities related to restructuring and economic deregulation of electric utility industry, and associated safety implications.

PLANT SYSTEMS (TJK) . . . . . **LEITCH,** Bonaca, Rosen, Sieber

- Consider generic safety implications of the performance of systems not assigned to other Subcommittees (e.g., air powered systems, cleanup systems, and chilled water systems).

- Consider gaseous, liquid, and solid radioactive waste handling systems and related activities at nuclear power plant sites.

- Consider biological effects of ionizing radiation, standards for protection against radiation (10 CFR Part 20), and associated regulatory guidance.

- Review mechanical component operability assurance and reliability, including the functioning of valves under accident loading conditions.

- Review systems interaction issues and criteria, including consideration of functional interactions for existing and future plants.

- Consider the effects of harsh and adverse environment on the plant safety systems.

- Follow-up on the implementation of the provisions of the Regulatory Guides, Standard Review Plan Sections, Branch Technical Positions related to digital instrumentation and control systems.

- Consider verification and validation of computer software used in control and protection systems.

- Review reliability of AC/DC power systems in nuclear facilities, including the potential for disruption of offsite power sources and backup power systems.

- Review lightning protection provisions for operating and future plants.

- Provide oversight and coordination of the prioritization and resolution of generic safety issues, handling those items it is competent to deal with and assigning others to appropriate Subcommittees for review.

- Review proposed resolution of GSI-168, "Environmental Qualification of I&C Equipment."

- Review NRC Staff activities associated with cable aging.

- Consider safety issues related to design and performance of instrumentation and control, electrical, electronic systems, as well as safety actuation systems, leak detection systems, and vibration and loose parts monitoring systems, as appropriate.

REACTOR FUELS (MME/MWW) ..... **POWERS**, Ford, Kress, Shack

- Review technical issues and construction authorization application by Duke Cogema Stone & Webster associated with the proposed Mixed Oxide (MOX) Fuel Fabrication Facility.

- Assume lead responsibility for activities related to reactor fuel and review fuel-related issues, as necessary.

- Review NRC staff activities related to revising fuel design acceptance criteria for high burnup fuel and the use of Phenomena Identification and Ranking Table (PIRT) process for high burnup fuel.

- Review programs of industry Fuel Vendors and Owners Groups to address concerns associated with use of high burnup fuel.

- Consider fuel performance during normal and abnormal conditions, including fuel failure propagation.

- Review NRC and Industry fuel performance codes.

- Review DOE Tritium Production Program using commercial nuclear power plants.

- Review the licensing of uranium enrichment facilities.

- Review safety issues associated with the use of MOX Fuel.

- Review changes to regulatory requirements associated with transportation of radioactive materials.

- Evaluate the design of spent fuel storage pools, including pool storage capability and provisions to preclude criticality and to cool the fuel under normal and abnormal conditions and following external events such as earthquakes.

- Review spent fuel pool accident risk for decommissioning and operating plants.

- Review safety issues associated with new and modified fuel designs.

REGULATORY POLICIES AND PRACTICES (RBE) . . . . . **SHACK**, Bonaca, Kress, Leitch, Powers, Ransom

- Examine the coherence and specific aspects of the NRC regulatory process, as appropriate, and consider changes in emphasis needed in safety-related NRC rules and regulatory practices. Consider technical road map for risk-informing 10 CFR Part 50, including Appendices A and B.

- Identify important safety issues needing increased (or less) attention and/or resolution in the NRC regulatory process.

- Consider possible changes to General Design Criteria (GDC) and other criteria (e.g., single failure criterion) as needed, based on operating experience, advances in related technology, and research results.

- Review proposed NRC safety-related rules not assigned to specific ACRS Subcommittees.

- Review NRC staff and industry activities associated with control room habitability.

- Review the NRC staff's reevaluation of the effectiveness of those existing regulations which were not assigned to other Subcommittees.

- Review proposed resolution of GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."

- Consider activities associated with the NRC oversight of DOE facilities.

- Consider methods for estimating offsite consequences given PRA results.

- Consider the use of defense-in-depth concept in the regulatory process.

- Consider the nature and characteristics of core-coolant interactions (e.g., steam explosions) and core-containment interactions (e.g., reactor with concrete and heating of structural elements).

- Consider severe accident chemistry (including effects of radiation on aerosols and post-accident hydrogen burning), codes for modeling such chemistry, and relation to the accident source term.

- Consider hydrogen control measures at nuclear power plants and associated regulatory requirements, including the need for requirements imposed following the TMI accident, such as the post accident sampling system.

RELIABILITY AND PROBABILISTIC RISK ASSESSMENT (MTM/AWC) . . . . . **APOSTOLAKIS**, Bonaca, Ford, Kress, Powers, Rosen, Shack

- Review the risk-informed and performance-based regulatory approaches, including risk-informed revisions to 10 CFR Part 50 and to Appendices A and B.
- Review proposed 10 CFR 50.69 and NEI 00-04, "Option 2 Implementation Guideline" and proposed revisions to Regulatory Guide 1.174.
- Follow-up on the implementation of the Standard Review Plan Sections and Regulatory Guides associated with risk-informed regulation.
- Consider application of risk insights in the regulatory process.
- Consider methods for incorporating human reliability in PRAs.
- Consider reliability studies of safety systems and the use of reliability and availability information in the regulatory process.
- Consider the Accident Sequence Precursor Program and the SPAR models as well as the application of the results of these in the regulatory process.
- Review risk-informed regulation implementation plan.
- Consider the use of the insights from IPEs and IPEEs.
- Consider the consistent and extended use of PRAs in the regulatory process and the associated NRC programs.
- Review proposed industry standards for PRA quality and low-power and shutdown operations.
- Review proposed industry program for certification of PRA quality.
- Gather information for developing recommendations to the Commission on the significance of low-power and shutdown operations risk and review the adequacy of the staff's analytical tools for independently assessing the risk significance of plant configurations during low-power and shutdown operations, especially during plant transitions.
- Review risk-based performance indicators and the Significance Determination Process in coordination with the Plant Operations Subcommittee.
- Review the initiatives related to risk-informed Technical Specifications in coordination with the Plant Operations Subcommittee.
- Consider methods for assessing uncertainties and other issues related to risk-informed decisionmaking.
- Review NRC/industry seismic design margins evaluation program.
- Review the impact of common-mode failures on the performance of plant safety systems.

SAFETY RESEARCH PROGRAM (RPS) ..... **FORD**, Apostolakis, Bonaca, Kress, Powers, Ransom, Rosen, Shack, Wallis

- Perform an annual review of the overall NRC Safety Research Program and prepare a report to the Commission on the quality, scope, and balance of the ongoing and proposed research activities, as well as on research priorities.
- Identify new areas of research that are essential for regulatory decisionmaking and research projects that are no longer cost effective and can be elimianted.
- Review the adequacy of the user office needs for research.
- Evaluate whether NRC research places proper emphasis on resolving important regulatory issues.
- Consider what research should be done by the NRC and the industry and cooperative research arrangements between NRC and other resent organizations.

SAFEGUARDS AND SECURITY (RPS/RKM) ..... **APOSTOLAKIS**, Bonaca, Leitch, Powers, Rosen, Shack, Sieber

- Keep Informed of the NRC activities in the area of safeguards and security, including the reevaluation of design basis threat and proposed activities to enhance the safeguards and security programs, and be prepared to advise the Commission and the staff as requested.
- Review RES-sponsored work related to vulnerability evaluations of nuclear facilities.
- Review technical and risk-management issues associated with nuclear facility vulnerabilities, including those related spent fuel pools and decommissioning plants.
- Review technical issues associated with the reevaluation of emergency planning.
- Review safeguards requirements for uranium enrichment facilities and MOX Fuel Fabrication Facility.
- Review proposed design features to mitigate the effects of sabotage and plant arrangements to enhance security.
- Review NRC staff's rulemaking activities associated with safeguards and security
- Consider material control and accounting of special nuclear material.

THERMAL-HYDRAULIC PHENOMENA (PAB) ..... **WALLIS**, Ransom (Vice Chair), Ford, Kress, Powers, Sieber

- Review requests for core power level increases greater than 5%.
- Consider evaluation of emerging safety issues associated with thermal-hydraulic phenomena.
- Review thermal hydraulic issues associated with the development of revised PTS screening criterion.
- Review issues related to water hammer and corrective measures.
- Review RES thermal-hydraulic research program, including the program to upgrade the NRC thermal-hydraulic codes and RES experimental program related to subcooled boiling phenomena.
- Review issues associated with core reload analysis for plants seeking power uprates.
- Review proposed Regulatory Guide and Standard Review Plan associated with the accident and transient codes.
- Review best estimate thermal-hydraulic codes submitted by licensees.
- Review issues associated with the use of industry-developed thermal-hydraulic codes.
- Review potential synergistic effects and margin reduction associated with concurrent regulatory activities (e.g., power uprates, license renewal, and risk-informed regulation).
- Review the thermal-hydraulic aspects of the AP1000 passive plant design and advanced reactor designs, in coordination with the Future Plant Designs Subcommittee.

ACRS/ACNW JOINT SUBCOMMITTEE (MTM/AWC) . . . . . **KRESS**, Powers, Garrick, Levenson

- Review proposed framework for risk-informed regulation in NMSS.
- Review the application of the defense-in-depth concept in a risk-informed regulatory system.
- Review Standard Review Plan Chapter 3 for Integrated Safety Assessment (ISA).
- Review PRA for dry cask storage.
- Review risk-informed case studies being developed by NMSS.
- Other tasks to be determined jointly by the ACRS and ACNW.

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From: Zünd Marianne <Marianne.Zuend@hsk.psi.ch>
To: '"Larkins John"' <JTL@nrc.gov>
Date: 4/25/02 7:14AM
Subject: Invitation Workshop in Switzerland 15-18 October 2002

Dear Mr. Larkins

It is a great pleasure for us to invite you or a representative(s) of your organisation to the

Joint HSK-IAEA-NEA Workshop on Regulatory Decision Making Processes
Switzerland, 15 - 18 October 2002

HSK, the Swiss Federal Nuclear Safety Inspectorate will act as the host of the event, with the support of the International Atomic Energy Agency IAEA and the OECD Nuclear Energy Agency NEA. The Workshop will be held at the Grandhotel Giessbach in the beautiful Bernese Alps.

Please find enclosed the announcement, and the preliminary agenda of the Workshop as well as the registration form (Word-Doc).

May we please ask for your reply to this invitation by sending the registration form to the HSK Workshop Secretariat by 31 May 2002 at the latest.

Regarding the practical arrangements, please see the enclosed document. Further information will be provided to you together with the confirmation of your registration.

If you have any queries about the workshop, please do not hesitate to contact Mr. Sabyasachi Chakraborty, who is the chair of the organising committee (chakraborty@hsk.psi.ch, Tel. +41-56-310 39 36).

We look forward to welcoming you to a fruitful Workshop.

Sabyasachi Chakraborty
Marianne Zuend
****

Marianne Zuend
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HSK in cooperation with IAEA and NEA

Workshop on Nuclear Regulatory Decision Making Processes

Switzerland, 15 – 18 October 2002

Registration Form

Please complete each section of this Registration form and send by 31st May 2002 to:

Marianne Zuend
HSK
CH-5232 Villigen-HSK
E-mail: zuend@hsk.psi.ch
Phone: +41-56-310 39 76
Fax: +41-56-310 49 76

Title: .....
First Name: .....
Last Name: .....
Badge Name: .....
Organisation/Institution: .....
Address: .....
City: .....
Postal Code: .....
Country: .....
E-mail address: .....
Telephone: ..... *(incl. country code number)*
Fax Number: ..... *(incl. country code number)*

Any special dietary requirements? (e.g. vegetarian): .....

Accomodation: *(please underline the appropriate)*
Single room / Double room (please indicate name of 2nd person)

Your remarks / special requests: .....

The costs per person will be around 700 Swiss Francs or approx. 475 Euro to be paid individually and directly to the hotel. This amount includes accomodation in the Grandhotel Giessbach for 3 nights (Tuesday through Friday) as well as all meals (breakfast, morning & afternoon coffee, lunches and dinners), and conference materials. Travel costs and personal expenses (e.g. mini-bar consumptions) are not included.

Confirmation of your registration and accommodation booking as well as travel information in Switzerland will be sent to you via e-mail. Participants should await confirmation from the registration office before committing to other travel arrangements.

Federal Office of Energy (Switzerland)
Bundesamt für Energie (Schweiz)
Office fédéral de l'énergie (Suisse)
Ufficio federale dell' energia (Svizzera)

Swiss Federal Nuclear Safety Inspectorate
Hauptabteilung für die Sicherheit der Kernanlagen
Division principale de la Sécurité des Installations Nucléaires
Divisione principale della Sicurezza degli Impianti Nucleari

Preliminary Version

HSK in cooperation with IAEA and NEA

Workshop on Regulatory Decision Making Processes

Switzerland, 15 – 18 October 2002

Restructuring of the nuclear industry due to market deregulation, the increasing competition that forces nuclear power plant operators to enhance efficiency, the long-term extension of operating licenses and plant lifetimes, urgent decommissioning safety issues, and the debate on phasing out nuclear power in several countries are challenging nuclear regulation today. Moreover, the constantly increasing scientfic and technical knowledge is leading to changes in regulatory requirements, and is at the same time providing improved and new methods and technologies for nuclear regulation.

In many countries, regulatory decision making processes are continuously being reviewed in response to the developments mentioned above. Following the latest US NRC developments some countries are embarking on implementation of "risk informed regulations". Optimizing regulatory body efficiency and effectiveness seems to be one of the objectives driving most of the regulatory bodies that have decided to apply risk informed regulation. Caution and fear of misinterpretation or overwhelming contribution of the PSA results in decision making are the main reasons why other regulatory bodies hesitate to consider or even reject the "risk informed regulation" concept.

IAEA and NEA have recognized the importance of revaluating nuclear regulatory activities in several working groups and publications. However, as of today, there is no comprehensive approach in terms of guidelines/best practices on how the regulatory decision making processes need to be refined and how they can be systemized.

This workshop aims at sharing information on approaches and methods for regulatory decision making (nuclear, aviation, space, food safety). Through presentations and panel discussions, attendees will illuminate issues, review the capacity of the current regulatory decision making process, discuss the implications of new technical and scientific knowledge for regulatory practices, and debate differences between countries. Thus, the workshop offers a good opportunity to assess in an open exchange the pending regulatory decision making issues, which have a direct impact on the safety of nuclear facilities and on the competitiveness of nuclear power plant operators.

The workshop results will be retained in the form of a written report which reflect the opinions expressed by the participants in the form of recommendations for the establishment of a regulatory decision making guidance document.

Telephone: +41 56/310 38 11
Fax: +41 56/310 39 07

Postadresse
adresse postale
indirizzio postale
postal address

Hauptabteilung für die Sicherheit der Kernanlagen
CH-5232 Villigen-HSK

Some of the issues that will be addressed in the workshop presentations, discussions and working groups will be:

- Regulatory decision making in circumstances of uncertainty – the precautionary approach
- Experience gained from new approaches from non-nuclear sectors which could be useful to the nuclear regulatory decision making processes
- Discussion of real decision making examples: basis and processes in practice
- Setting of priorities in regulatory decision-making
- Consideration of costs in regulatory decision-making
- Impact of risk informed decision making on regulatory inspection and assessment programmes
- Safety philosophy for adequate safety and protection, and risk-informed regulation, safety performance indicators
- How to create regulatory decision-making processes that are transparent, open, credible and accountable to the public
- Communication and dialogue with the public in regulatory decision making
- Engagement of stakeholders in decision-making processes
- Feedback and monitoring mechanism for regulatory decison making
- Clarification of the role of external experts in regulatory decision making processes

Workshop Goal and Objectives

This 3-day invitational workshop is designed to advance the discussion on nuclear regulatory decision making processes.

The goal of the workshop is to develop recommendations to the attention of the IAEA and the NEA to foster the identification and elaboration of effective guidance for nuclear regulatory decision making processes. The recommendations will address the legal, institutional, organizational, and financial factors that influence the regulatory decision making processes in the different participating countries.

Specific Objectives

- Refine the understanding of regulatory decision making processes by reviewing the current approaches, methods and tools
- Discuss concrete examples of regulatory decision making: what is the basis and which are the processes that lead to a decision?
- Collect information on national experience with "risk informed decisions" taken by regulatory bodies
- Identify criteria for activities and approaches that will have a positive impact on regulatory decision making processes
- Publish a post-workshop report that identifies recommendations for the improvement of nuclear regularory decision making processes.

Organizing committee

The organizing committee for the workshop will be chaired by S. Chakraborty, HSK, and will comprise representatives of the IAEA, the NRC and the NEA. Members of the organizing committee will help to complete the program, to create and disseminate the invitation and will also assist in drafting and reviewing of the workshop report.

Location, date, and organization

The workshop will be held in Switzerland in the Grandhotel Giessbach in the beautiful Bernese Oberland (http://www.giessbach.com).

Date: **Tuesday, 15 October, 2002 - Friday, 18 October, 2002**

The workshop will begin with the arrival of the participants and dinner on Tuesday evening 15 October, and will run from Wednesday morning through Friday noon.

Approximately 40 people will be invited to the workshop. Delegates from Eastern European countries are **particularly** welcomed.

The costs per person will be around 700 Swiss Francs or approx. 475 Euro (3 nights in a single room including all meals) to be paid individually and directly to the hotel at check-out on Friday 18th October 2002.

Workshop agenda

The organizing committee will devise the detailed and final workshop agenda on the basis of the preliminary programme as outlined on the next pages (as of 25th April 2002). Please note that the preliminary programme is subject to change.

Registration

Please register by filling in and returning the attached **registration form** by e-mail or fax to HSK:

Ms. Marianne Zuend
zuend@hsk.psi.ch
Tel. +41-56-310 39 76
Fax +41-56-310 49 76

Registration is required by **31st May 2002** at the latest.

HSK in cooperation with IAEA and NEA

Workshop on Nuclear Regulatory Decision Making Processes

Switzerland, 15 – 18 October 2002

Preliminary Programme

(version of 25th April 2002)

Tuesday, 15th October 2002

Time	Activity	Remarks
Afternoon	Individual arrival of participants by train or car Transfer by hotel bus from railway station Brienz to the Grandhotel Giessbach http://www.giessbach.com	
19:00	Dinner at the Grandhotel Giessbach Restaurant	

Wednesday, 16th October 2002

Time	Activity	Remarks
08:30 – 09:15	***Welcome address by HSK, IAEA and NEA*** Layout of workshop goals	presenters to be determined
09:20 – 10:20	***IAEA activities in the field of risk-informed decision making and risk-informed regulation*** Ms. Vesselina Ranguelova, IAEA	Working title
10:20 – 10:50	Coffee Break	
10:55 – 11:55	***U.S. NRC activities in the field of regulatory decision making*** Ashok C. Thadani, Director of Office of Nuclear Regulatory Research, Nuclear Regulatory Commission	
12:00	Lunch at the Grandhotel Giessbach Restaurant	
14:00 – 15:00	***U.S. ACRS perspectives on regulatory decision making*** ??, Advisory Committee on Reactor Safeguards	Presenter and title to be determined
15:05 – 15:40	***Practical Example: Decision making in regulatory licensing processes at HAEA (license renewal, major safety upgrading)*** Lajos Vöröss, Hungarian Atomic Energy Authority	Working title
15:40 – 16:10	Coffee Break	
16:15 – 16:50	***Practical Example: Communication and dialogue with the public in regulatory decision making in SNRCU*** Sergiy Peleshenko, State Nuclear Regulatory Committee of Ukraine	Working title
16:55 – 17:30	***Practical Example: Process of regulatory decision making at SNSA*** ??, Slovenian Nuclear Safety Administration	Presenter and title to be determined
17:35 – 18:30	***Panel discussion:*** Issues and opportunities for the evaluation and improvement of nuclear regulatory decision making processes Chairman: U. Schmocker (Director HSK). Panel participants: V. Ranguelova (IAEA), B. Kaufer (NEA), V. Gryschenko (Chairman SNRCU), L. Vöröss (Director HAEA), Representatives of US NRC and US ACRS	
19:00	Dinner at the Grandhotel Giessbach Restaurant	

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Thursday, 17th October 2002

Time	Activity	Remarks
08:30 – 09:00	***Long-term radioactive waste management: challenges and approaches to regulatory decision making*** Thomas Flüeler, Swiss Federal Nuclear Safety Commission KSA	Working title
09:05 – 09:45	***Decision making processes in ESA's flight programmes*** Christian Preyssl, European Space Agency	Working title
09:45 – 10:15	Coffee Break	
10:20 – 11:00	***Application of the system safety process within the FAA's aviation safety inspector workforce*** Geoffrey R. McIntyre, US Federal Aviation Administration FAA	Working title
11:05 – 11:45	***Approaches of the Food Standards Agency for involving consumers and other stakeholders in the regulatory processes on food safety*** Mr. Richard Burt, UK Food Standards Agency	Working title
11:50 – 12:30	***OECD-NEA activities in the field of regulatory decision making*** Mr. Barry Kaufer, OECD-NEA	Working title
12:40	Lunch at the Grandhotel Giessbach Restaurant	
14:00 – 16:00	***Take up of work in 2 working groups*** : identify issues, characteristics and opportunities, and formulate recommendations for the attention of the IAEA and the NEA to foster the identification and elaboration of effective guidance for regulatory decision making processes. Chairman of working group 1: Geoffrey R. McIntyre Chairwoman of working group 2: Vesselina Ranguelova	
16:00 – 16:30	Coffee Break	
16:35 – 17:30	***Continuation of work in 2 working groups***	
18:15	***Cocktail Reception*** offered by HSK	
19:15	Dinner with ***Invited presentation*** at the Grandhotel Giessbach Restaurant	Invited presenter to be determined

Friday, 18th October 2002

Time	Activity	Remarks
08:30 – 09:10	***Presentation of results of working group 1***	
09:15 – 09:55	***Presentation of results of working group 2***	
10:00 – 10:30	Coffee Break	
10:35 – 11:30	***Closing plenary discussion***: wrap up of presentations, concrete examples and working group results, next steps. Chairman: S. Chakraborty, HSK	
11:30	***Hotel Check-Out*** of participants	
12:15	Lunch at the Grandhotel Giessbach Restaurant	
Afternoon	Individual departure of participants by train or car Transfer by hotel bus from the Grandhotel Giessbach to the railway station Brienz.	

ACRS MEMBER'S COMPENSATION REPORT

TO: **TANYA X. G. WINFREY**

FROM: _____ **SIGNATURE:** _____

[Note: For Activity Codes, see reverse side]

DATE	ACTIVITY CODE [e.g., S00029]	NATURE OF WORK [e.g., PREPARATION, MEETING NAME, TRAVEL, ETC.]	TOTAL
5/1	S00006	Meeting preparation	8
5/2	S00006	Meeting attendance	8
5/3	S00006 S00006	Meeting attendance Travel	4 3
		OR	
5/1	S00006 S00017 S00016	494th - Meeting preparation 494th - Meeting preparation 494th - Meeting preparation	2 2 2
5/2	S00006 S00017 S00016 S00026	494th - Meeting attendance 494th - Meeting attendance 494th - Meeting attendance Travel	3 3 2 3

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ACTIVITY CODE	TITLE
S00000	LICENSE RENEWAL - MA6509 - GENERIC (NON-BILLABLE)
S00005	NAVAL REACTORS - NAVY NUCLEAR PROPULSION (GENERAL) - MA6509
S00006	AP-1000 - MA8871
S00007	SRELAPS5 LARGE BREAK LOCA - CODE REVIEW
S00008	TRACG A00 - CODE REVIEW
S00009	TRACG LARGE BREAK LOCA - CODE REVIEW
S00010	WCOBRA/TRAC SMALL BREAK LOCA - CODE REVIEW
S00018	REACTOR OVERSIGHT PROGRAM - REACTOR OPERATIONS
S00019	ANNUAL RESEARCH PROGRAM
S00020	GENERIC SAFETY ISSUES
S00021	RULEMAKING
S00022	RISK-INFORMED REGULATIONS
S00023	POWER UPRATES (OTHER REGULATORY ACTIVITIES)
S00024	MIXED OXIDE FUEL FABRICATION FACILITY (OTHER REGULATORY ACTIVITIES)
S00025	CONTROL ROOM HABITABILITY (OTHER REGULATORY ACTIVITIES)
S00026	ACRS MEETING SUPPORT
S00027	SRP ON THERMAL-HYDRAULIC CODES (OTHER REGULATORY ACTIVITIES)
S00037	NAVAL REACTORS - VA CLASS SUBMARINE
S00038	CRDM CRACKING BULLETIN - URGENT GENERIC COMM. ON CRDM NOZZLES - MB2060
S00039	ADVANCED REACTORS - GENERIC
S00041	SOUTH TEXAS PROJECT - MA6057 & MA6058
S00043	SKIN DOSE LIMIT
S00044	ACRS ACTION PLAN
S00050	OTHER REGULATORY ACTIVITIES
S00052	HIGH-BURNUP FUEL
S00053	SURRY 1 - LICENSE RENEWAL - MB1988 (50-280)
S00054	SURRY 2 - LICENSE RENEWAL - MB1989 (50-281)
S00055	NORTH ANNA 1 - LICENSE RENEWAL - MB1990 (50-338)
S00056	NORTH ANNA 2 - LICENSE RENEWAL - MB1991 (50-339))
S00057	CRDM CRACKING
S00058	PTS RULE
S00063	MCGUIRE UNIT 1 - LICENSE RENEWAL (50-369)
S00064	MCGUIRE UNIT 2 - LICENSE RENEWAL (50-370)
S00065	CATAWBA UNIT 1 - LICENSE RENEWAL (50-413)
S00066	CATAWBA UNIT 2 - LICENSE RENEWAL (50-414)
S00067	PEACH BOTTOM UNIT 2 - LICENSE RENEWAL (50-277)
S00068	PEACH BOTTOM UNIT 3 - LICENSE RENEWAL (50-278)
Z02001	GENERAL ADMINISTRATION

From: "Powers, Dana A" <dapower@sandia.gov>
To: "'jtl@nrc.gov'" <jtl@nrc.gov>, "APOSTOLAKIS, George" <apostola@mit.edu>, "BONACA,Mario" <mvbonaca@snet.net>, "FORD, F. Peter" <FPCTFord@aol.com>, "KRESS, T.S." <TSKress@aol.com>, "LEITCH, Graham" <gmleitch@aol.com>, "ROSEN, Steve" <historyart@computron.net>, "SHACK, Bill" <wjshack@anl.gov>, "sieber, jACK" <jdsieber@aol.com>, "WALLIS, Graham B." <Graham.B.Wallis@Dartmouth.edu>
Date: 5/22/02 2:30PM
Subject: PRA, License Renewal, Power Uprates and the Future

There have been quite a few discussions within the ACRS meetings and via e-mail concerning the utility of PRA with respect to Option 2, power uprates, license renewal, shutdown operations, fire, and external events. Frequently, the conclusion reached is that PRAs simply do not treat the pertinent material. That is, the PRAs are just too limited to apply broadly to the issues at hand.

Recently, I have had a glimpse of what I think to be the future in computation provided by what is going on at the national laboratories. Today, a 64 processor computer is considered a 'small' system that has heroic stability (operating for months without a component failure or random failure). Such systems are becoming routine equipment for groups of engineers as small as 10. Middle level computing for production purposes now involves systems with on the order of several hundred to a thousand processors. Each processor is a Pentium IV (sorry Bill Macs just are not players here) operating at 2.5 GHz. Each node has a GB of memory so truly heroic data bases can be handled. These systems and their software are being farmed for operation by commercial ventures because they no longer represent the state of the art in computing and are grossly inadequate for problems in the NNSA labs!

The point of all this is that PRAs were designed when computational horsepower was small. Computational horsepower is exploding and industries and technical fields are taking advantage of this enhanced horsepower to address problems that they could never address before. But, we don't see the same drive in the PRA world. All that is happening is whining about the limitations of the PRA. If a plant has 40,000 components in it, why can't the PRA address each and everyone of them? Why can't the PRA be intimately coupled to detailed phenomenological codes? Why can't we treat aging effects in PRAs? Why can't we do the massive Monte Carlo shutdown analysis that Dr. Kress advocates?

It appears to me that the nuclear industry is being left behind in the current computational era with codes that can run on laptops but don't yield anything useful. Should not ACRS address this? Are we content to have manpower at NRC remain in its current low productivity phase because there is not the willingness to make use of computational resources the way others are doing?

Dana

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