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

January 25, 2001

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

SUBJECT: SUMMARY REPORT - 478TH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, ON DECEMBER 6-9, 2000, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

Dear Chairman Meserve:

During its 478th meeting, December 6-9, 2000, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report and letters. In addition, the Committee authorized Dr. John T. Larkins, Executive Director, ACRS, to transmit the memoranda noted below:

REPORT

- Issues Associated with the Thermal-Hydraulic Codes (Report to the Honorable Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated January 11, 2001)

LETTERS

- Nuclear Energy Institute Draft Report, NEI 99-03, "Control Room Habitability Assessment Guidance" (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, ACRS, dated December 14, 2000)

- Proposed Final Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, ACRS, dated December 15, 2000)

- Differing Professional Opinion on Steam Generator Tube Integrity (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, dated December 29, 2000)

MEMORANDA

- South Texas Project, Units 1 and 2 - Draft Safety Evaluation on Exemption Requests from Special Treatment Requirements of 10 CFR Parts 21, 50, and 100 (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated December 13, 2000)

- Proposed Revision to the Commission's Safety Goal Policy Statement for Reactors (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated December 14, 2000)

- Physical Separation of Circuits for Low Pressure Emergency Core Cooling Systems (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated December 20, 2000)

HIGHLIGHTS OF KEY ISSUES CONSIDERED BY THE COMMITTEE

1. Issues Associated with Core Power Uprates

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning issues associated with core power uprates. In light of licensees submitting requests for extended 15-20% of nominal core power) power uprates, the Committee members expressed concern regarding the scope and method of the staff's review procedure and the potential for synergistic effects with other licensing actions that could compete with existing safety margins. NRC staff representatives presented the staff's review procedure for uprate requests, application of risk-informed decision making on power uprates, potential synergistic effects such as high burnup fuel and erosion/corrosion, and the perspective of the Office of Nuclear Regulatory Research on extended uprate applications.

Conclusion

The Committee will take this matter under advisement, pursuant to its review of proposed licensee power uprate applications. The Committee will also pursue follow-on action regarding this issue during its "Planning and Procedures Meeting" on January 22-24, 2001, when it holds a discussion on the topic of "Identification and Quantification of Design Margins."

2. Differing Professional Opinion (DPO) on Steam Generator Tube Integrity

The Chairman of the ACRS ad hoc Subcommittee on DPO provided a report to the Committee on the proposed conclusions and recommendations of the subcommittee associated with the technical merits of the DPO issues. The NRC staff, DPO author, and the Committee members discussed the impact of the tube support plate movement during depressurization on the steam generator tubes.

Conclusion

The Committee approved the conclusions and recommendations of the Subcommittee. Also, it approved a letter to the EDO to transmit the NUREG-xxx report prepared by the subcommittee subject to editorial changes to the report.

3. Thermal-Hydraulic Phenomena Subcommittee Report

The Thermal-Hydraulic Phenomena Subcommittee Chairman provided a report regarding the results of the November 13-14, 2000 Subcommittee meeting, associated with its review of the General Electric (GE) Nuclear Energy TRACG realistic thermal-hydraulic code. The Committee's discussion focused on the strengths and weaknesses of GE's code modeling approach and the application of the code to the modeling of anticipated operational occurrences.

Conclusion

The Subcommittee plans to continue its discussion of the TRACG code at future meetings.

4. Plant Systems Subcommittee Report

The Committee received a report by the Chairman of the Plant Systems Subcommittee on the results of a meeting held on October 31, 2000, regarding ABB/CE Siemens digital instrumentation and control (I&C) applications and the insights gained from a meeting held in November with the German Reactor Safety Committee (RSK) on digital I&C. Dr. Uhrig stated that the meeting with the German RSK was a followup to an exchange held at MIT in Cambridge, MA, in 1999. A principal focus of the meeting was to discuss safety issues associated with the utilization of digital I&C systems in light water reactors. The representatives of the NRC staff also clarified the issues related to digital I&C presented by the staff at the International Symposium on Software Reliability Engineering.

Conclusion

The Committee will continue its discussion of the digital I&C topical reports and other digital I&C issues during future ACRS meetings.

5. Meeting with NRC Commissioner Diaz

The Committee held discussions with Commissioner Diaz regarding the NRC Safety Research Program and other items of mutual interests.

6. South Texas Exemption Request

The Committee heard presentations by and held discussions with representatives of the NRC staff and South Texas Project (STP) concerning the recently issued draft safety evaluation on the STP request for exemptions from certain special treatment requirements of 10 CFR Parts 21, 50, and 100.

The safety evaluation was developed in response to the risk-informed exemption requests from the special treatment requirements of 10 CFR Parts 21, 50, and 100 submitted by STP on July 13, 1999, and supplemented on October 14 and 22, 1999, and January 26 and August 31, 2000. The submittal sought approval of processes for categorizing the safety significance of structures, systems, and components (SSCs) and treatment of those SSCs as the principal basis for granting the exemptions. STP used the NRC approved graded Quality Assurance (GQA) program to categorize certain SSCs in the plant.

During the meeting, STP provided a list of the regulations that they were seeking to exclude SSCs from the scope of requirements. These were: 10CFR Part 21 (Defect Notification); 10CFR50.34 (Appendix B Treatment); 10CFR 50.49 (EQ); 10CFR 50.54 (QA Program); 10CFR 50.55a (ASME); 10CFR 50.59 (Change Evaluation); 10CFR 50.65 (Maintenance Rule); Appendix A, GDCs 1,2,4,18 (QA, Seismic, EQ, 1E); Appendix B (QA Program); Appendix J (RCB Leak Testing); and 10CFR Part 100 (Seismic).

The NRC staff presentation emphasized the importance of the fact that risk-informed decisionmaking meets current regulations, is consistent with defense-in-depth, maintains sufficient safety margins, and that increases in CDF/risk are small. The staff listed a number of open items relating to the categorization and the treatment processes.

Conclusion

A memorandum was sent to the Executive Director for Operations dated December 13, 2000, indicating that the safety evaluation was not complete because of a number of open items. Because of the significance of the open items, an ACRS Subcommittee plans to discuss this matter at a future meeting.

7. Control Room Habitability

The Committee heard presentations by and held discussions with representatives of the Nuclear Energy Institute (NEI) and the NRC staff regarding the resolution of issues associated with control room habitability. Mr. P. Lagus, Lagus Technology Inc., also provided comments relative to aspects of control room tracer gas inleakage measurement. In response to NRC staff concerns, NEI developed a guidance document, NEI 99-03, "Control Room Habitability Assessment Guidance." NEI believes that the current draft of this document adequately addresses the staff's concerns with control room habitability. The staff agrees that the industry has made significant progress, but some key issues, primarily associated with testing and maintenance of acceptable inleakage values, remain unresolved. The staff has decided to pursue a regulatory approach to resolve this matter, with the expected development of a regulatory document and use of the public comment process.

Conclusion

The Committee provided a report to the Executive Director for Operations on this matter, dated December 14, 2000.

8. Proposed Final Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning the proposed final Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The Committee also considered the guidance provided in NUREG/CR-6115, "PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions." The Committee and staff discussed accuracy and reliability issues associated with the various methods used to determine vessel fluence. The wide variation in calculation methods has resulted lengthy plant-specific reviews. The proposed Regulatory Guide was developed to provide standardized methods and procedures to simplify and expedite these reviews.

Conclusion

The Committee sent a letter dated December 15, 2000, to the Executive Director for Operations on this matter.

9. Proposed Modifications to the Commission's Safety Goal Policy Statement for Reactors

The Committee heard a presentation by and held discussions with a representative of the NRC staff concerning the proposed modifications to the Commission's Safety Goal Policy Statement for reactors. The Committee considered the staff's proposed modifications which reflect Commission direction in the Staff Requirements Memorandum dated June 27, 2000 (SECY-00-0077). Individual ACRS members suggested a number of changes that the staff agreed to consider.

Conclusion

The ACRS Executive Director sent a memorandum dated December 14, 2000, to the Executive Director for Operations on this matter.

10. NRC Safety Research Program

The Committee continued its discussion of the NRC Safety Research Program and the format and content of the ACRS 2001 report to the Commission.

Conclusion

The Committee will continue its discussion and preparation of the ACRS 2001 report to the Commission on the NRC research program during future ACRS meetings.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

The Committee did not receive any EDO responses for discussion during this meeting.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from November 1 through December 5, 2000, the following Subcommittee meetings were held:

- Safety Research Program - November 1, 2000

  The Subcommittee discussed the 2001 draft ACRS report to the Commission regarding the NRC Safety Research Program.

- Thermal Hydraulic Phenomena - November 13-14, 2000

  The Subcommittee reviewed the GE Nuclear Energy TRACG thermal-hydraulic code, and continued the review of the NRC Office of Nuclear Regulatory Research (RES) thermal-hydraulic research program pursuant to development of the ACRS annual safety research report.

- Severe Accident Management - November 15, 2000

  The Subcommittee reviewed RES' severe accident management program and continued the review of activities between the NRC staff and the nuclear industry pursuant to the revisions of NEI document 99-03, "Control Room Habitability Assessment Guidance."

- Materials and Metallurgy - November 16, 2000

  The Subcommittee discussed the proposed draft regulatory guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

- Planning and Procedures - December 5, 2000

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

LIST OF FOLLOW-UP MATTERS FOR THE EXECUTIVE DIRECTOR FOR OPERATIONS

- An ACRS Subcommittee plans to review the South Texas Project exemption requests for special treatment requirements and the associated NRC staff safety evaluation after resolution of significant open issues.

- The Committee requested an opportunity to review the staff's resolution of the issues associated with physical separation of circuits for low pressure emergency core cooling systems.

PROPOSED SCHEDULE FOR THE 479TH ACRS MEETING

The Committee agreed to consider the following topics during the 479th ACRS Meeting, February 1-3, 2001:

Treatment of Uncertainties in the Elements of the PTS Technical Basis Reevaluation Project

Briefing by and discussions with representatives of the NRC staff regarding treatment of uncertainties in the elements of the Pressurized Thermal Shock (PTS) Reevaluation Project.

Siemens S-RELAP5 Appendix K Small-Break LOCA Code

Briefing by and discussions with representatives of the NRC staff and Siemens Power Corporation regarding the Siemens S-RELAP5 Appendix K Small-Break Loss-of-Coolant Accident (LOCA) Code and the associated NRC staff Safety Evaluation Report.

Proposed ANS Standard on External-Events PRA

Briefing by and discussions with representatives of the American Nuclear Society (ANS) regarding the proposed ANS Standard on external events PRA.

Reprioritization of Generic Safety Issue-152, "Design Basis for Valves that Might be Subjected to Significant Blowdown Loads"
Briefing by and discussions with representatives of the NRC staff regarding reprioritization of Generic Safety Issue-152 and the reasons therefor, and related matters.

Regulatory Effectiveness of the ATWS Rule

Briefing by and discussions with representatives of the NRC staff regarding the staff's assessment of the regulatory effectiveness of the Anticipated Transients Without Scram (ATWS) Rule (10 CFR 50.62).

Overview of Mixed Oxide Fuel Fabrication Facility

Briefing by and discussions with representatives of the Department of Energy (DOE) and the NRC staff regarding the proposed Mixed Oxide Fuel Fabrication Facility to be constructed at the DOE's Savannah River Plant site.

Meeting with the NRC Chairman

Meeting with the NRC Chairman Meserve to discuss items of mutual interest.

NRC Safety Research Program

Discussion of the annual ACRS report to the Commission on the NRC Safety Research Program.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

CERTIFIED

Date Issued: 2/8/2001
Date Certified: 2/16/2001

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December 6-9, 2000

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- Differing Professional Opinion on Steam Generator Tube Integrity (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, dated February 1, 2001)

MEMORANDA

- South Texas Project, Units 1 and 2 - Draft Safety Evaluation on Exemption Requests from Special Treatment Requirements of 10 CFR Parts 21, 50, and 100 (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated December 13, 2000)

- Proposed Revision to the Commission's Safety Goal Policy Statement for Reactors (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated December 14, 2000)

- Physical Separation of Circuits for Low Pressure Emergency Core Cooling Systems (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated December 20, 2000)

APPENDICES

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

MINUTES OF THE 478TH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS DECEMBER 6-9, 2000 ROCKVILLE, MARYLAND

The 478th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on December 6-9, 2000. Notice of this meeting was published in the *Federal Register* on November 17, 2000 (65 FR 69578) (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 was kept and 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 Ann Riley & Associates, Ltd., 1025 Connecticut Avenue, N.W., Suite 1014, Washington, D.C. 20036, and on the ACRS/ACNW Web page at (www.NRC.gov/ACRS/ACNW).]

ATTENDEES

ACRS Members: Dr. Dana A. Powers (Chairman), Dr. George Apostolakis (Vice Chairman), Dr. Mario V. Bonaca, Dr. Thomas S. Kress, Mr. Graham M. Leitch, Dr. William J. Shack, Dr. Robert L. Seale, Mr. John D. Sieber, Dr. Robert E. Uhrig, 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. Dana A. Powers, 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. Issues Associated with Core Power Uprates (Open)

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

Dr. Bonaca, cognizant ACRS Member for this matter, introduced this topic to the Committee. He noted that concern had been expressed about the numerous plants (BWRs) that are planning significant core power uprates (15-20% of nominal). Specifically, he said that concerns were cited regarding: lack of a NRC Standard Review Plan (SRP) section for power uprate reviews, potential synergistic effects with other licensing actions that may compete with existing margins, and the potential increase in plant risk that is not evaluated for such deterministic licensing actions. Dr. Bonaca said that representatives of the NRC staff were present to address these and other related matters.

NRC Staff Presentation

Mr. Barrett, Office of Nuclear Reactor Regulation (NRR), said that the staff had three objectives for this discussion; (1) provide information on the issues the staff has considered in uprate reviews, both deterministic and risk-based; (2) engage the Committee in a dialogue to flesh out the ACRS's specific concerns; and, (3) determine if any course corrections are needed on the staff's part with regard to this issue, and/or, is any effort needed by the Office of Nuclear Regulatory Research (RES).

NRC representatives provided presentations on the following topics:

- Overview and Need for SRP Section
- Application of Risk-Informed Decision Making on Power Uprates
- Use of More "Realistic" Analyses in Support of Power Uprates
- Potential Synergistic Effects - High Burn-Up Fuel at Uprate Conditions, Accelerated Erosion/Corrosion Due to Uprates/Aging
- RES Perspectives on Extended Power Uprates

Key points noted by the staff were:

- A number of plants have received power uprates over the years, beginning with so-called capacity recapture pursuant to the Atomic Energy Commission (AEC) policy for provisional operating license plants. Lately, the majority of uprates have been for GE BWR plants in accordance with the GE initial 5% uprate program and the extended uprate program for power increases of 15-20% of nominal. The staff does not plan to develop a Standard Review Plan Section for uprate reviews, believing that past reviews have formed templates for subsequent reviews.

- The current scope of risk evaluations for extended power uprates provides adequate insights to support these uprates. No operational data has shown significant degradations in plant structures, systems, and components (SSCs) availabilities, reliability, or initiating event frequencies, which calls into question the need for performing risk assessments to address all possible synergistic uprate effects. NRC is considering the need to assess operational data for plants with extended power uprates.

- NRR is mindful of the potential for reduction in plant margin and increase in risk associated with extended power uprates. A review process is in place and the staff plans to conduct audits of selected power uprate calculations to ensure that methodologies are being used in accordance with conditions of approval. Use of "best-estimate" analyses may be necessary for future BWR power uprates. These analyses will be reviewed by the staff. NRC has a program plan in place to address use of high-burnup fuel.

- The staff has concluded that BWR power uprates will not cause an adverse increase in flow-induced erosion/corrosion damage to the reactor coolant system (RCS) piping. Licensees are required to reexamine their inspection programs in light of plant-specific uprate concerns.

- RES is considering an initiative to explore issues related to power uprates using both deterministic and risk-informed approaches. This program does not have user need support and would be conducted under RES's role to improve knowledge of uncertainties and margins to permit improved regulatory positions.

Committee Comments

Committee Members noted the following points during the discussion:

- The staff needs to ensure adequate review time is available and not be driven solely by a licensees' schedule.

- Copies of the Maine Yankee Lessons Learned report were requested by the Committee.

- Citing the Swiss (Leibstadt plant) study showing a significant increase in risk for fission produce release, the staff said that this work may offer risk insights that warrant its attention.

- Dr. Kress opined that the synergetic effects of higher burnup given power uprate should be evaluated by the staff. Dr. Apostolakis asked if power uprates will impact the risk associated with shutdown operations.

- Dr. Bonaca said that the risk impacts of power uprate on such issues as containment margin and fatigue are not being evaluated in probability risk assessments (PRAs).

- Dr. Eltawila noted that RES lacks resources to initiate the above-noted work at this time. The RES goal with this work is to determine what parameters are limiting for power uprates. Dr. Apostolakis suggested that RES attempt to quantify the impact of power uprate on safety margins.

Mr. Barrett assured the Committee that the staff is addressing issues associated with core power uprates, and intends to continue on the review course noted above. He said that plant risk is evaluated under the guidelines specified in SECY-99-246. NRR will take the results of the RES initiative into account for its uprate reviews.

Conclusion

The Committee will take this matter under advisement, pursuant to its review of proposed licensee power uprate applications. The Committee will also pursue follow-on action regarding this issue during its expanded Planning and Procedures Subcommittee on January 22-24, 2001, when it holds a discussion on the topic of "Identification and Quantification of Design Margins".

III. Differing Professional Opinion (DPO) on Steam Generator Tube Integrity (Open)

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

Dr. Powers, the Chairman of the ad hoc Subcommittee on the DPO provided a preamble, stated that the Subcommittee held a meeting in October 2000 to gather information from the DPO author and the NRC staff regarding the DPO issues for use in developing proposed positions for consideration by the full Committee. In the course of developing a report, the Subcommittee realized the need to get additional information on the impact of tube support plate movements during depressurization events on steam generator tubes. To obtain more information on this issue, the staff and the DPO author were invited to provide a presentation to the Committee.

NRC Staff Presentation - Mr. Karwoski, NRR

Mr. Karwoski stated that Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes," allows predominantly axially oriented outside diameter stress corrosion cracking at the tube support plate elevation to remain in service. The degradation occurs in the crevice between the tube and tube support plate, and the crevice is typically packed with corrosion products such as magnetite. The fundamental goal of the repair criteria specified in Generic Letter 95-05 is to ensure the structural and leakage integrity of the steam generator tubes. Burst and peak testing performed on tubes pulled from operating steam generators and on specimens produced at the laboratories, without the presence of the tube support plates, exposed all degradation at the support plates. If the tube support plates move during a depressurization event, the Generic Letter 95-05 methodology will provide an appropriate determination of the conditional probability of burst and the postulated accident conditions. The results of the assessment performed by a licensee to determine the potential for tube support plate displacement during postulated steam line break revealed that the plates are essentially locked in place due to corrosion product buildup in the tube-to-tube support plate crevice. The staff has raised several issues with the licensee's assessment.

In response to a question from Dr. Powers, Mr. Karwoski said that if the tube support plate bends or flexes, the corrosion products may get loose and expose the degradation resulting tube leak. In response to another question from Dr. Powers, Mr. Strosnider, NRR, said that if the support plate is assumed to pick up some load during a depressurization event that load will be distributed to all the tubes that are locked into the support plate.

In response to a question from Dr. Powers regarding the impact of the tube support plate movement on the tube cracks that may exist away from the support plates, Mr. Karwoski said that if the plate flexes and puts different types of forces on the tube there is a potential to deform the tube. If the tube is deformed to a certain level, it could open up the crack.

Dr. Hopenfeld, the DPO author reiterated his views on this matter, stating that tube support plates can be lifted during the depressurization events and this can cause cracks in tubes penetrated through the tube walls resulting in additional flow from the primary coolant system to the secondary side of the coolant system. The staff did not provide adequate guidance to the licensees to evaluate this situation.

Proposed Subcommittee Conclusions and Recommendations

Dr. Powers, Chairman of the ad hoc Subcommittee on the DPO, presented the proposed conclusions and recommendations of the Subcommittee. The members discussed each conclusion and recommendation and suggested some changes.

Conclusion

The Committee endorsed the conclusions and recommendations of the Subcommittee, which are included in a Draft NUREG-xxx report prepared by the Subcommittee Chairman with input from Subcommittee members and consultants. The Committee approved a letter to transmit the NUREG-xxx report to the EDO subject to editorial changes to the report.

IV. Thermal-Hydraulic Phenomena Subcommittee Report (Open)

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

The Thermal-Hydraulic Phenomena Subcommittee discussed the results of its November 13-14, 2000 Subcommittee meeting, regarding the initiation of its review of the GE Nuclear Energy TRACG realistic thermal-hydraulic code. The discussion focused on the strengths and weaknesses of GE's code modeling approach and the application of the code to the modeling of anticipated operational occurrences.

V. Plant Systems Subcommittee Report (Open)

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

The Committee received a report by the Chairman of the Plant Systems Subcommittee on the results of a meeting held on October 31, 2000, regarding ABB/CE Siemens digital instrumentation and control (I&C) applications and the insights gained from a meeting held in November with the German Reactor Safety Committee (RSK) on digital I&C. Dr. Uhrig stated that the meeting with the German RSK was a followup to an exchange held at the Massachusetts Institute of Technology in Cambridge, MA. A principal focus of the meeting was to discuss safety issues associated with the utilization of digital I&C systems in light water reactors. The representatives of the NRC staff also clarified the issues related to digital I&C presented by the staff at the International Symposium on Software Reliability Engineering.

Conclusion

The Committee will continue its discussion of the digital I&C topical reports and other digital I&C issues during future ACRS meetings.

VI. Meeting with Commissioner Diaz

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

The Committee held discussions with Commissioner Diaz regarding the NRC Safety Research Program and other items of mutual interests.

VII. South Texas Project Exemption Request

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

Mr. John D. Sieber, Chairman of the ACRS Subcommittee on Plant Operations, introduced this topic to the committee. He indicated that the Committee had not discussed this topic or the safety evaluation previously.

NRC Staff and Industry Presentations

The staff presentation on the South Texas Project exemption request safety evaluation was made by Mr. John Nakoski, NRR. In attendance and responding to questions as needed were Mr. Richard Barrett, Mr. Gene Imbro, and Mr. Stuart Richards, all of NRR. The South Texas Project presentation on the NRR safety evaluation was made by Mr. Mark McBurnett, Mr. Rick Grantom, and Mr. Glen Schinzel.

NRC regulations in 10 CFR Parts 21, 50, and 100 contain special treatment requirements that impose controls to ensure the quality of SSCs that are safety-related, important to safety, or otherwise within the scope of the regulations. These special treatment requirements include quality assurance (QA), environmental and seismic qualification, inspection and testing, and performance monitoring. The scope of these regulations applies to some SSCs that have little or no safety or risk significance.

The NRC approved a risk-informed process for determining the safety significance of SSCs as part of the Graded Quality Assurance (GQA) Program for South Texas Project (STP), Units 1 and 2. Using this process, STP Nuclear Operating Company (STPNOC) categorized certain SSCs in the plant as low safety significant or non-risk significant, and other SSCs as high safety significant or medium safety significant. STPNOC found that in practice, the GQA Program was limited in implementation by the special treatment requirements imposed by the regulations for SSCs that are safety-related, important to safety, or otherwise within the scope of the regulations.

The safety evaluation was developed in response to the risk-informed exemption requests from the special treatment requirements of 10 CFR Parts 21, 50, and 100

submitted by STPNOC on July 13, 1999, and supplemented on October 14 and 22, 1999, and January 26 and August 31, 2000. The submittal sought approval of processes for categorizing the safety significance of SSCs and treatment of those SSCs consistent with their categorization as the principal basis for granting the exemptions. The scope of the exemption requests include those safety-related SSCs categorized as low safety significant or non-risk significant using STPNOC's categorization process.

During the STP presentation, a list was given of the regulations that they were seeking to exclude SSCs from the scope of requirements and feedback from the draft safety evaluation pending resolution of open items. These were: 10CFR Part 21 (Defect Notification); 10CFR50.34 (Appendix B Treatment); 10CFR 50.49 (EQ); 10CFR 50.54 (QA Program); 10CFR 50.55a (ASME); 10CFR 50.59 (Change Evaluation); 10CFR 50.65 (Maintenance Rule); Appendix A, GDCs 1,2,4,18 (QA, Seismic, EQ, 1E); Appendix B (QA Program); Appendix J (RCB Leak Testing); and 10CFR Part 100 (Seismic).

STP finds SECY-98-0300 Option 2 a challenge. They feel that it is important to focus on the adequacy of categorization of SSCs and that benefits are expected in procurement of seismic, EQ, ASME, and 1E replacement components.

The NRC staff presentation focused on some of the same issues as STP. They emphasized the importance that risk-informed decision making meets current regulations, is consistent with defense-in-depth, maintains sufficient safety margins, increases in CDF/risk are small, and monitors its impact.

The staff listed a number of open items relating to the categorization process and the treatment processes. The preliminary assessment of exemptions is to approve 10 CFR 21.3; 50.55a; 50.59; 50.65(b); Part 50, App. B; and Part 50, App. J. The staff's preliminary finding is that exemptions to GDC 1, 2, 4, 18 are not necessary. They would deny 10 CFR 50.34(b)(6)(ii); 50.54(a)(3); and 50.55a(g). And finally, the staff has determined that more information is needed on 10 CFR 50.34(b)(11); 50.49(b); 50.55a(h); and Part 100, App. A, VI, (a)(1) and (2).

The ACRS Committee is concerned about the number of open items and the Risk Achievement Worth equation contained in the response to RAI # 20.

Conclusion

A letter was sent to the Executive Director of Operations indicating that the safety evaluation was not complete because of a number of open items. Because of the significance of the open items, the Committee requested a subcommittee briefing.

VIII. Control Room (Open)

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

Dr. Kress, cognizant ACRS Member for this matter, introduced this topic to the Committee. He noted that the Severe Accident Management Subcommittee discussed this matter during its November 15, 2000 meeting. He noted that in response to concerns regarding control room (CR) habitability, dating from the early-1980s, the Nuclear Energy Institute (NEI) developed a guideline document, NEI 99-03. Subsequently, NRC and NEI engaged in extensive interaction on revision of 99-03, and some open issues remain. Two of these are: is the use of component testing for establishment of control room baseline inleakage acceptable, and, should the limiting inleakage rate be included in plant technical specifications.

NEI Presentation

NEI representatives provided an overview of the revised 99-03 document. Key points noted in the discussion included:

- NEI advocates use of component testing to establish baseline inleakage rates. Component testing would be limited to use on pressurized control rooms, where it provides a reasonable assurance of accurate inleakage measurement.

- NEI has established a CR habitability program that includes periodic assessment of inleakage, configuration control procedures, and training requirements.

- A quantitative program has been developed to address the issue of CR smoke infiltration.

- NEI has instituted analysis improvements to make use of the Alternative Source Term insights. Improvement requests have been proposed for source term limits and meteorology and dispersion modeling. NRC has these improvement requests under review.

- With the adoption of a CR habitability program, NEI does not believe that technical specification requirements on inleakage limits is necessary.

- NEI plans to develop alternatives to the staff's positions on technical specification requirements, inleakage test options, and elimination of excessive conservatisms. NRC comments on the revised 99-03 document were requested by the end of December. The revised report is to be issued to industry for review in January 2001.

Dr. Kress questioned NEI's position regarding lack of a specified periodic assessment program. NEI indicated that they are reevaluating this matter and are developing more robust guidance. In response to Dr. Kress, NEI said that its opposition to incorporating inleakage requirements in technical specifications is a fear of unnecessary plant shutdowns. Dr. Kress asked if NEI 99-03 could be endorsed via a regulatory guide. NEI indicated that NRC does not agree with that approach.

NRC Staff Presentation

Mr. J. Hayes, NRR, discussed the background, chronology of development of NEI 99-03 accomplishments to date, and remaining discussion areas. He indicated that the staff is asking the industry to work together to develop realistic evaluations for radiological considerations, where both conservatisms and non-conservatisms are removed from the analyses.

The staff has decided that it will seek a regulatory approach to resolution of this matter, by development of a generic licensing document (most likely a regulatory guide). This will both ensure an equitable resolution of this matter and allow public comment on the approach chosen.

Comments by P. Lagus

Dr. P. Lagus, Lagus Applied Technology, Inc., provided comments relative to uncertainty and repeatability of tracer gas inleakage measurements. His remarks focused on methodology for assessing measurement uncertainty given a small repeat sample, treatment of systematic and random uncertainties, and defining an acceptable value for inleakage.

Conclusion

The Committee issued a report to the Executive Director for Operations on this matter dated December 14, 2000

IX. Proposed Final Regulatory Guide DG-1053, "Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (Open)

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

Dr. William Shack, Chairman of the ACRS Subcommittee on Materials and Metallurgy introduced the topic to the Committee. He stated that the purpose of this meeting was to review the proposed final Regulatory Guide DG-1053, "Calculational and Dosimetry

Methods for Determining Pressure Vessel Neutron Fluence," and associated guidance in NUREG/CR-6115, "PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions." Dr. Shack stated that proposed guidance uses a best-estimate methodology rather than a bounding conservative estimate of the neutron fluence. The ACRS Subcommittee on Materials and Metallurgy met on November 16, 2000, to discuss this matter.

NRC Staff Presentation

Dr. Nilesh Chokshi, RES, provided an overview presentation and led the discussion for the NRC staff. Mr. William Jones, RES, presented the general content and objectives of the proposed regulatory guide. Mr. John Carew, NRC contractor at Brookhaven National Laboratories, presented the methodology and technical bases. Mr. Lambrose Lois, NRR, provided supporting discussion. Significant points made during the presentation include:

- Requirements for fracture toughness during normal and anticipated operational occurrences are provided in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," and for potential pressurized thermal shock events in 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock."

- In the past, various methods of determining fast neutron fluence were implemented in numerous ways of varying reliability and accuracy resulting in lengthy plant-specific reviews. He noted that the proposed regulatory guide was developed to provide standardized methods and procedures to simplify and expedite NRC review of licensee submittals.

- Vessel neutron fluence is very difficult to measure because the core barrel, thermal shield, and other vessel internal components attenuate the neutron flux. The proposed fluence calculational methods are best-estimate rather than bounding analyses. The methodology is designed to provide accuracy within the regulatory requirement of 20%, for fluence in the energy in the range from 15MeV to 0.1 MeV. NUREG/CR-6115 provides for qualification via benchmarking and uncertainty analysis. Vessel fluence uncertainty is determined by calculating a weighted mean value. A major enhancement in the proposed guidance is the applicability of the methodology to boiling water reactors.

Dr. Powers noted that there is always data scatter in nil-ductile transition and questioned how accurate measurements need to be. The staff reiterated the regulatory requirement that measures must be accurate within 20%. However, the staff noted that the Monte Carlo calculations are very extensive and produce useful results.

Dr. Wallis questioned what activities the industry was pursuing while the NRC developed the DG-1053 methodology. The staff stated that several licensees used the draft methodology in preparing submittals for consideration by the NRC. The staff noted that other licensees used a variety of methods.

Dr. Wallis questioned what was meant by a "weighted mean." He also questioned whether the results could be manipulated by varying the weighting factors. The staff stated that it represents the weighted average of the inputs from three experts. The staff explained that the methodology allows for licensees to use good engineering judgment in making calculations. The staff noted that NUREG/CR-6115 provides additional guidance with problem definitions and reference solutions. The staff acknowledged, however, that the proposed regulatory guide and associated NUREG will not prevent willful manipulation of the analysis.

Conclusion

The Committee sent a letter dated December 15, 2000, to the Executive Director for Operations on this matter.

X. Proposed Modifications to the Commission's Safety Goal Policy Statement for Reactors (Open)

Dr. George Apostolakis, Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment introduced the topic to the Committee. He stated that the purpose of this meeting was to review the proposed revision to the Commission's Safety Goal Policy Statement for reactors. The Committee considered the staff's proposed modifications in response to Commission direction provided in the Staff Requirements Memorandum (SRM) dated June 27, 2000 (SECY-00-0077).

NRC Staff Presentation

Mr. Joseph Murphy, RES, provided a brief presentation to the Committee. Significant points made during the presentation include:

- The Safety Goal Policy Statement has been revised consistent with the SRM dated June 27, 2000. The Commission supported the use of a qualitative statement expressing the Commission's intent to protect the environment and that the NRC will consider the need to minimize adverse environmental impacts in its regulatory decision making. The staff modified the Policy Statement to include a statement that "there be no adverse impact on the environment."

- The policy statement was also modified to state that safety goals are "goals" and not limits. The staff also modified the policy statement to maintain core damage

frequency (CDF) and large, early release frequency (LERF) as subsidiary objectives.

- The policy statement was modified to provide a discussion on treatment of uncertainty using Regulatory Guide 1.174, to incorporate the Commission's White Paper (SECY-98-144) definition of defense-in-depth, and to present a working definition of "how safe is safe enough."

Dr. Wallis questioned what value there is in a goal if you do not achieve it. The staff stated that the safety goal is a multi-dimensional entity that is met in some ways but may not be in others.

Dr. Kress questioned whether the policy statement defines "how safe is safe enough" qualitatively as directed by the Commission. He suggested that CDF and LERF also be considered in addition to meeting the regulations. Dr. Kress also suggested that it may be worthwhile to consider favorable results in the revised reactor oversight process as qualitatively meeting the safety goals. The staff reiterated that the policy statement was revised to be consistent with current regulatory practice and Regulatory Guide 1.174 for risk-informed decisionmaking.

Dr. Apostolakis questioned the use of individual versus societal risk. He noted that the proposed Policy Statement allows for the use of Safety Goals on a plant-specific basis, yet they are still compared to national averages. The staff stated that the risk is based on a critical individual located in close proximity to the plant, i.e., within the 10-mile emergency planning zone.

Individual ACRS members suggested a number of changes to the proposed policy statement that the staff agreed to consider.

Conclusion

The ACRS Executive Director sent a memorandum dated December 14, 2000, to the Executive Director for Operations on this matter.

XI. NRC Safety Research Program (Open)

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

The Committee continued its discussion of the NRC Safety Research Program and the format and content of the ACRS 2001 report to the Commission. Some of the safety research programs that were discussed included the uncertainty in thermal-hydraulic codes, materials and metallurgy, radiation assisted stress-corrosion cracking,

probabilistic risk assessment, digital instrumentation and control, and the maintenance of core competency.

Conclusion

The Committee will continue its discussion and preparation of the ACRS 2001 report to the Commission on the NRC research program during future ACRS 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 did not receive any EDO responses for discussion during this meeting.

B. Report on the Meeting of the Planning and Procedures Subcommittee

Review of the Member Assignments and Priorities for ACRS Reports and Letters for the December ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the December 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 of the ACRS members through March 2001 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.

Election of ACRS Officers for CY 2001

The election of Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee were held during the December 6-9, 2000 ACRS meeting.

ACRS Retreat for 2001

During the November meeting, the Committee decided to have a retreat on January 22-24, 2001 at Two White Flint North, Room T-2-B3. A proposed list of topics for discussion during the retreat was discussed.

CY 2000 Self-Assessment

During the November ACRS meeting, a list of ACRS work products that were subject to a critical analysis by the ACRS members at the January 2001 ACRS retreat and proposed metrics for use in identifying the lessons learned were provided to the members for comment. A revised list of work products and metrics, which reflect incorporation of the comments received from some members, as well as proposed assignments for each work product discussed.

ACRS Action Plan for CY 2001-2002

During the May 2000 ACRS meeting, the Committee approved the development of an ACRS Action Plan for CY 2001-2002. A draft Action Plan prepared by the ACRS staff was provided to the members for review and comment during the November ACRS meeting. A revised draft that incorporates, as appropriate, comments received from some members and the ACRS staff management was reviewed by the Planning and Procedures Subcommittee during its December 5, 2000 meting. The ACRS Action Plan and Operating Plan will be forwarded to the Commission.

Concern Expressed by a Private Citizen

A concerned citizen contacted Dr. Dana Powers in November 2000 regarding physical separation of circuits for low pressure emergency core cooling systems. The citizen contends that the NRC staff has not adequately reviewed or acted on potential deficiencies in electrical circuit separation. Copies of a General Electric Nuclear Energy Services Information Letter and a Deficiency Evaluation Report (DER) were provided. The DER concludes that due to the large number of circuits involved and the complex redundancy arrangements established by accident analyses, it is unlikely that functionally redundant sets of equipment actually have the extent of physical independence as contemplated in NEDO-10139 and the June 24, 1975 submittal to the NRC.

The NRC staff is aware of this concern and has requested additional information from the licensee for use in determining the adequacy of electrical circuit separation. The staff expects to complete its review of this concern by the end of February 2001.

Burnup Credit

A representative of the RES staff has contacted Dr. Powers recently requesting a meeting with the ACRS/ACNW Joint Subcommittee to discuss a program, which includes analysis of issues related to burnup credit for transportation, that is being developed by RES in response to a request from NMSS. The issue of burnup credit has previously been reviewed by the ACNW as it relates to transportation shipping casks.

Bilateral Exchange

While in Germany for the Bilateral Exchange between the RSK and several ACRS members, the issue of the next quadripartite meeting was discussed. In a recent E-mail from the GRS there was an inquiry as to whether or not the ACRS would support the next quadripartite meeting in Germany and if there were dates and possible issues for the agenda.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 479th ACRS Meeting, February 1-3, 2001.

The 478th ACRS meeting was adjourned at 11:51 a.m. on December 9, 2000.

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

February 16, 2001

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

FROM: Dana A. Powers, Chairman
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFIED MINUTES OF THE 478th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), DECEMBER 6-9, 2000

I certify that based on my review of the minutes from the 478th 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.

Dana A. Powers

Dana A. Powers, Chairman

February 16, 2001
Date

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

February 8, 2000

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 478th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - DECEMBER 6-9, 2000

Enclosed are the proposed minutes of the 478th 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

conference call at the NCD office. Those interested in joining this conference call should contact the appropriate staff member listed above. Records will be kept of all International Watch meetings/conference calls and will be available after the meeting for pubic inspection at NCD.

Signed in Washington, DC, on November 14, 2000.

Ethel D. Briggs,

Executive Director.

[FR Doc. 00–29515 Filed 11–16–00; 8:45 am]

BILLING CODE 6820–MA–M

NUCLEAR REGULATORY COMMISSION

Agency Information Collection Activities: Submission for OMB Review; Comment Request

AGENCY: U.S. Nuclear Regulatory Commission (NRC).

ACTION: Notice of the OMB review of information collection and solicitation of public comment.

SUMMARY: The NRC recently submitted to OMB for review the following proposal for the collection of information under the provisions of the Paperwork Reduction Act of 1995 (44 U.S.C. Chapter 35). The NRC hereby informs potential respondents that an agency may not conduct or sponsor, and that a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

1. *Type of submission, new, revision, or extension:* New.

2. *The title of the information collection:* General Licensee Registration.

3. *The form number, if applicable:* NRC Form 664.

4. *How often the collection is required:* Annually.

5. *Who will be required or asked to report:* General Licensees of the NRC who possess devices subject to registration under 10 CFR 31.5.

6. *An estimate of the number of annual respondents:* 4,300.

7. *An estimate of the total number of hours needed annually to complete the requirement or request:* 1,433 hours annually (4,300 respondents x 20 minutes per form).

8. *An indication of whether Section 3504(h), Pub. L. 96–511 applies:* Applicable.

9. *Abstract:* NRC Form 664 would be used by NRC general licensees to make reports regarding certain generally licensed devices subject to registration. The registration program is intended to allow NRC to better track general licensees, so that they can be contacted or inspected as necessary, and to make sure that generally licensed devices can be identified even if lost or damaged, and to further ensure that general licensees are aware of and understand the requirements for the possession of devices containing byproduct material. Greater awareness helps to ensure that general licensees will comply with the requirements for proper handling and disposal of generally licensed devices and would reduce the potential for incidents that could result in unnecessary radiation exposure to the public and contamination of property.

A copy of NRC Form 664 and the final OMB supporting statement may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room 1F23, Rockville, MD 20852. OMB clearance requests are also available at the NRC Worldwide web site: *http://www.nrc.gov/NRC/PUBLIC/OMB/index.html.* The OMB supporting statement, and NRC Form 664 will be available on the NRC home page for 60 days after the signature date of this notice.

Comments and questions should be directed to the OMB reviewer by December 18, 2000. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after this date.

Amy Farrell, Office of Information and Regulatory Affairs (3150–), NEOB–10202, Office of Management and Budget, Washington, DC 20503, (202) 395–7318.

The NRC Clearance Officer is Brenda Jo. Shelton, 301–415–7233.

Dated at Rockville, MD, on this 13th day of November, 2000.

For the Nuclear Regulatory Commission.

Beth C. St. Mary,

Acting NRC Clearance Officer, Office of the Chief Information Officer.

[FR Doc. 00–29458 Filed 11–16–00; 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 will hold a meeting on December 6–9, 2000, in Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Thursday, October 14, 1999 (64 FR 55787).

Wednesday, December 6, 2000

1 p.m.–1:05 p.m.: Opening Statement by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

1:05 p.m.–3 p.m.: Issues Associated with Core Power Uprates (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding issues associated with core power uprates, including: Staff plans for developing a Standard Review Plan Section for power uprate reviews; staff position regarding the need for applying risk-informed decisionmaking in the review of significant power uprate applications; and other related matters.

3:15 p.m.–4:45 p.m.: Differing Professional Opinion (DPO) on Steam Generator Tube Integrity (Open)—The Committee will hear a report by the Chairman of the Ad Hoc Subcommittee on DPO regarding conclusions and recommendations of the Ad Hoc Subcommittee on the technical merits of the DPO issues, and will meet with representatives of the NRC staff and the DPO author, as needed.

4:45 p.m.–5 p.m.: Subcommittee Report (Open)—The Committee will hear a report by the Chairman of the Thermal-Hydraulic Phenomena Subcommittee regarding the status of review of the GE Nuclear Energy TRACG best-estimate thermal-hydraulic code.

5 p.m.–5:15 p.m.: Subcommittee Report (Open)—The Committee will hear a report by the Chairman of the Plant Systems Subcommittee regarding ABB/CE and Siemens digital I&C applications and insights gained from a meeting with the RSK on digital I&C in Germany during November 2000.

5:30 p.m.–7 p.m.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports. In addition, it will prepare a response to the Commission request that the ACRS provide a detailed discussion on how the perceived weaknesses with industry-developed thermal-hydraulic codes may adversely affect the NRC's regulatory role and provide more specific recommendations on how those weaknesses should be addressed.

Thursday, December 7, 2000

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

8:35 a.m.–9:30 a.m.: Meeting with NRC Commissioner Diaz (Open)—The Committee will meet with NRC Commissioner Diaz to discuss the NRC Safety Research Program and other items of mutual interest.

9:45 a.m.–11:45 a.m.: South Texas Project Exemption Request (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and South Texas Project

Nuclear Operating Company (STPNOC) regarding the STPNOC's exemption request to exclude certain components from the scope of special treatment requirements in 10 CFR Part 50 and the associated NRC staff's Draft Safety Evaluation Report.

12:45 p.m.–2:15 p.m.: Control Room Habitability (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the nuclear industry regarding issues associated with control room habitability and the staff and industry efforts in resolving those issues.

2:30 p.m.–4 p.m.: Proposed Final Regulatory Guide DG–1053, "Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed final Regulatory Guide DG–1053, including the staff's resolution of public comments.

4 p.m.–5 p.m.: Break and Preparation of Draft ACRS Reports (Open)—Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

5 p.m.–7 p.m.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Friday, December 8, 2000

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

8:35 a.m.–10 a.m.: Proposed Modifications to the Commission's Safety Goal Policy Statement for Reactors (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed modifications to the Commission's Safety Goal Policy Statement for reactors.

10:15 a.m.–11:30 a.m.: Annual Report to the Commission on the NRC Safety Research Program (Open)—The Committee will discuss the draft ACRS report to the Commission on the NRC Safety Research Program, and will meet with representatives of the NRC staff, as needed.

1 p.m.–1:30 p.m.: Future ACRS Activities/ Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

1:30 p.m.–1:45 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.

1:45 p.m.–2:15 p.m.: Election of ACRS Officers for CY 2001 (Open)—The Committee will elect a Chairman and Vice Chairman for the ACRS and a Member-at-Large for the Planning and Procedures Subcommittee for CY 2001.

2:15 p.m.–3:15 p.m.: Break and Preparation of Draft ACRS Reports (Open)—Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

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

Saturday, December 9, 2000

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

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

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 11, 2000 (65 FR 60476). In accordance with these 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 Mr. James E. Lyons, 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 Mr. James E. Lyons 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 Mr. James E. Lyons 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 Mr. James E. Lyons (telephone 301–415–7371), between 7:30 a.m. and 4:15 p.m., EST.

ACRS meeting agenda, meeting transcripts, and letter reports are available for downloading or viewing on the internet at http://www.nrc.gov/ACRSACNW.

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., EST, 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 facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

The ACRS meeting dates for Calendar Year 2001 are provided below:

ACRS meeting No.	Meeting dates
.....	January 2001—No meeting.
479	February 1–3, 2001.
480	March 1–3, 2001.
481	April 5–7, 2001.
482	May 10–12, 2001.
483	June 6–8, 2001.
484	July 11–13, 2001.
.....	August 2001—No meeting.
485	September 6–8, 2001.
486	October 4–6, 2001.
487	November 8–10, 2001.
488	December 6–8, 2001.

Dated: November 13, 2000.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 00–29460 Filed 11–16–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on December 5, 2000, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c)(2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

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

November 13, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
478TH ACRS MEETING
DECEMBER 6-9, 2000

WEDNESDAY, DECEMBER 6, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 1:00 - 1:05 P.M. Opening Remarks by the ACRS Chairman (Open)
   - 1.1) Opening statement (DAP/JTL)
   - 1.2) Items of current interest (DAP/NFD/SD)
   - 1.3) Priorities for preparation of ACRS reports (DAP/JTL/SD)

2) 1:05 - 3:00 P.M. Issues Associated with Core Power Uprates (Open) (MVB/GBW/PAB/AWC)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding issues associated with core power uprates, including: staff plans for developing a Standard Review Plan Section for power uprate reviews; staff position regarding the need for applying risk-informed decisionmaking in the review of significant power uprate applications; and other related matters.

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

3:20 - 3:30 ~~3:00 - 3:15~~ P.M. ***BREAK***

3) ~~3:15 - 4:45~~ P.M. 3:30 - 4:10 Differing Professional Opinion (DPO) on Steam Generator Tube Integrity (Open) (DAP/SD/US)
   - 3.1) Report by the Chairman of the Ad Hoc Subcommittee on DPO regarding conclusions and recommendations of the Ad Hoc Subcommittee on the technical merits of the DPO issues.
   - 3.2) Discussion with representatives of the NRC staff and the DPO author, as needed, regarding additional information on DPO issues.

5:05-5:32 (rpt discussion)

4:10 - 4:27 Break

4) ~~4:45 - 5:00~~ P.M. 4:27 - 4:47 Subcommittee Report (Open) (GBW/PAB)
   Report by the Chairman of the Thermal-Hydraulic Phenomena Subcommittee regarding the status of review of the GE Nuclear Energy TRACG best-estimate thermal-hydraulic code.

5) ~~5:00~~ - ~~5:15~~ P.M. 4:47 - 5:05 — Subcommittee Report (Open) (REU/AS)
Report by the Chairman of the Plant Systems Subcommittee regarding ABB/CE and Siemens digital I&C applications and insights gained from meeting with the RSK on digital I&C in Germany during November 2000.

5:24 - ~~5:15~~ - **5:30 P.M.** — *****BREAK*****

6) ~~5:30~~ - ~~7:00~~ P.M. 5:32 - 5:45 — Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:
6.1) Issues Associated with Core Power Uprates (MVB/GBW/PAB/AWC)
6.2) Response to the Commission request for a detailed discussion on how the perceived weaknesses with industry-developed thermal-hydraulic codes may adversely affect the NRC's regulatory role and for more specific recommendations on how those weaknesses should be addressed (GBW/PAB)

5:50 Recess

THURSDAY, DECEMBER 7, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) 8:30 - 8:35 A.M. — Opening Statement by the ACRS Chairman (Open) (DAP/SD)

8) 8:35 - ~~9:30~~ 9:35 A.M. — Meeting with NRC Commissioner Diaz (Open) (DAP/AS)
8.1) Remarks by the ACRS Chairman
8.2) Meeting with NRC Commissioner Diaz regarding the NRC Safety Research Program and other items of mutual interest.

9:35 - 9:50 ~~9:30~~ - **9:45 A.M.** — *****BREAK*****

9) ~~9:45~~ 9:50 - 11:45 A.M. — South Texas Project Exemption Request (Open) (JDS/GA/MWW)
9.1) Remarks by the Subcommittee Chairman
9.2) Briefing by and discussions with representatives of the NRC staff and South Texas Project Nuclear Operating Company (STPNOC) regarding the STPNOC's exemption request to exclude certain components from the scope of special treatment requirements in 10 CFR Part 50 and the associated NRC staff's Draft Safety Evaluation Report.

11:50 - 12:50 ~~11:45~~ - ~~12:45~~ **P.M.** — *****LUNCH*****

10) ~~12:45~~ 12:50 - 2:15 P.M. — Control Room Habitability (Open) (TSK/PAB/AS)
10.1) Remarks by the Subcommittee Chairman
10.2) Briefing by and discussions with representatives of the NRC staff and the nuclear industry regarding issues associated with control room habitability and the staff and industry efforts in resolving those issues.

2:45 - 3:00 ~~2:15~~ - ~~2:30~~ **P.M.** — *****BREAK*****

11) 3:00 - 4:30 ~~2:30~~ - ~~4:00~~ P.M. Proposed Final Regulatory Guide DG-1053, "Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Open) (WJS/NFD)

11.1) Remarks by the Subcommittee Chairman

11.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed final Regulatory Guide DG-1053, including the staff's resolution of public comments.

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

4:30-4:45 Break

12) ~~4:00 - 5:00 P.M.~~ Break and Preparation of Draft ACRS Reports (Open)
Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

13) 5:00 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:

13.1) South Texas Project Exemption Request (JDS/GA/MWW) Larkingram

13.2) Control Room Habitability (TSK/PAB/AS)

13.3) Proposed Final Regulatory guide DG-1053 (WJS/NFD)

13.4) Issues Associated with Core Power Uprates (MVB/GBW/PAB/AWC) No Letter

5:00-6:00 13.5) Response to the Commission request for a detailed discussion on how the perceived weaknesses with industry-developed thermal-hydraulic codes may adversely affect the NRC's regulatory role and for more specific recommendations on how those weaknesses should be addressed(GBW/PAB)

13.6) DPO on Steam Generator Tube Integrity (DAP/SD/US) Final Proposed Final Regulatory Guide DG-1053 (WJS/MTM)

6:00 Recess

FRIDAY, DECEMBER 8, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 8:35 A.M. Opening Statement by the ACRS Chairman (Open) (DAP/JTL)

15) 8:35 - 9:40 ~~10:00~~ A.M. Proposed Modifications to the Commission's Safety Goal Policy Statement for Reactors (Open) (GA/MTM)

15.1) Remarks by the Subcommittee Chairman

15.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed modifications to the Commission's Safety Goal Policy Statement for reactors.

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

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

16) 10:15 - 11:30 A.M. NRC Safety Research Program (Open) (DAP/MME)

16.1) Discussion of the draft ACRS report to the Commission on the NRC Safety Research Program.

16.2) Discussion with representatives of the NRC staff, as needed.

11:30 - ~~1:00~~ 1:10 P.M. *****LUNCH*****

17) ~~1:00~~ 1:10 - 1:30 P.M. <u>Future ACRS Activities/Report of the Planning and Procedures Subcommittee</u> (Open) (DAP/JTL/SD)

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

17.2) 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.

18) 1:30 - 1:45 P.M. <u>Reconciliation of ACRS Comments and Recommendations</u> (Open) (DAP, 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.

19) ~~1:45~~ 2:00 - 2:15 P.M. <u>Election of ACRS Officers for CY 2001</u> (Open) (JTL)
Election of a Chairman and Vice Chairman for the ACRS and a Members-at-Large for the Planning and Procedures Subcommittee for CY 2001.

20) ~~2:15 - 3:15 P.M.~~ ~~<u>Break and Preparation of Draft ACRS Reports</u>~~
~~Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.~~

21) ~~3:15~~ 2:30 - 7:00 P.M. <u>Proposed ACRS Reports</u> (Open)
Discussion of proposed ACRS reports on:

2:30-2:35 21.1) Proposed Modification to the Commission's Safety Goal Policy Statement for Reactors (GA/MTM) LarKinsgram Final

2:35-2:40 21.2) South Texas Project Exemption Request (JDS/GA/MWW) LarKinsgram Final

2:50-3:05 21.3) Control Room Habitability (TSK/PAB/AS) Final

2:40-2:50 21.4) Proposed Final Regulatory guide DG-1053 (WJS/NFD) Final

4:30-4:40 21.5) Issues Associated with Core Power Uprates (MVB/GBW/PAB/AWC) No letter

3:30-4:25 21.6) Response to the Commission request for a detailed discussion on how the perceived weaknesses with industry-developed thermal-hydraulic codes may adversely affect the NRC's regulatory role and for more specific recommendations on how those weaknesses should be addressed(GBW/PAB)

21.7) DPO on Steam Generator Tube Integrity (DAP/SD/US) (draft/final)

21.8) Research Report to the Commission (DAP/MME)

SATURDAY, DECEMBER 9, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

22) 8:~~30~~ 32 - 1:00 P.M. Proposed ACRS Reports (Open)
The Committee will continue its discussion and preparation of proposed ACRS reports listed under item 21.

23) 1:00 - 1:30 P.M. Miscellaneous (Open) (DAP/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.**
- **Number of copies of the presentation materials to be provided to the ACRS - 35.**

8:38 DPO Ltr Discussion
9:00 Research
10:00 Break
10:25 Research
10:45 TH Codes (approved)
(DP & GW to edit Appendix)
11:51 Adjourned

APPENDIX III: MEETING ATTENDEES

478TH ACRS MEETING
December 6-9, 2000

NRC STAFF (December 6, 2000)

I. Schoenfeld, OEDO
R. Landry, NRR
A. Cubbage, NRR
T. Ulses, NRR
C. Wu, NRR
S. Wong, NRR
R. Barrett, NRR
T. Kim, NRR
S. Hoffman, NRR
S. Lee, NRR
J. Wermiel, NRR
L. Rossbach, NRR
F. Rubin, NRR
S. Dinsmore, NRR
J. Hannon, NRR
K. Manoly, NRR
C. Carpenter, NRR
S. Bajwa, NRR
J. Strosnider, NRR
K. Karwoski, NRR
Y. Orechwa, NRR
W. Lyon, NRR
B. Bateman, NRR
T. Sullivan, NRR
S. Arndt, RES
J. Muscara, RES
R. Spence, RES
J. Mitchell, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

T. Browning, Alliant Energy-Duane Arnold
C. Willbanks, NUS
M. Knapik, McGraw-Hill
I. Nir, GE
C. Nelson, NMC, Duane Arnold
A. Wyche, SERCH Licensing, Bechtel
C. Brinkman, Westinghouse

NRC STAFF (December 7, 2000)

T. Hsia, OCM/NJD
I. Schoenfeld, OEDO
K. Heck, NRR
T. Alexion, NRR
R. Young, NRR
E. McKenna, NRR
G. Imbro, NRR
S. Richards, NRR
J. Hannon, NRR
D. Fischer, NRR
J. Calvo, NRR
J. Hayes, NRR
M. Blumberg, NRR
M. Hart, NRR
M. Reinhart, NRR
H. Walker, NRR
L. Lois, NRR
J. Nakoski, NRR
R. Barrett, NRR
S. Dinsmore, NRR
J. Williams, NRR
S. Lee, NRR
P. Balmain, NRR
B. Gramm, NRR
J. Fair, NRR
H. Garg, NRR
M. Mitchell, NRR
T. Scarbrough, NRR
D. Thatcher, NRR
S. West, NRR
S. Malik, RES
S. Basu, RES
S. Bahadur, RES
N. Choksi
W. Jones, RES
F. Cherny, RES
M. Federline, RES
P. Lewis, RES
J. Mitchell, RES
N. Kadambi, RES
A. Malliakos, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

R. Grantom, STPNOC
S. Head, STPNOC
G. Schinzel, STPNOC
S. Frantz, ML&B
M. McBurnett, STPNOC
N. Chapman, SERCH/Bechtel
M. Knapik, McGraw-Hill
A. Meyer, NEI
K. Taplett, STPNOC
T. Mscisz, Exelon Nuclear
S. Schultz, Duke Energy
S. Ahmed, Dominion Generation
J. Duffy, PSEG Nuclear
J. Riley, NEI
J. Kloeckner, Dominion Generation
S. Leonard, NMPC
K. Cozens, NEI
M. Ruby, PG&E
C. Nelson, NMC, Duane Arnold
P. Lagus, Lagus Applied Tech. Inc.
R. Campbell, TVA
J. Carew, BNL

NRC STAFF (December 8, 2000)
I. Schoenfeld, OEDO
J. Murphy, RES
J. Mitchell, RES
E. McKenna, NRR

APPENDIX IV: FUTURE AGENDA

The Committee agreed to consider the following during the **XX**th ACRS Meeting, **XX**, 2000:

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
RD ACRS MEETING
, 2000

[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 December 6-9, 2000
2	Items Associated With Core Power Uprates
	2. The Power Uprate Program presentation by NRR [Viewgraphs]
3	Differing Professional Opinion (DPO) on Steam Generator Tube Integrity
	3. Presentation by ACRS Chairman [Viewgraphs]
4	Subcommittee Report, Thermal Hydraulic Phenomena
	4. Subcommittee meeting, November 13-14, 2000, Thermal Hydraulic Phenomena, report of ACRS consultant V. Shrock dated November 28, 2000, report of ACRS consultant N. Zuber dated November 25, 2000 (proprietary information), comments by G. Wallis, "TRACG Model Description," undated (ACRS Internal Use Only) [Handout 4-1]
5	Subcommittee Report, Plant Systems
	5. Report to the ACRS from the Plant Systems Subcommittee Chairman on the Topical Reports for ABB/CE and Siemans Digital Applications presentation by Dr. Robert E. Uhrig [Viewgraphs]
	6. Instrumentation and Control System Failures in Nuclear Power Plants, Robert W. Brill, RES [Handout]
8	Meeting with NRC Commissioner Diaz
	7. Questions on Research [Handout]
9	South Texas Project Exemption Request
	8. South Texas Project Insights into Option 2 and the Proto-Type Pilot Approach presentation by utility representatives [Viewgraphs]
	9. NRC Draft Safety Evaluation, South Texas Project Requested Exemptions from Special Treatment Requirements presentation by J. Nakoski, NRR [Viewgraphs]

10 Control Room Habitability

10. Overview of Revised NEI 99-03 presentation by the Nuclear Energy Institute [Viewgraphs]
11. Uncertainty and Repeatability of Tracer Gas Inleakage Measurements presentation by Peter L. Lagus, Ph.D., Lagus Applied Technology, Inc. [Viewgraphs]

11 Proposed Final Regulatory Guide DG-1053, "Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"

12. Calculation of Neutron Fluence Regulatory Guide presentation by Dr. Nilesh Chokshi, RES [Viewgraphs]
13. DG-1053 Calculation of Neutron Fluence presentation by John Carew, Brookhaven National Laboratory [Viewgraphs]

15 Proposed Modifications to the Commission's Safety Goal Policy Statement for Reactors

14. Modifications to the Safety Goal Policy Statement presentation by J. Murphy, RES [Viewgraphs]

16 NRC Safety Research Program

17 Future ACRS Activities/Report of the Planning and Procedures

15. Future ACRS Activities [Handout #**17-2**]
16. Final Draft Minutes of Planning and Procedures Subcommittee Meeting - **December 5**, 2000 [Handout #**17.2**]

18 Reconciliation of ACRS Comments and Recommendations
No issues.

19 Election of ACRS Officers for CY 2001
Chairman - George Apostolakis
Vice-Chairman - Mario V. Bonaca
Member-at-Large - Thomas S. Kress

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Discussion of Issues Associated with Core Power Uprate
	1. Table of Contents
	2. Presentation Schedule
	3. Project Status Report dated December 6, 2000
	4. "Potential Synergistic Effects of Industry Initiatives to Extend Plant Life, Increase Production, and Reduce Regulatory Burden" (Revision 3), M. Bonaca, ACRS, dated September 12, 2000
	5. Memorandum to ACRS Members and Staff, from G. Cronenberg, ACRS Sr. Fellow, Subject: Central Issues Related to Power Uprate Reviews
	6. Report to G. Dicus, Chairman, NRC, from D. Powers, Chairman, ACRS, Subject: Draft Commission Paper Regarding Proposed Guidelines for Applying Risk-Informed Decisionmaking in License Amendment Reviews, dated October 8, 1999
	7. Report to the Commission from W. Travers, EDO, Subject: Proposed Guidelines for Applying Risk-Informed Decisionmaking in License Amendment Reviews, SECY-99-246, dated October 12, 1999
9	South Texas Project Exemption Requests from Special Treatments
	8. Table of Contents
	9. Proposed Schedule
	10. Status Report
	11. Letter dated November 15, 2000, from J. Zwolinski, NRR, to W. Cottle, STP Nuclear Operating Company, Subject: South Texas Project, Units 1 and 2-Draft Safety Evaluation on Exemption Requests from Special Treatment Requirements of 10 CFR Parts 21, 50, and 100
	12. Memorandum dated August 17, 2000, from D. Mathews, NRR, to J. Larkins, ACRS, Subject: Risk-Informing Special Treatment Requirements
	13. Meeting Summary, dated August 7, 2000, Subject: Treatment Issues on the Multipart Exemption from Special Treatment Rules of 10 CFR Part 50
	14. Letter dated July 13, 1999, from J. Sheppard, Vice President, Nuclear Operating Company, South Texas Project, Subject: Request for Exemption to Exclude Certain Component from the Scope of Special Treatment Requirements Required by Regulations.
10	Resolution of Issues Associated with Control Room Habitability
	15. Table of Contents
	16. Presentation Schedule
	17. Project Status Report dated December 7, 2000
	18. Excerpt from Minutes of November 15, 2000, Severe Accident Management

Subcommittee meeting, NRC/NEI Report, NEI 99-03, Control Room Habitability Assessment Guidance (+5)

19. Letter to D. J. Modeen, NEI, from R. J. Barrett, NRR, Subject: Control Room Habitability, dated November 13, 2000 (+1)
20. E-mail from M. Bonaca, ACRS, to P. Boehnert, ACRS, Subject: Forward, Control Room Habitability (+7)

11 <u>DG-1053 – Calculation of Neutron Fluence</u>

21. Table of Contents
22. Proposed Schedule
23. Status Report dated December 7, 2000
24. Letter from J. Ernest Wilkins, Jr. Chairman ACRS, to James M. Taylor, EDO, NRC, Subject: Proposed Draft Regulatory Guides DG-1023, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 FT-LB," and DG-1025, "Calculational and Dosimetry Methods for Determing Pressure Vessel Neutron Fluence," dated July 15, 1993
25. Memorandum from Nicholas Tsoulfanidis, University of Missouri-Rolla, to J. Larkins, Executive Director, ACRS, Subject: Draft Regulatory Guide DG-1053 Determination of Neutron Fluence at the Pressure Vessel of a LWR, dated November 27, 2000
26. Regulatory Analysis from Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," received October 23, 2000 [prepared for internal committee use]
27. Table 1, "Summary of Regulatory Positions on Calculation and Dosimetry from Draft Regulatory Analysis from DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," received October 23, 2000 [prepared for internal committee use]

15 <u>Proposed Revision to the Commission's Safety Goal Policy Statement for Reactors</u>

28. Table of Contents
29. Proposed Schedule
30. Status Report dated December 8, 2000
31. Memorandum dated November 14, 2000, from Ashok C. Thadani, RES, to J. T. Larkins, Executive Director, ACRS, Subject: Safety Goal Policy Statement (pre-decisional, draft provided for ACRS use only)
32. Letter dated April 7, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Reactor Safety Goal Policy Statement
33. Memorandum dated June 27, 2000, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, EDO, NRC, Subject: Staff Requirements - SECY-00-0077 - Modifications to the Reactor Safety Goal Policy Statement

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

December 19, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
479TH ACRS MEETING
FEBRUARY 1-3, 2001

THURSDAY, FEBRUARY 1, 2001, 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)
   - 1.2) Items of current interest (GEA/SD)
   - 1.3) Priorities for preparation of ACRS reports (GEA/JTL/SD)

2) 8:35 - 10:15 A.M. Treatment of Uncertainties in the Elements of the PTS Technical Basis Reevaluation Project (Open) (WJS/GBW/MTM)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding treatment of uncertainties in the elements of the Pressurized Thermal Shock (PTS) Reevaluation Project.

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

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

3) 10:30 - 12:00 Noon Siemens S-RELAP5 Appendix K Small-Break LOCA Code (Open/Closed) (GBW/PAB)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff and Siemens Power Corporation regarding the Siemens S-RELAP5 Appendix K Small-Break Loss-of-Coolant Accident (LOCA) Code and the associated NRC staff Safety Evaluation Report.

   NOTE: A portion of this session may be closed to discuss Siemens Power Corporation proprietary information applicable to this matter.

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

4) 1:00 - 2:30 P.M.. Proposed ANS Standard on External-Events PRA (Open) (GEA/MTM)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the American Nuclear Society (ANS) regarding the proposed ANS Standard on external-events PRA.

Representatives of the NRC staff will provide their views, as appropriate.

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

5) 2:45 - 4:00 P.M. Reprioritization of Generic Safety Issue-152, "Design Basis for Valves that Might be Subjected to Significant Blowdown Loads" (GML/AS)

5.1) Remarks by the Acting Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding reprioritization of Generic Safety Issue-152 and the reasons therefor, and related matters.

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

6) 4:00 - 5:00 P.M. Break and Preparation of Draft ACRS Reports (Open)
Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

7) 5:00 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:

7.1) Treatment of Uncertainties in the Elements of the PTS Technical Basis Reevaluation Project (WJS/GBW/MTM)

7.2) Siemens S-RELAP5 Appendix K Small-Break LOCA Code (GBW/PAB)

7.3) Proposed ANS Standard on External-Events PRA (GEA/MTM)

7.4) Reprioritization of Generic Safety Issue-152 (GML/AS)

FRIDAY, FEBRUARY 2, 2001, 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. Regulatory Effectiveness of the ATWS Rule (Open) (TSK/MWW)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's assessment of the regulatory effectiveness of the Anticipated Transients Without Scram (ATWS) Rule (10 CFR 50.62).

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

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

10) 10:15 - 11:45 A.M. Overview of Mixed Oxide Fuel Fabrication Facility (Open) (JDS/AS)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the Department of Energy (DOE) and the NRC staff regarding the proposed Mixed Oxide Fuel Fabrication Facility to be constructed at the DOE's Savannah River Plant site.

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

11) 1:00 - 2:00 P.M. Meeting with the NRC Chairman (GEA/JTL)

11.1) Remarks by the ACRS Chairman

11.2) Meeting with the NRC Chairman Meserve to discuss items of mutual interest.

2:00 - 2:15 P.M. ***BREAK***

12) 2:15 - 3:15 P.M. NRC Safety Research Program (Open) (DAP/MME)

12.1) Remarks by the Subcommittee Chairman

12.2) Discussion of the annual ACRS report to the Commission on the NRC Safety Research Program.

Representatives of the NRC staff will provide their views, as appropriate.

13) 3:15 - 3:45 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL/JEL)

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

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

14) 3:45 - 4:00 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.

15) 4:00 - 5:00 P.M. Break and Preparation of Draft ACRS Reports

Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

16) 5:00 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:
16.1) Regulatory Effectiveness of the ATWS Rule (TSK/MWW)
16.2) Reprioritization of Generic Safety Issue-152 (GML/AS)
16.3) Proposed ANS Standard on External-Events PRA (GEA/MTM)
16.4) Siemens S-RELAP5 Appendix K Small-Break LOCA Code (GBW/PAB)
16.5) Treatment of Uncertainties in the Elements of the PTS Technical Basis Reevaluation Project (WJS/GBW/MTM)
16.6) NRC Safety Research Program Report (DAP/MME)

SATURDAY, FEBRUARY 3, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

17) 8:30 - 12:30 P.M. Proposed ACRS Reports (Open)
(10:15-10:30 A.M.-BREAK) Continue discussion of proposed ACRS reports listed under Item 16.

18) 12:30 - 1:00 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.**
- **Number of copies of the presentation materials to be provided to the ACRS - 35.**

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

DECEMBER 6-9, 2000
Date(s)

DECEMBER 6, 2000
Today's Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	BADGE #	NRC ORGANIZATION
FAROUK ELTAWILA	A6364	RES/DSARE
RALPH LANDRY	B-6634	NRR/DSSA/SRXB
Amy Cubbage	B-6017	NRR/DSSA/SRXB
Tony WILSES	A-7611	NRR/DSSA/SRXB
Cheng-H. Wu	B-7288	NRR/DE/EMEB
See-Meng Wong	B-8181	NRR/DSSA/SPSB
RICHARD BARRETT	A-7330	NRR/DSSA/SPSB
T.J. Kim	B-8098	NRR/DLPM
Steve Hoffman	B6256	NRR/DRIP/RLSB
Samuel Lee	B-7501	NRR/DSSA/SPSB
Jared Wermiel	A-6824	NRR/DSSA/SRXB
Larry Rossbach	B-8536	NRR/DLPM
Mark Rubin	B7052	NRR/DSSA/SPSB
Stephen Dinsmore	B7898	NRR/SPSB
John Hannon	A-6149	NRR/DSSA
KAMAL MANOLY	B-8035	NRR/DE/EMEB
C.E. CARPENTER	B6462	NRC/DE/EMCB
S. SINGH BAJWA	TB-6020	NRC/DLPM/PDIII
JACK STROSNIDER	A 6986	NRC/NRR/DE
Ken Karwoski	A-7510	NRC/NRR/DE/EMCB
Yuri Orechwa	B-8716	NRR/DSSA/SRXB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

DECEMBER 6-9, 2000
Date(s)

DECEMBER 6, 2000
Today's Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	BADGE #	NRC ORGANIZATION
Diablo Schoenfeld	6983	DEDO
WARREN LYON	B8535	NRR/DSSA/SRXB
Bill Bateman	A6043	NRR/DE/EMCB
Ted Sullivan	A7059	"
Steven Arndt	B8390	RES
J. MUSCARA	B6836	RES/DET/MEB
R. Spence	B-8421	RES/DSARE/REAHFB
Jocelyn Mitchell	B6685	RES

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478TH FULL COMMITTEE MEETING

DECEMBER 6, 2000
Today's Date

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Charles Willbanks	NUS Info Services
MICHAEL KNAPIK	McGRAW-HILL
Israel Nir	GE
Chuck Nelson	NMC Duane Arnold
Altheia Wyche	SERCH Licensing / Bechtel
Charles Brinkman	Westinghouse

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

DECEMBER 6-9, 2000
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
478TH FULL COMMITTEE MEETING

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ITEMS OF INTEREST

478th ACRS MEETING

DECEMBER 6-9, 2000

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
478th MEETING
DECEMBER 6-9, 2000

NRC NEWS

UNITED STATES NUCLEAR REGULATORY COMMISSION

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Excellence in Nuclear Safety in Today's Regulatory Environment
A Status Report and a Look Toward the Future at the NRC

Remarks by Chairman Richard A. Meserve
U.S. Nuclear Regulatory Commission

at the Institute for Nuclear Power Operations Conference
Atlanta, GA, November 2, 2000

Some of you may recall that I spoke briefly with you at last year's conference. At the time, I had been the Chairman for less than one week and that meeting was my first opportunity to interact with many of you. I would like to spend a moment reflecting on the events of the past year.

As all of you are aware, this year has been a period of remarkable change. We are today in a period of transition in several dimensions, probably experiencing more rapid change than in any other period in the history of civilian nuclear power, certainly since Three Mile Island. As economic deregulation of electric utilities proceeds, we are seeing significant restructuring among our licensees and the start of the consolidation of nuclear generating capacity among a smaller group of operating companies. This has no doubt brought significant alterations to the lives of many of you here today.

Even more striking than industry consolidation is the changing attitude, principally in the business world, toward nuclear power. Only a short time ago, pundits claimed that the deregulation of electricity prices would result in the premature shutdown of many nuclear plants. Now, in striking contrast, attention is focused on reactor license renewal. In fact, we now expect that as much as 85 percent of the current fleet will be the subject of applications for license renewal. If these are successful, nuclear energy will contribute to our Nation's energy security well into this century. In the last few months, there has even been the first stirring of interest in the possibility of new construction in the U.S. In short, in the period of a single year, we have seen a remarkable change in the attitude toward and future prospects for nuclear power in the United States.

The credit for this change must go to an industry that has achieved remarkable gains in both economic and safety performance over the past decade and thus could seize the opportunity presented by electricity price deregulation. I hope nonetheless that the NRC has helped to set the stage by its efforts to establish a regulatory system that is technically sound, that is fair, that is predictable, and that reaches decisions with reasonable dispatch.

As I reflect on my first year on the Commission, I have several impressions that I would like to share with you. First, I am very impressed by the dedication of the industry -- and our other stakeholders -- not only to ensuring safety, but also to helping to improve the NRC's ability to achieve its mission efficiently and effectively. You and other stakeholders have dedicated time, energy, and resources in a most productive and helpful way in pursuit of our common goal of assuring public health and safety. My colleagues on the Commission and I appreciate your efforts.

Second, I am very impressed by the skills, professionalism and commitment of the NRC staff. It is clear to me that the staff is an under-appreciated national resource. Strong efforts to maintain and enhance its competence are warranted. Our capacity to provide a regulatory system that serves both our licensees and the public interest is dependent on the skills and dedication of our staff.

Third, I am very impressed by my colleagues on the Commission. I find that all of my fellow Commissioners are hard-working and knowledgeable. They explore the issues in a conscientious effort to ensure that our decisions are sound. Moreover, I believe that we are working well together and respect each other's views. I am sure that this synergy results in Commission decisions that are better than those that

any one of us could have reached alone. I am thus particularly pleased that two of my colleagues, Nils Diaz and Jeffrey Merrifield, are also attending this conference.

Nonetheless, I believe that all of us -- the nuclear industry, the staff, the Commission -- will be severely challenged in the years ahead. Today, I will try to characterize the current environment, indicating along the way the status of our progress in important areas of interest, and to frame several issues that I believe are critical for the maintenance of the NRC's capacities to work with you to assure nuclear safety into the future.

The Current Landscape

Let me start by surveying some of the current landscape.

License Transfers

One of the more immediate results of the economic deregulation of the electric power industry has been the development of a market for nuclear power plants as capital assets themselves. As a result, the Commission has seen a significant increase in the number of requests for approval of license transfers. These requests increased from an historical average of about two or three per year, to an average of 15 in calender year 1998 and 1999. So far this year, the staff has reviewed and approved six applications involving a total of 16 reactors.

The NRC is working hard to review license transfer applications efficiently. These reviews sometimes require a significant expenditure of talent and energy by our staff to insure a high quality and timely result. So far, I believe that our record is very good. For example, in CY 2000, we have reviewed and approved transfers in periods ranging from four to eight months, depending on the complexity of the applications. The Commission will strive to continue to perform at this level of proficiency even in the face of the anticipated increased demand.

License Renewals

Another result of the new economic conditions is an increasing interest in plant life extension beyond the original 40-year term. That term, which was established in the Atomic Energy Act, did not reflect a limitation that was determined by engineering or scientific considerations, but rather was based on financial and anti-trust concerns. We now have experience on which to base judgments about the potential useful life and safe operation of facilities and have been turning to the question of extensions beyond the original 40-year term.

The focus of our review of applications is on maintaining plant safety, with the primary concern directed at the effects of aging on important systems, structures, and components. Applicants must demonstrate that they have identified and can manage the effects of aging so as to maintain an acceptable level of safety during the period of extended operation.

We have now renewed the licenses of two plants, Calvert Cliffs and Oconee, comprising a total of 5 units.[1] The reviews of these licenses were completed ahead of schedule, which is testament to the care exercised by licensees in the preparation of the applications and the dedication of staff. Applications from three additional plants -- Hatch, ANO-1, and Turkey Point -- are currently under review. And, as I indicated earlier, we anticipate many more applications for renewal in the coming years.

Although we have met our projected schedules for the first reviews, we would like the renewal process to become as effective and efficient as possible. The extent to which are able to sustain or improve on our performance depends on the rate at which applications are actually received, the quality of the applications, and our ability to staff the review effort. We recognize the importance of license renewal and are committed to provide high-priority attention to this effort. As you know, we encourage licensees to notify us well in advance of their intentions to seek renewals and to establish a queue for applications so as not to create unmanageable demands on staff resources.

Risk-Informed Regulation and Oversight

We also are in a period of dynamic change on the safety regulatory front as we move from a prescriptive, deterministic approach to a risk-informed and performance-based paradigm. This initiative is extraordinarily ambitious, because it involves both rethinking the entire regulatory structure and changing attitudes. Nonetheless, we need to make this change to take advantage of what we have learned over the past 40 years, employing that knowledge both to focus on concerns that are truly significant for safety and to reduce needless regulatory burdens. I am pleased to acknowledge the support and active participation of our stakeholders in this effort.

Perhaps the most visible aspect of our efforts to risk-inform our regulatory system is our new reactor oversight process. The process was initiated on a pilot basis last year and fully implemented this past April. The new process was developed to focus our inspection effort on those areas involving greater risk to the plant and thus to workers and the public, while simultaneously providing a more objective and transparent process. Although we continue to work with our stakeholders to assess the effectiveness of the revised oversight process, the feedback we are receiving from the industry and the public is quite favorable.

Nonetheless, the Commission is committed to continued improvement of the overall process. An example of this work is the effort to improve Performance Indicators (PIs). The NRC staff and an industry group are working on revising two PIs that deal with scrams so as to ensure that concern about PIs does not cause operators to hesitate to take necessary actions. Similarly, we will review the PI addressing unplanned power changes to reduce the potential for unintended consequences. Finally the review process has uncovered inconsistencies in the ways in which the "unavailability" of safety systems is assessed in the Maintenance Rule, the Performance Indicator, and the counterpart WANO indicator. We are working with an industry group organized by NEI to address these problems.

I am encouraged by the success of the oversight effort, and believe that we have implemented a program that is appropriately focused to support our overall performance objectives. However, we have established a review panel, including both NRC staff and external stakeholders, to evaluate the initial implementation. We see the oversight program as a work-in-progress, and I fully expect that it will continue to change and evolve as our experience increases.

Other efforts to improve the substance of our regulations using risk insights are also underway. We have proceeded thus far along two tracks that eventually will converge. The first track involves a focus on selected activities for which the opportunities are clear. For example, consideration of risk is used to modify the allowed outage times in technical specifications for safety-related equipment. Similar approaches are used in other areas, such as quality assurance and in-service inspection and testing. We also have extended this approach to modifications of existing regulations. For example, the rule governing plant changes and testing, 10 CFR 50.59, was modified to provide for greater flexibility when the risk consequences are minimal. The Maintenance Rule, 10 CFR 50.65, also was modified to require assessment of the risk associated with maintenance activities.

The second track involves a more comprehensive and systematic examination of the body of reactor regulations in 10 CFR Part 50. For example, we have launched an examination of special treatment requirements. With the insights provided by probabilistic risk assessments, we now realize that some equipment that has been categorized as safety-related, and thus subjected to special restrictions, in fact can be shown to have limited contribution to risk. Conversely, other equipment that was not previously categorized as safety-related is now understood to have safety significance. We are engaged in an extensive process to rethink the regulatory requirements that bear on these categories of equipment. The other major effort includes the examination of the technical requirements in 10 CFR Part 50. The pilot rule for this effort is 10 CFR 50.44, standards for combustible gas control systems. The next rule to be to included in this effort is 10 CFR 50.46, which delineates the acceptance criteria for Emergency Core Cooling Systems. This latter rulemaking presents a considerable challenge since conceptually it would allow fundamental changes to the design basis and safety analysis for light-water reactors. Other risk-informed initiatives are underway in the areas of fire protection and reactor pressure vessel integrity.

It is completely clear that we have taken only the first steps in our efforts to risk-inform our regulatory system. In a series of studies conducted in 1993, the National Academy of Engineering found that it took the Environmental Protection Agency an average of 15 years to modify its standards after scientific consensus had been reached on a better understanding of the underlying risks. I intend for the NRC to do better. Nonetheless, we have a long and difficult road ahead of us for which we will require informed input from the industry and other stakeholders.

Security and Safeguards

An area of continuing importance is the security and safeguards of nuclear facilities and materials. Physical protection against the threat of radiological sabotage or theft of nuclear material is a fundamental obligation of all licensees. To understand the significance of maintaining a robust security program, we need only consider the ramifications, both locally and worldwide, that would result from the success of a saboteur who directs his efforts toward a nuclear facility.

I am aware that the NRC's policy on security matters has not been transparent and that we have not been consistent in our requirements. Although the design-basis threat defined in our regulations has been fairly stable over time, in the past the adversary characteristics that define the details were revealed to licensees only in the context of a drill using mock adversaries -- an Operational Safeguards Response Evaluation (OSRE). We now recognize that the adversary characteristics utilized during OSREs have varied over time and from site to site. In short, we have not had a disciplined process to define the adversary characteristics and we have not clearly and consistently communicated our expectations.

The program for conducting OSREs has been improved. For example, the staff has developed and shared with the industry the specific list of characteristics of adversaries that form the basis for these exercises. Based on early feedback, we believe that we have taken an important first step in both clarifying our expectations and improving communications in this area. To improve the process further, NRC staff has been working diligently with its stakeholders to enable the agency's endorsement of an acceptable Safeguards Performance Assessment Program, which could eventually replace the OSRE program.

The dynamic nature of potential threats means, however, that ensuring nuclear plant security and safeguards requires continuing examination. As a result, the Commission is assessing our fundamental policies in the area of security requirements. For example, the Commission is working with the staff in developing a process for the systematic evaluation of the design basis threat and the adversary characteristics to which our licensees are expected to respond. I expect that the Commission will devote considerable effort in this area in the future.

Funding

I could discuss many other current activities, but I hope that this brief tour of some of the highlights has given you a sense of a Commission that is working hard and effectively to address important issues. Before leaving this survey of our current activities, however, I want to say a few words about funding.

I know that our licensees have a legitimate and direct interest in our expenditures because, except for certain activities supported from the Nuclear Waste fund, the NRC is a fee-based agency. I understand, appreciate, and accept the general principle that caused Congress to adopt this approach - namely, that licensees, who directly benefit from our work, should pay for it. However, I also agree with the corollary -- namely, that licensees should not be required to pay for activities that do not directly benefit them. As a result, my colleagues and I have been working to modify budgetary policy regarding the NRC's status as a fee-based agency. Our goal is to ensure that an appropriate portion of the agency's budget is funded from general revenues rather than fees. The FY01 Energy and Water Development Appropriations Act, recently signed by the President, provides for phasing in general revenue funding, eventually to reach 10 percent of the total NRC budget. I know that the industry has strongly supported federal funding of at least some of NRC's budget and I appreciate that, through our mutual efforts, we have

been able to achieve this important goal.

Looking to the Future

Let me now shift gears. The topics I have just been addressing represent, I believe, various arenas in which the NRC and industry are both working hard and are on the right track. There is much still to do, but the goals are clear and achieving them is within NRC's authority and ability. Our stakeholders are vitally interested in progress in these areas and, as a general rule, support our efforts.

I want to turn now to several issues for which the path is somewhat less clear. My purpose today is to frame the issues in the hope that we can start a dialog that will lead to consensus on the nature of the problems and on the avenues most likely to lead to progress toward solutions.

Maintaining NRC's Core Competence

First, let me address the need to ensure NRC's core technical competence. I believe that it is in both the public interest and the regulated industries' interest that the NRC is able to reach sound technical judgments in an efficient manner. To be able to respond to changing environments - not just in the nuclear power industry, but in other civilian uses of radioactive materials, such as in nuclear medicine - the NRC has to be both knowledgeable and agile.

Most of your operations are subject to NRC regulations and, in turn, depend on our ability to write technically sound, risk-informed rules; to make sound licensing decisions without undue delay; and to conduct fair, meaningful oversight. The efficiency and effectiveness of the implementation of our regulations, of our licensing reviews, and of our oversight and inspections are a direct reflection of the core technical capabilities of our staff. Moreover, the Commission relies on the sound, independent technical judgment of the NRC staff. As our experience with risk-informing our regulatory processes shows, the ability of the agency to address new and emerging issues depends on a staff that can understand, analyze, and use scientific and technical information at the cutting edge.

Finally, the staff's reputation for technical competence is a crucial element in building public confidence and trust. In order to establish this confidence, we must be -- and must be perceived to be -- scientifically and technically knowledgeable and able. This fact is reinforced when I consider the varying degrees of public trust that are afforded federal regulatory agencies concerned with public health and safety -- such as the EPA, the Food and Drug Administration, or the Consumer Product Safety Commission. Each agency strives to have clear principles, fair processes, and strategies for open communications. What distinguishes one from another on the scale of public trust, I believe, is each agency's scientific and technical reputation. I am therefore convinced that for NRC to continue to be effective and efficient into the future, we must ensure the agency's core competence.

Why am I concerned? Two demographic trends raise questions about our future. One trend reflects the aging of the NRC staff. The ratio of NRC employees who are over 60 years of age to those under 30 is 6:1. The same ratio at NASA, for comparison, is 2:1. Moreover, seventeen percent of NRC's engineers are already eligible for retirement and another four percent of the current workforce of engineers will become eligible for retirement each year for the next few years. Twenty-five percent of the employees in the Office of Nuclear Regulatory Research and twenty percent of the employees in the Office of Nuclear Reactor Regulation are eligible for retirement today.

Despite our efforts to hire new engineers, we have experienced a net loss of engineers over the past five years. That loss is equivalent to roughly eight percent of our engineering workforce. The bottom line is that we are losing expertise and, along with it, valuable institutional knowledge.

The demographics of our workforce are the result of several intersecting factors. First, we have experienced declining real budgets over a number of years (until the slight upturn in FY2001). We have lost technically skilled personnel not only because of reductions in Full Time Equivalents (FTEs) in the budget process, but also because budgetary retrenchment adversely affects morale. And the government is increasingly challenged in recruiting the best and brightest.

The second trend, not unrelated to the first, is in the supply of new graduates from nuclear engineering programs. Student interest in nuclear engineering declined sharply during the 1990s. In fact, by the late 1990s undergraduate enrollments in nuclear engineering programs were only about 40 percent of the average for the second half of the 1980s. Similarly, undergraduate degrees in the field were only two-thirds the average number for that reference period.[2] Another recent study indicated that the current annual average supply of nuclear engineers with B.S. and M.S. degrees is about 160 new graduates, whereas the annual demand for new engineers is estimated to be at least 300 and possibly as high as 600.[3]

The combination of these long-term trends raises a red flag: how will NRC be able to maintain its core technical competence into the future? We need to plan for turnover and retirements, as any employer would, but we also need to judge carefully what expertise we must have among our employees. I recently asked our Executive Director for Operations (EDO) to begin the process of developing such a plan. I am raising this issue with you because the current situation deserves careful attention and you should be aware of it.

Research in the NRC

My second concern relates to the support of research at the NRC. This subject is important because such research is both central to our regulatory functions and vital for maintaining core technical competence.

In the parlance of science policy, there are at least two kinds of research. One is described as curiosity-driven, inquiry-based fundamental research. Its objective is to advance scientific or technical understanding without specific applications in mind. The National Science

Foundation, for example, funds this kind of research.

The other type of research is need-driven, problem-solving research. It has specific applications in mind. This work is intended to be very practical -- in industry it might be called applied research and development. Government supports such research for a number of reasons. For example, for years the government has funded research on both coal and nuclear technologies to promote the development of those industries in the national interest because individual firms do not have the financial resources for that purpose.

The NRC undertakes the second type of research. Such need-driven work is undertaken to support the independent evaluations of safety that we must provide. Better understanding of - and reduced uncertainties about - risk and safety margins are prerequisite to both enhancing safety and reducing unwarranted conservatism in our regulations. I believe that some of our licensees may not recognize the benefits that our past research effort has provided. Let me give a few examples that reflect the fruits from our past efforts:

- the revised source term, which has the potential for substantial cost savings for licensees;
- license renewal, for which the results of the NRC's nuclear plant aging research program have proven indispensable;
- new guidelines on burnup credit in spent fuel storage, which can reduce the number of casks needed for storage or transportation;
- and, of course, risk-informed regulation, with its promise for both improving safety and reducing unnecessary burden, which owes its existence to AEC- and NRC-sponsored research in probabilistic risk assessment.

Similarly, our current research efforts will provide the foundation for many other on-going initiatives, including:

- revisions to the pressurized thermal shock rule;
- review of advanced digital instrumentation and control systems;
- utilization of fuel to higher burnups;
- and advanced, best-estimate thermal-hydraulic analysis codes, which provide the technical bases for evaluation of power uprates, longer operating cycles, advanced fuel designs, and other similar industry activities.

Given these accomplishments and our continuing efforts, you may well ask why I have mentioned concern about the research program. My concern arises from the fact that our research program has been subject to a steady decline in funding and scope for nearly two decades. The NRC's funding of research activities declined from nearly $200 million in 1981 to roughly $43 million today -- without any adjustment for the corrosive effects of inflation. The striking magnitude of the change - and the commensurate impact that the reductions have had on the relevant staff - raises the question as to whether we are adequately supporting the research that will enable the Commission to respond to future regulatory needs.

Recently our Office of Nuclear Regulatory Research organized a panel of experts from our stakeholder communities to provide it with insights about the role, direction and scope of research in NRC. I have read the draft report of the first phase of this activity and found many thoughtful and helpful comments in it. We will also benefit later this year with a report from the Advisory Committee on Reactor Safeguards on this subject. Guided by these reviews, it is my aim, with the help of my fellow Commissioners, to ensure that we have a research program that is sufficient to our needs and that is appropriate in scope.

In this connection, let me hasten to add that we are conscious of the need to be cost-efficient in all of our programs, including the research program. We recognize that we must explore further means to collaborate with industry, with DOE, and with our international partners in a way that conserves resources, but does not compromise our independence. I would like to strengthen these collaborations not only because more research can be conducted through cooperation, but also because we need these interactions to ensure that we benefit from other perspectives in assessing priorities.

It is our aim to ensure that our research effort is adequate to the challenges that confront us and that we are allocating the right resources, undertaking the right work, and employing the right research performers. In short, we want to ensure that our research program is technically sound, is efficient, and is agile. My colleagues on the Commission and I welcome your suggestions and advice.

New Reactors

Let me turn to another area in which developments over the past months have raised an emerging challenge for the agency. I started this talk with a reference to reports that new reactors may be constructed in the United States in the not too distant future. The NRC must watch these developments so that our processes do not serve as a needless impediment.

The NRC has certified three standardized plant designs and has recently begun a review of Westinghouse AP1000 design for possible certification. Moreover, we have been following DOE's work on new reactor approaches so that we can develop familiarity with potential advanced reactor concepts that may be of commercial interest. Nonetheless, there are many challenges for the NRC if the early discussions of construction result in real projects. It has been many years since our staff has had regulatory responsibility for a reactor construction project and old skills will have to be revitalized. Moreover, our current reactor regulations may not translate well to the licensing of new reactor designs, particularly if the new designs are not water-cooled. Our efforts to risk-inform our regulations may be somewhat helpful in the certification or licensing of new designs, but in some cases, the best approach may well be to start with a clean sheet of paper.

My colleagues and I are following developments in this area with great interest because, if construction is to commence, we will have to prepare for it.

International

One other challenge that I must mention is a continuing one, and one that is not directly subject to NRC control. The hard reality is that, like it or not, our Nation's nuclear program is interconnected with and dependent on nuclear activities elsewhere. The incident last year at Tokaimura reminds us that a nuclear-related event anywhere in the world will cause heightened concern about nuclear enterprises everywhere. A serious accident might dissolve the emerging optimism about nuclear power that is developing in the U.S.

This vulnerability reinforces the need for NRC to continue to work with our counterparts abroad to advance nuclear safety throughout the world. We benefit not merely because domestic nuclear enterprises are linked in the public consciousness with activities elsewhere, but also because we all gain from sharing experiences and insights with our colleagues in other countries. In helping others, we help ourselves.

My fellow Commissioners and I have sought through our international interactions to advance the cause of global nuclear safety. Indeed, my colleague Greta Dicus is not here today because she is representing the Commission at the Pacific Basin Conference in South Korea. I can offer no easy solutions to the risk that affects our licensees as a result of events that may occur abroad. But the Commission is seeking to do what it can to address safety through cooperation with our regulatory counterparts around the globe. I urge our licensees to use their international contacts to pursue the same goal.

Public Confidence

Let me turn to a final issue that must be an abiding concern for us all. Perhaps the key issue for the future of the nuclear industry is the establishment and maintenance of public confidence that nuclear energy is acceptably safe. You, through your actions, obviously must play the central role: you have to demonstrate excellence in nuclear safety in every hour of every day.

I recognize, however, that the NRC also has important obligations to seek to maintain public confidence as well. We must be, and be seen, as a rigorous, independent, and capable regulator. In this connection, I have previously mentioned the need to maintain the core competence of the staff. Moreover, in order to maintain public confidence, we must provide open processes so that the public has the opportunity to raise concerns and to observe that those concerns have been weighed and fairly evaluated. We cannot hope that everyone will agree with our decisions, but we can aspire to show that no legitimate concern has been ignored.

The significance that the Commission places on its obligations in this regard is reflected in the fact that we have identified public confidence as one of our four performance goals in our strategic planning. We have sought wide participation in our meetings, encouraged the staff's efforts to hold public meetings in affected communities, and sought to harness the Internet, including even web-casting our meetings, as a vehicle for providing the public with the opportunity to observe and participate in our processes.

I mention public confidence as a future challenge because continuing efforts are required. The renewed interest in nuclear energy that is now emerging will not be sustained without public confidence. A continuing willingness to engage the public, and the search for new means to facilitate that engagement, will be a continuing task for us all.

Conclusion

Let me now conclude by returning to where I started. We are in a period of remarkable change and challenge and I hope that my talk has revealed some of the many dimensions by which we all are affected. I believe we are at a moment of great opportunity.

It is human nature to seek to resist change and organizational settings only exacerbate that tendency. But those of us here in this room have the responsibility to embrace change, to manage change, and to prepare for the future. We should seek to encourage the attitude among our colleagues and the public that change offers opportunities for improvement.

That means that we -- all of us -- must accept the responsibility not only of maintaining our institutional capacities to meet current needs, but also of building the capabilities to meet the changing needs of a dynamic environment. My colleagues and I on the Nuclear Regulatory Commission take this responsibility very seriously. The NRC not only must be effective and efficient as a regulator, but also must respond to the changes in the communities that it regulates. We recognize this obligation and are committed to meeting it.

Thank you again for the opportunity to share my views with you. I will be happy to take questions and comments.

[NRC Home Page | News and Information | E-mail]

1. The review of the Calvert Cliffs application for renewal took 23 months; that for Oconee, 22 months.

2. Friedberg, J.P. (1999). Nuclear engineering in transition: a vision for the 21st century. In Nuclear News, June, p.50.

3. Was, G.S., T. Quinn, and D. Miller (1999) Manpower Supply and Demand in the Nuclear Industry. Presented at the American Nuclear Society 1999 Winter Meeting. Long Beach, CA.

NRC NEWS

UNITED STATES NUCLEAR REGULATORY COMMISSION

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Nuclear Safety and Public Acceptance in the United States

Remarks by Dr. Richard A. Meserve
Chairman, U.S. Nuclear Regulatory Commission
2000 ANS/ENS International Meeting
November 13, 2000

Good morning. I am pleased to be able to participate in this session, and to provide my perspective as a regulator on the question of nuclear technology's role in supporting sustainable development. I will place particular emphasis on safety and public acceptance issues.

Introduction

Demographers estimate that the world's population will reach 9 billion in 2050 and will level off at 10 to 11 billion by the end of the century -- nearly double the current world population of 6 billion. Most of the growth is projected in Africa, Asia, and Latin American -- regions which are already stressed by the need to reduce hunger, disease, and poverty. Moreover, the percentage of people living in urban areas is expected to grow from 50 percent today to 80 percent in 2050. These figures imply, astonishingly, the creation of the equivalent of 400 new urban complexes, each of 10 million people, over the next 50 years.

Meeting the needs of a growing population will result in greater production and consumption of goods and services, intense pressures on the environment and on living resources, and increased demand for land, water, materials and energy. The challenge is made all the more difficult by the fact that we must meet these growing demands while somehow nurturing and restoring the planet's already overtaxed life-support systems.

The signs of serious strains are already apparent: the buildup of greenhouse gases in the atmosphere, the decline of marine fisheries, increasing regional shortfalls in the quality and quantity of fresh water, expanding tropical deforestation, the continuing loss of both species and sensitive ecosystems, and the emergence of serious infectious disease. Somehow we must reconcile mankind's developmental aspirations with the limitations imposed by nature. The challenge we confront -- indeed, the imperative we confront -- is to recognize the interdependence of societal demands and environmental limitations and to find a path to sustainable development.

The focus of this conference is energy -- an essential ingredient for the production of goods and services, enhanced life style, and improved environmental performance. I emphasize environmental performance both because of the adverse effects of production of energy on the environment through loadings of heat and pollutants, and because energy drives the technologies that allow improved use of land and water, the recycling of materials, and the minimization of environmental impacts. Thus, finding acceptable and sustainable sources of energy will literally prove to be the engine that drives the entire process. In this connection, I note that a recent study of sustainable development by the National Academy of Sciences recommended accelerated improvement in the use of energy as a priority, suggesting as a "reasonable goal" the doubling of historical rates of improvements in energy and materials use, including in particular the long-term reduction in the amount of carbon produced per unit energy.[1]

It is apparent in this connection that nuclear technologies offer promise as a means to achieve sustainable development. Generation of electricity by using nuclear fuels avoids emission of greenhouse gases, can minimize other effluents and wastes, and has the advantage of abundant fuel supply. Indeed, the increasingly urban nature of future populations reinforces the need for central-station power technologies that minimize adverse environmental impacts. The challenge is one of harnessing this technology in a manner that ensures safety -- a matter on which many in our society continue to have concerns. The NRC does not have a promotional role -- our mission is to protect public health and safety and the environment -- and thus I cannot advocate that society choose a nuclear path over other alternatives. Nonetheless, consistent

with our mission, the NRC seeks to ensure that the regulatory system does not stand as an impediment to the role that nuclear energy could play in achieving sustainable development, so long as it does not unduly threaten public health and safety or the environment.

In this talk I will address some of the activities underway at the Commission that bear on our efforts to assure such protection. I will also discuss the key role of ensuring the public's confidence in the ability of both government and the industry to assure such protection. Before turning to these matters, however, it may be appropriate to take a moment to discuss the current status of nuclear power in the U.S.

Current Status

Nuclear power is today experiencing a quiet renaissance in the U.S., in large part as a result of electricity price deregulation. My fellow panelist, Corbin McNeill, will perhaps address this issue in more detail. A few years ago, prognosticators forecast the demise of the nuclear power industry. The accepted wisdom was that nuclear power plants were uneconomic, that many nuclear plants would be shut down even before their licenses had expired, and that our present reliance on nuclear power would diminish and die away.

Today, in contrast to these predictions, nuclear plants are being sold in auctions with many bidders and at prices that exceed $1 billion. Moreover, instead of premature termination of licenses and a turn to decommissioning activities, we see great interest in the renewal of licenses for extended operation. The original 40-year term of operational licenses, which was established in the Atomic Energy Act, did not reflect a limitation that was determined by engineering or scientific considerations, but rather was based on financial and antitrust concerns. We now have experience on which to base judgments on the effects of aging and the potential useful life and safe operation of facilities, and have been turning to the question of extensions beyond the original 40-year term.

The interest in license renewal has arisen because the existing nuclear fleet has proven to be a reliable and, once the capital cost is amortized, a low-cost source of electrical power. The NRC has renewed the licenses of two plants comprising five units and has renewal applications of three more plants in process. In fact, the NRC has been formally advised that licensees of about 40% of currently operating plants will seek license renewal, and we have been informed that the number could ultimately exceed 85% of the current fleet. Under these circumstances, nuclear power could continue to make a substantial contribution to the U.S. energy portfolio through at least the first half of the 21st century. Furthermore, in the last few months, there has even been the first stirring of interest in the possibility of new construction of nuclear power plants in the U.S. In short, in the period of a few years, we have seen a remarkable change in the prospects for nuclear power in the United States.

The renewed interest in nuclear energy by the business community is a result of many factors -- steadily improving economic performance, changing economic conditions as a result of electric price deregulation, the increasing and highly variable cost of fossil fuels, and, importantly, the environmental advantages of nuclear power at a time when emissions from conventional plants must necessarily be subject to more stringent limits. I have termed the current climate for nuclear in the U.S. a "quiet renaissance," however, because a resurgence of interest in nuclear power in the business community has been largely unnoticed by the public and even the mainstream business press.

Fortunately, the data show that strong economic performance goes hand-in-hand with strong safety performance: attention to safety serves to make plants more reliable. Safety and economic performance are not merely compatible; they are mutually reinforcing. It is no accident that the steadily improving economic performance of nuclear plants over the past decade is accompanied by remarkable improvements in the safety performance indicators -- fewer scrams, greater availability of critical equipment, fewer unplanned shutdowns, and reduced radiation exposure of workers.

The Regulatory Role

The task for the NRC in this environment is to maintain acceptable safety performance, while simultaneously eliminating needless regulatory burden. By "needless regulatory burden," I mean those rules and regulations that are not needed to assure public health and safety, to protect the environment, or to safeguard nuclear materials. I would like to describe briefly some of our efforts in this regard.

One of the most significant programs that we have undertaken is the use of risk insights in revising our regulations and regulatory processes. The use of risk information, along with more traditional deterministic analyses and engineering judgment, has permitted the agency to focus attention on those aspects of plant design and operation that are important to safety, while eliminating regulatory requirements that do not serve to enhance safety. This offers a win-win opportunity: an improved focus on safety, while eliminating needless burden.

One of the more important initiatives in this regard is an improved reactor oversight process. Our aim was to produce a system that allocated the inspection effort to those matters important to safety, and while providing greater objectivity and transparency. We developed Performance Indicators (PIs) to allow the systematic assessment of plant performance over time, coupled with a baseline inspection program that was focused on risk-significant issues. Objectivity is served by a new and systematic process for assessing performance, coupled with predefined actions that flow from risk-significant findings. Transparency is served by making the process, Performance Indicator data, and inspection reports available to all, including by means of our NRC website. The program was developed with substantial input of our stakeholders and, although we have had only about 6 months of industry-wide implementation, the early response is quite favorable by licensees and the public.

Other efforts to improve the substance of our regulations using risk insights are also underway. For example, consideration of risk is used to modify the allowed outage times in technical specifications for safety-related equipment. Risk insights have also allowed us to approve modifications related to in-service inspection and testing requirements. And the rules governing plant changes and testing and maintenance were modified to couple regulatory requirements to risk considerations.

We have also launched a close examination of special treatment requirements - the rules governing equipment deemed to be safety-related. With insights provided from probabilistic risk assessments, we now realize that some equipment that has been categorized as safety-related,

and thus subject to special restrictions, in fact can be shown to make a limited contribution to risk. Conversely, other equipment that was not previously categorized as safety-related is now understood to have safety significance. We are engaged in an extensive process to rethink the regulatory requirements that bear on these categories of equipment. Other efforts include the standards for combustible gas control systems [10 CFR 50.44] and the acceptance criteria for Emergency Core Cooling Systems [10 CFR 50.46].

Although our effort is extensive, we have taken only the first steps in our efforts to risk-inform our regulatory systems. In a series of studies conducted in 1993, the National Academy of Engineering found that it took the Environmental Protection Agency an average of 15 years to modify its standards after scientific consensus had been reached on a better understanding of the underlying risks.[2] I expect the NRC to do much better. Nonetheless, we have a long and difficult road ahead of us for which we will require informed input from the industry and from other stakeholders.

Public Acceptance

Let me now turn to the question of public acceptance. It is completely clear to me that public attitudes will be crucial in the determination whether nuclear technologies will be part of the portfolio of energy technologies on which the world will rely to confront the challenges of the 21st century. Moreover, understanding and confronting public concerns about nuclear matters is a central obligation of nuclear regulators in any event.

There are two issues that bear on public confidence that I would like to address briefly: the first is nuclear waste disposal, and the second is the NRC's role in affecting public attitudes.

As I have traveled around the country and visited our licensees, I have noted that many people who live in the communities near nuclear power plants have become comfortable with the plants as neighbors. But one of the persistent questions about the use of nuclear power is: "what are we going to do about the waste?" A ready answer is not at hand. The "compact" system that was supposed to address low-level waste has clearly not worked in the way that Congress intended. In fact, now that the Barnwell disposal facility is part of the Atlantic compact, there is currently no widely available disposal facility for several types of low-level waste. While this does not present an immediate problem for most plants, it may well become one unless action is taken to devise a workable system for dealing with low-level waste.

Even more serious is the challenge of disposing of spent fuel and high-level waste. As most of you no doubt know, the Department of Energy is scheduled next year to make a recommendation to the President on the suitability of the Yucca Mountain site as a location for a deep geologic repository. If that recommendation is positive, and if it is accepted by the President and the Congress, DOE will then make a formal application to the NRC to license the repository. The NRC will confront technical challenges in evaluating such an application. For example, both NRC and EPA contemplate the assessment of impacts over a 10,000-year period, which is longer than recorded history. The Commission is obliged to make an informed and impartial judgment, based on the best information available, about whether DOE has adequately demonstrated compliance with the applicable EPA standards and NRC requirements. Moreover, we anticipate that any NRC decision will be the subject of intense litigation. An acceptable strategy for the disposition of spent fuel must be resolved, however, if there is to be public acceptance of nuclear technology. For the Nation to accept significant growth in the application of nuclear power, the resolution of the full cycle of impacts of the technology must be addressed.

The more general problem, and one that is not independent of the concerns about waste, is one of public attitudes. As I have mentioned, the NRC does not have a promotional role with respect to the use of nuclear technology. Nonetheless, we do recognize that the way in which we perform our jobs can have a significant impact on public attitudes. We must both be, and be seen, as a rigorous, independent and capable regulator. The significance that the Commission places on its obligations in this regard is reflected in the fact that we have identified enhancing public confidence in the NRC as one of our four major goals in our strategic planning.

In order to enhance public confidence, we must provide open processes so that the public has the opportunity to raise concerns and to see that those concerns have been weighed and fairly evaluated. To achieve this objective, we seek to involve our stakeholders - everyone with an interest in the impact of our licensees' activities - in the development and consideration of our regulatory processes. We also seek to ensure that the information that we receive from our licensees and other interested parties and that which we develop ourselves is readily available to the public - with due consideration for protection of legitimate proprietary interests. Finally, we have sought to reach and justify our regulatory decisions and policies by means that are open, so that the considerations that have guided decisions and the bases for our actions are fully accessible to the public. We cannot expect that everyone will agree with our decisions, but we can aspire to show that no legitimate concern has been ignored.

Conclusion

To sum up, I believe that, in order for nuclear power is to play a significant role in sustainable development, there must be a demonstrated commitment to safety. To this end, the industry and the NRC both have a responsibility to ensure safe operations. Moreover, the fulfillment of nuclear technology's potential is dependent on its acceptance by the public. The NRC's responsibility is to ensure that the public has reason to be confident in the NRC's capabilities as a strong, competent, and fair regulator. Other issues, such as waste disposal, must also be dealt with promptly and forthrightly. If these issues can be resolved, the renaissance of nuclear power that appears on the horizon may be realized.

Thank you.

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1. National Research Council, "Our Common Journey: A Transition Toward Sustainability," National Academy Press (1999).

2. National Academy of Engineering, "Keeping Pace with Science and Engineering: Case Studies in Environmental Protection," National Academy Press (1993).

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U.S. NUCLEAR REGULATORY COMMISSION

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COMMISSIONER DICUS ELECTED TO GOVERNING BODY OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION

Commissioner Greta Joy Dicus of the U.S. Nuclear Regulatory Commission has been elected to the Main Commission of the International Commission on Radiological Protection (ICRP). It is the first time that a member of the NRC or its predecessor agency, the Atomic Energy Commission, has been named to that world body. She is the second of only three women ever named to the Main Commission.

The election of Ms. Dicus to the 13-member governing body of the ICRP was announced by Professor Roger H. Clarke, its Chairman. He is the Director of the United Kingdom's National Radiological Protection Board and also his country's representative to the United Nations Scientific Committee on the Effects of Atomic Radiation. The term is for four years beginning July 1, 2001.

"I am extremely honored to have been elected to serve on so prestigious and important a body as the ICRP," Ms. Dicus said. "While continuing my present responsibilities at the NRC, I intend to devote such time as is necessary to help further the ICRP's work in advancing the public's understanding of radiation protection by providing sound recommendations and guidance on all aspects of protection against ionizing radiation."

The ICRP, established in 1928, is an advisory body which offers recommendations on the fundamental principles and quantitative bases upon which safe and effective radiation protective measures can be taken by various governments, legislatures and regulatory bodies around the world.

Commissioner Dicus, a member of the NRC since 1996, is a health physicist whose primary interests include formulating and helping make policy decisions regarding nuclear materials and facilities under the authority of the Atomic Energy Act, as amended. Her current term as an NRC Commissioner will run until June 30, 2003. From July 1 until October 29, 1999, she served as Chairman of the agency.

An Arkansas native, Ms. Dicus was educated in Texas. She graduated with a Bachelor of Arts degree in biological sciences from Texas Woman's University in 1961 and earned a Master's degree in radiation biology from the University of Texas Southwestern Medical School in 1967. For 16 years, from 1961 to 1977, she conducted research in radiation health effects at Harvard Medical School, Rice University, and the University of Texas Southwestern Medical School.

Since 1962, there have been four standing Committees appointed by the ICRP's Main Commission: Radiation Effects, Doses from Radiation Exposures, Protection in Medicine, and Application of the Commission's Recommendations. In addition, the ICRP uses Task Groups and Working Parties to prepare reports to be discussed by the Committees which are then approved by the Main Commission. At any one time about 100 eminent scientists are actively involved in its work. A scientific secretary coordinates the activities of the Commission and its Committees.

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NOTE TO EDITORS: NRC ISSUES "LESSONS LEARNED" REPORT ON INDIAN POINT 2 STEAM GENERATOR TUBE FAILURE

The Nuclear Regulatory Commission has issued a "lessons learned" report on the Indian Point 2 steam generator tube failure that led to a declaration of an alert at the site in Buchanan, N.Y. on February 15.

The report evaluated the NRC staff's regulatory processes related to assuring steam generator tube integrity. It identifies and recommends areas for improvements applicable to the NRC and the industry.

The NRC staff is developing an action plan that will include the disposition of the lessons learned report in an integrated manner with other ongoing steam generator issues. Recommendations in the report that apply to the industry will be considered by the staff in the context of its review of ongoing industry initiatives.

The report's executive summary is appended to this note. The "Indian Point 2 Steam Generator Tube Failure Lessons Learned Report," is available from the NRC Public Document Room, Rockville, Maryland, (301) 415-4737. The full text of the 144-page report has been posted at: http://www.nrc.gov/NRC/REACTOR/IP/index.html on the NRC web site.

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EXECUTIVE SUMMARY

The February 15, 2000 Steam Generator Tube Failure Event

On February 15, 2000, a single tube in one of four steam generators (SGs) at Consolidated Edison's (Con Ed's) Indian Point 2 (IP2) plant failed, leading to a transient and shutdown of the reactor. In addition to the reactor itself, the SGs are the major components that transfer reactor heat into steam to drive the electric turbine at a nuclear power plant. They are located inside the containment structure and are equipped with safety features to detect and initiate automatic protection actions and provide indications to the plant operators if problems develop. The tube failure consisted of a through-wall crack in one of the 3,260 tubes in one of the SGs that allowed reactor cooling water to flow through the crack into the steam generating side of the SG at the rate of about 150 gallons per minute. The reactor was safely shutdown by the plant systems and operators. The event resulted in a minor radiological release to the environment that was well within regulatory limits.

Charter

The IP2 SG Tube Failure Lessons-Learned Task Group and Charter were proposed by the Director of the Office of Nuclear Reactor Regulation (NRR) and approved by the Executive Director for Operations in June 2000. The objective of the effort was to evaluate the NRC staff's regulatory processes related to assuring SG tube integrity in order to identify and recommend areas for improvements applicable to the

NRC and/or the industry. A multi-disciplined Task Group was established in accordance with the charter consisting of staff from the Office of Research, Region I and NRR. Support was provided by the Office of the General Counsel.

The Task Group was not expected to identify the processes for resolving areas of potential weakness. The responsibility for dealing with the recommendations would be with the applicable line organization.

The Charter directed that the Task Group review the staff safety evaluation report (SER) associated with restart of IP2 with their current SGs and provide concerns or issues to the staff for action. This activity was terminated when Con Ed decided to replace their SGs before restart.

Report

This report is the result of the Task Group effort. Conclusions and recommendations were developed by the Task Group based on reviews of documents and discussions with NRC staff, nuclear industry representatives involved in SG programs, and NRC SG expert consultants. Public input was not sought as part of the Task Group effort based on the understanding that the report and other efforts would be integrated into an activity that would allow for input from a broad range of stakeholders.

The Task Group was directed to focus attention on issues directly related to the February 15, 2000, tube failure event and operation of the current SGs at IP2. Documents reviewed by the Task Group included Con Ed SG examination information and NRC SG inspection procedures and reports, nuclear industry generic SG examination guidance and associated NRC review information, NRC and Con Ed license amendment proposals and safety evaluation reports, and the Con Ed event root cause analysis and the associated NRC Special Inspection Report.

The Task Group also reviewed the following reports:

1) The Office of Research (RES) independent technical review dated March 16, 2000. Following the IP2 tube failure event, NRR requested RES to review the NRC safety evaluation associated with an IP2 license amendment that approved an extension to the SG inspection interval. The Task Group considered the issues raised in the RES review as discussed in Sections 6.2, 6.4, 7.0, and 8.1 of this report.

2) The Office of the Inspector General's (OIG) Event Inquiry on the "NRC's Response to the February 15, 2000, Steam Generator Tube Rupture at Indian Point Unit 2 Power Plant," dated August 29, 2000. The Task Group addressed the findings of the OIG report related to SG issues as discussed in Sections 6.3 and 8.1 of this report.

The Task Group effort did not consider IP2 issues unrelated to SG tube integrity or issues being addressed by other regulatory processes, such as a 2.206 petition or a differing professional opinion. The Task Group review included the licensee's results of the IP2 SG examinations and root cause evaluation in accordance with the charter. The Task Group did not evaluate Con Ed performance relative to regulatory requirements.

The conclusions and recommendations in this report represent the views of the Task Group. The recommendations were developed to address the conclusions/lessons-learned that were reached, so that with respect to SG tube integrity, the NRC can continue to maintain safety, increase public confidence, increase the efficiency and effectiveness of NRC programs, and reduce unnecessary regulatory burden. Quantitative costs and benefits were not developed for each recommendation. The objective was to provide a basis for each recommendation to support both the internal NRC planning process and the appropriate regulatory process for considering actions for the industry.

Safety Significance

The Task Group evaluated the safety significance of the event using safety assessment studies performed before and after the event. The NRC Special Inspection Team noted that there were no actual radiological consequences of the event, and that the event did not impact the public health and safety.

The Task Group agreed with this assessment.

The Task Group also considered the NRC staff's preliminary risk assessment of the IP2 event associated with the NRC significance determination process (SDP). The staff concluded that the IP2 tube failure resulted from degraded conditions allowed to exist in the SGs during the operating cycle. The staff determined that the licensee's SG tube integrity and quality assurance program was deficient and did not detect the degraded conditions. These tube conditions presented a safety concern because of a reduction in safety margin and an increased risk of SG tube rupture (SGTR) during IP2's operating cycle 14. The Task Group considers the preliminary staff assessment appropriate for the SDP process and agrees with its conclusion.

The Task Group also evaluated the overall significance of the event and condition of SG tubes relative to the NRC measures for maintaining safety in the NRC's Strategic Plan. The risk from the IP2 SG event and risk from the tube condition prior to the event were well within NRC Strategic Plan measures for maintaining public health and safety.

The Task Group concluded that the weaknesses in the Con Ed program that contributed to the poor condition of the failed SG tube have generic implications. The examination guidance in use is common throughout the pressurized water reactor (PWR) industry. While the IP2 SGs now being replaced are the last of their particular model, Task Group review of other SG designs and tube materials indicate potential generic applicability of the IP2 lessons. Review of PWR risk analysis confirms that SG tube integrity is important at all PWRs. Therefore, the Task Group concludes that a high priority should be assigned to improvements in the SG tube integrity program at IP2, for the industry guidance on SG tube integrity programs, and associated NRC regulatory programs.

The Task Group concluded that communicating the safety significance of the IP2 experience is difficult. During the NRC significance determination process related to the IP2 tube failure, the staff found that the SG tube condition during Cycle 14 was risk significant due to the degradation of safety margin. Notwithstanding the loss of safety margin, IP2 is designed to mitigate the effects of SG tube failure or tube rupture, IP2 shut down safely following the tube failure, and the IP2 event resulted in no adverse consequences to the public health and safety. This distinction may not be understood by all stakeholders. NRC will probably face this communications challenge again because SG tube failures and ruptures have occurred before and will likely occur again. Therefore, the Task Group recommends that the NRC should incorporate experience gained from the IP2 event and the SDP process into planned initiatives on risk communication and outreach to the public.

Steam Generator Tube Integrity Program Regulatory Framework

All PWR reactor plant licensees are required by NRC regulations to provide reasonable assurance of SG tube integrity. A significant number of NRC regulations and standards apply and are incorporated into the licensing basis of each facility. These requirements include design, operation, and surveillance activities. The surveillance requirements are important to maintaining integrity since different types of tube degradation are expected to occur over the life of the SG. Current plant technical specifications typically require that a representative sample of tubes be examined for defects using eddy current testing once every two to four years during the periodic plant shutdown period. Eddy current testing is a method of inspecting SG tubes by passing a probe that generates an electromagnetic field through the tubes. Tubes that are identified as containing defects of a specified depth are removed from service, typically by plugging both ends of the defective or degraded tube.

In recent years, the NRC staff has examined the regulatory programs which comprise the framework for ensuring the integrity of SG tubes. In the mid 1990's, the staff concluded that existing regulations provided an adequate regulatory basis for dealing with SG issues, but thought them to be prescriptive, out of date, and not fully effective. In 1997, the Commission approved the staff's approach to upgrade plant technical specifications, and the Nuclear Energy Institute voted to adopt NEI 97-06, "Steam Generator Program Guidelines," as a formal industry initiative to provide a consistent industry approach for managing SG programs and for maintaining SG tube integrity. In 1998, the Commission approved a revised approach to work with the industry consistent with Direction Setting Initiative 13, "The Role of

Industry," to more efficiently resolve program concerns and move toward NRC endorsement of NEI 97-06, coupled with voluntary industry implementation of improved SG technical specifications.

Steam Generator Tube Integrity Program Lessons-Learned

The Task Group concludes that there are a number of plant-specific and generic lessons- learned that support recommendations to improve industry SG tube integrity programs.

Con Ed

The Task Group reviewed major aspects of the 1997 Con Ed SG examinations and plans leading up to these examinations. These same activities were the subject of an NRC Special Inspection Team review and are documented in its report of August 31, 2000. The Task Group agrees that the inspection findings are of potential high significance, as proposed. The key deficiencies noted were that:

1) During the 1997 SG eddy current examination by Con Ed, a defect caused by primary water stress corrosion cracking (PWSCC) was identified for the first time in a tube similar in type and location to the tube that failed at IP2, and Con Ed did not effectively evaluate the susceptibility of similar tubes to this degradation during the upcoming operating cycle.

2) During the 1997 SG examination, a form of degradation called tube denting was identified when restrictions were encountered as the eddy current probes were inserted into the U-bend portion of similar tubes. Con Ed did not evaluate the potential for, and significance of, this degradation.

3) During the 1997 examination, significant eddy current signal interference (noise) was encountered in the data obtained from a number of tubes similar to the tube that failed, and Con Ed's program was not adjusted to compensate for the noise, particularly when the new PWSCC defect was found in this area of the SG.

The Task Group believes that the findings of the Special Inspection Team are reasonable and that corrective actions at IP2 should proceed in accordance with the ongoing inspection and enforcement process.

Industry / NEI / EPRI

Along with the plant-specific SG examinations conducted by Con Ed at IP2 during 1997, the Task Group reviewed the industry SG examination guidance used by Con Ed during the 1997 outage and concluded that there were weaknesses in the guidance as well as in their implementation. The guidance was developed and is maintained by the Electric Power Research Institute (EPRI). Since the EPRI guidance is a cornerstone of the industry initiative now being coordinated with the Nuclear Energy Institute (NEI), the Task Group believes that the industry should be requested by the NRC to expeditiously ensure that the lessons-learned from the IP2 event are incorporated into the guidelines and implemented by all licensees and that feedback be provided to the NRC on the status.

Particular improvements to the EPRI guidelines to improve the effectiveness of SG examinations are discussed in detail in Section 6 of this report. The Task Group believes that the guidance in use during the 1997 IP2 examinations was not explicit with respect to the quality of eddy current data and the significance of noise in the data. The need for increased licensee attention when "new" types of degradation are found should be emphasized in the guidance. The Task Group understands that industry is already taking steps to make improvements and believe they should be discussed with the staff, and schedules determined for their incorporation.

The following additional issues that should be pursued with the industry for improvements in the guidance and implementation by licensees were identified by the Task Group:

1) Licensees should review generic industry guidelines carefully to ensure that the conditions/assumptions supporting the guidelines apply to their plant-specific situation. The

plant-specific qualification of eddy current techniques to perform inspections is fundamental to an adequate inspection.

2) Licensees should use caution when assessing SG tube structural integrity by using unqualified sizing techniques for growth rates and threshold of detection. Licensees should use a conservative approach to screen tubes for in-situ testing.

3) A noise study performed by NEI indicates that SG tube U-bend noise may be significant regardless of tube age or outside deposits. Flaw detection capabilities in the U-bend region should be assessed for all SGs.

4) Vendors that conduct the actual examinations, including collection and analysis of the data, are important to the SG examination process. The industry initiative should address vendor oversight by licensees.

Other recommendations to improve the effectiveness of the guidance can be found in Section 6 of this report.

Industry Initiative and Framework

The Task Group considered the implications on the industry initiative and framework, given the IP2 event and its lessons-learned, the weaknesses in the EPRI guidance, and the safety significance of the issues. The Task Group believes that the industry initiative remains an effective means to continue to maintain safety in this area. However, the lessons-learned discussed above identify issues that should be incorporated into the framework in an integrated way. The Task Group concludes that the industry should be requested to evaluate and propose modifications to the framework that consider the lessons-learned from IP2. These should include, as a minimum:

1) means to ensure plant-specific licensee attention to lessons-learned;

2) improvements to the EPRI guidelines, and

3) content of the improved technical specifications relating to SG degradation mechanisms, examination techniques, primary-to-secondary leakage limits, and reporting requirements (both content and schedule of reports).

As stated above, the Task Group believes these activities should receive a high priority. Therefore, in the interim, the Task Group believes that the NRC should issue a generic communication to clarify the current NRC position on industry guidance and to highlight SG tube integrity program weaknesses manifested by the IP2 experience that could exist at other plants.

NRC Regulatory Processes

Based on a review of the licensing, inspection and oversight processes associated with the IP2 event and SG tube integrity, the Task Group believes that there are areas that should be improved in these processes to make them more effective, as discussed below.

Licensing

The license amendment process is used by the NRC to review facility operating license changes proposed by a licensee. Such a request was made by Con Ed in December 1998 to extend its SG examination from June 1999 to June 2000. In effect, because of an approximate 10 month period the plant was shut down, the licensee was actually requesting an extension of the examination interval of approximately 2 months beyond the already authorized 24 months (June 1997 to June 1999). This is illustrated in the Appendix A timeline of this report. Because the licensee followed industry guidelines for maintaining water chemistry in the SGs to minimize corrosion of the SG tubes and the reactor coolant system was at low temperature conditions during the shutdown, any degradation that would have

occurred during the shutdown period should have been negligible.

The 1997 SG examination performed by Con Ed, which has now been determined to be deficient as discussed above, was the underlying basis for the SG inspection interval extension amendment that was requested by Con Ed. Thus, the Con Ed amendment request and the NRC licensing review provided an opportunity for Con Ed and the NRC to reevaluate the adequacy of the 1997 examination. After the February tube failure event, NRR requested RES to review this extension request along with the associated NRR safety evaluation of the proposal. The RES technical review was provided in a report dated March 16, 2000. The OIG also evaluated this licensing review and provided its findings in a report dated August 29, 2000. Both of these reports identified shortcomings in the licensing review. They were considered in detail by the Task Group, along with the specific licensee and staff documents and review guidance, in reaching conclusions and recommendations.

The significant conclusions from the Task Group review of the licensing review process associated with the Con Ed amendment request to extend the SG inspection interval are:

1) There was an opportunity for Con Ed during preparation of the amendment request and subsequent response to an NRC request for additional information to recognize the significance of a new degradation mechanism that was observed during the 1997 SG examination in a tube similar to the one that failed in February 2000 (PWSCC at tube apex in a small radius U-bend).

2) In hindsight, during the amendment review process, the issue regarding the PWSCC degradation could have been pursued further by the NRC staff. If the staff had denied the amendment request, an examination would have been required prior to the tube failure. However, based on a review of information available to the licensee and the staff during the amendment review, it is not clear to the Task Group if additional staff questions posed during the review would have changed the outcome of the license amendment request or uncovered the issues related to the root cause of the tube failure. For example, Con Ed had performed an examination of all other similar tubes using an inspection plan previously reviewed and approved by the staff.

3) The IP2 tube failure occurred on February 15, 2000, which was approximately 8 months after the originally scheduled inspection date (i.e., less than the duration justified by the 10 month shutdown). Therefore, the extension of approximately 2 months did not contribute to the tube failure event. This is illustrated in the Appendix A timeline of this report.

4) While the staff used existing NRC review guidance in performing the review, no specific guidance exists for SG inspection interval extensions, especially how to consider previous inspection reports, or how to consider or reference the inspection program.

A detailed discussion of the Task Group review of the licensing review process associated with the Con Ed amendment request to extend the SG inspection interval (including Task Group comments on the related OIG report findings) is included in Section 8.1 of this report. The Task Group's review of the issues addressed in the RES technical review is included in Sections 6.2, 7.0, and 8.1 of this report.

While the Task Group did not evaluate the area of staff SG expertise in detail, this was brought up by the OIG report, and was mentioned in conversations with NRC staff and managers responsible for these programs. The Task Group believes that agency SG expertise is limited and focused primarily at headquarters. The Task Group recommends that NRC take steps to evaluate SG expertise needs to support the licensing (as well as inspection) program.

In summary, the Task Group believes that the problem relates back to the quality of the Con Ed 1997 examination. Improvements to industry SG examinations (discussed above) and NRC regulatory inspection processes that focus on these examinations (discussed below) will maintain plant safety and improve the efficiency and effectiveness of NRC programs. The Task Group believes that additional review guidance for SG examination license amendments will improve the effectiveness and efficiency of these reviews.

Inspection

The objective of the NRC inspection program is to obtain factual information providing objective evidence that power reactor facilities are operated safely. The SG tube failure at IP2 occurred at a time when the NRC was transitioning to a new reactor oversight process (ROP). Effective April 2, 2000, the NRC implemented this new process for all plants. The Task Group reviewed both the old and new NRC inspection processes to develop lessons-learned and recommendations.

The baseline inspection in the new ROP for inservice inspection (ISI) is to be performed at all operating reactors, once every two years during a refueling outage. Supplemental inspections are performed as a result of risk-significant licensee performance issues that are identified by either PIs, baseline inspections, or event analysis.

Prior to April 2000, an NRC ISI inspection was performed at each facility in accordance with the core inspection program. This program was in effect during the NRC inspection of IP2 in 1997. The scope of the inspector's review was based on a judgement regarding current significant issues and also as directed by the inspector's supervisor. The planning did not usually involve NRC headquarters personnel. It did not require that industry information be factored in, although it sometimes was. New industry and generic information, such as Information Notices and Generic Letters, did not always get to the regional inspectors in time enough to be factored into their inspection activities. The site inspection involved one inspector for a period of one week and was not necessarily limited to SG activities, but it could also include non-destructive examination (NDE) activities on other components.

NRR has routinely held conference calls with each licensee during their refueling outage to assess the adequacy of the licensee SG tube eddy current inspections. These conference calls involve regional participation on occasion and include discussion of the results of the licensee generator inspections and repair plans. In the last few years, the staff has focused on plants with known SG tube degradation issues. This effort has not been a formal part of the inspection program, and the results are not documented in inspection reports. During consideration of the NRC inspection activities, the Task Group interviewed NRC staff involved in the phone calls and reviewed some of the records of the 1997 outage NRC/Con Ed telephone calls held on June 2, 3, and 29, 1997. Some staff members interviewed by the Task Group indicated that they had specifically asked Con Ed during the phone calls if any U-bend degradation in small radius U-bends had been identified. There was no indication that the crack discovered in the tube similar to the tube that failed was discussed. The timing of the phone calls relative to when the flaw was identified was not clear. The Task Group determined that these calls are important activities that should be factored into the inspection process.

The new ROP baseline inspection procedure for ISI does not include guidance on the scope and depth of NRC inspection of licensee SG tube examinations. The inspection procedure contains significantly less guidance for conduct of the inspection than the previous core inspection procedure. Available supplemental procedures contain considerably more detail. Under the new ROP, risk-informed thresholds are to be applied to inspection findings to determine when a significant degraded condition has occurred that warrants additional NRC interaction and supplemental inspection above the baseline program. Such thresholds do not currently exist to identify when the number or types of SG tube defects have reached a level that warrants additional NRC action.

There are no specific requirements for ISI inspector training or expertise. Region staff interviewed indicated that as part of the training program, prior to conducting individual inspections, inspectors assist other inspectors on NRC's NDE inspections at other reactor sites. A number of inspectors have received detailed training in eddy current examination and have personal NDE experience.

The Task Group also carefully reviewed the licensee submittal to the NRC dated July 29, 1997, regarding the IP2 1997 SG examination. The level of detail provided in the 1997 examination report submitted by Con Ed was not sufficient to pinpoint the technical and implementation problems, such as the eddy current data quality and noise issues discussed above (and in Section 6.1 of this report). The Task Group noted that the tube that failed was not reflected in the report as a degraded tube, since it was not identified by the licensee as such during the 1997 examination. The NRC's OIG report dated August

29, 2000, concluded that had the NRC staff or contractor with technical expertise evaluated the 1997 results of the IP2 SG inspection, the NRC could have identified the flaw in the U-bend of the row 2, column 5 (R2C5) tube in SG 24 that was indicated in the licensee's inspection (examination) report. After careful review, the Task Group concluded that the NRC staff could not have identified the tube that failed from its review of the licensee's inspection report. That report did not indicate that there was a flaw in the tube or provide any information on the tube. Even if the staff should have been prompted by the report's identification of a new degradation mechanism (PWSCC) in a similar tube that was plugged, it would have required further discussion with the licensee, additional staff review of the 1997 raw eddy current data of the failed tube, and identification of the flaw from the data, which clearly was of poor quality due to noise. Experts that the Task Group interviewed held different views on whether the flaw in R2C5 could have reasonably been detected from the data. Licensee reports in general, and this report in particular, do not provide this information or related discussions or evaluation of eddy current data. For the NRC to have this information, an eddy current specialist would have to review the raw data independently. This is not typically included within the scope of NRC inspection or review.

Details of the Task Group review of the NRC SG inspection program are provided in Section 8.2 of the report. Overall, the Task Group believes that:

1) The NRC should develop additional SG inspection guidance for the baseline inspection program.

2) Inspector training should be reviewed and tailored to support the objectives of the SG inspection program.

3) Information needs and processes to support the objective of the SG inspection program should be determined. In this regard, the Task Group believes that the telephone calls conducted with licensees during the outages are effective and should be formally incorporated into the inspection program.

4) Risk-informed thresholds should be established to identify when increased NRC interaction is warranted in response to SG tube degradation.

5) The baseline program and/or performance indicators should be modified to identify adverse trends in primary-to-secondary leakage. Risk-informed thresholds should be established to identify when increased NRC interaction is warranted in response to an adverse trend.

Conclusions and Recommendations

Sections 5.0 through 8.0 of this report include sub-sections (as indicated in the report Table of Contents) that provide the conclusions/lessons-learned and recommendations for the respective section. Section 9.0 provides a table (Table 9-1) that lists all the report recommendations with a reference back to the supporting section in the report. The recommendations fall into the following areas:

1) Con Ed must correct the deficiencies in its SG tube integrity program;

2) Industry should improve the EPRI guidelines;

3) Industry should improve the SG technical specifications;

4) Industry should improve the NEI 97-06 initiative;

5) The NRC should improve its SG oversight and inspection process;

6) The NRC should improve its licensing review process;

7) The NRC should assign a high priority to its review of the NEI initiative and the associated EPRI guidelines;

8) The NRC should issue a generic communication regarding SG tube integrity program guidance; and

9) The NRC should improve risk communication to the public.

Overall, the Task Group believes that the lessons-learned from IP2 are important relative to assuring SG integrity and that the industry initiative should expeditiously incorporate the lessons-learned into the regulatory framework.

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[**NRC Home Page** | **News and Information** | **E-mail**]

November 3, 2000

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO-I-00-035

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 Region I staff on this date.

Facility	Licensee Emergency Classification
North Atlantic Energy Service Corporation Seabrook	Notification of Unusual Event Alert
Route 1 Seabrook, New Hampshire 03874	Site Area Emergency
Docket No: 050-443	General Emergency
License No: NPF-86	X Not Applicable

Subject: Seabrook Station "B" Emergency Diesel Generator Damage During Testing - Special Inspection Team

Seabrook Station is currently in a refueling outage with all fuel removed from the reactor vessel.

Between October 29 and November 1, 2000, the licensee attempted to perform a 24-hour endurance surveillance test on the "B" Emergency Diesel Generator (EDG). The attempts on October 29 and 30, 2000, resulted in operators securing the EDG based on high differential pressure across the lubricating oil filter. The attempt on November 1 encountered "crankcase high pressure" and "EDG high vibration" alarms and required emergency EDG shutdown. In addition, the operators in the EDG room noted a flash from the turbo-charger area and heavy black smoke. Subsequent preliminary assessment by the licensee has revealed damage to at least one of the EDG cylinder liners.

The licensee formed an Event Team on October 31, 2000, to assess the problems with the lubricating oil system, and expanded the scope of that team on November 1, 2000, based on the last test failure.

Region I received initial notification of the third test failure resulting in damage to the EDG by telephone on November 1 from the senior resident inspector. On November 2, Region I, in consultation with NRR, decided to send a Special Inspection Team to the site to monitor and assess the licensee's root cause evaluation and corrective actions, independently evaluate the risk significance of the EDG test failures, and determine possible generic implications. The resident inspector is a member of that team and will monitor licensee actions until the team is dispatched.

The State of New Hampshire has been notified of this event and the NRC's decision to conduct the special inspection.

Region I Public Affairs is issuing a press release to announce the special inspection and is prepared to respond to media inquiries.

The information presented herein has been discussed with the licensee, and is current as of November 3, 2000.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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

No. 00-175 November 13, 2000

NRC APPROVES TRANSFER OF OPERATING LICENSES FOR INDIAN POINT 3 AND FITZPATRICK TO ENTERGY

The Nuclear Regulatory Commission staff has approved the transfer of the operating licenses for the Indian Point 3 and James A. FitzPatrick nuclear power plants to subsidiaries of Entergy Corporation. Both licenses had been held by the Power Authority of the State of New York (PASNY).

The Indian Point 3 plant is located in Buchanan, New York, about 24 miles north of New York City. The James A. FitzPatrick plant is located in the town of Scriba, about eight miles northeast of Oswego, New York.

In May, applications were submitted to the NRC requesting approval for the license transfers. The key issues considered by the NRC technical staff included the prospective licensees' technical and financial qualifications to operate the plant, and decommissioning funding assurance.

Notice of the requests for approval and opportunities for a hearing were published in the *Federal Register* on June 28. The Commission received hearing requests from municipalities that purchase power from these utilities, from labor unions, and from a public interest group. Commission review of these requests, which include issues such as decommissioning funding, is pending.

In these cases, the technical staff's approval becomes effective immediately. However, if the Commission decides to grant a hearing and rules in favor of any of the petitioners, it could rescind the license transfer approval or modify its terms.

The Entergy Corporation, through its subsidiaries, owns and operates six nuclear power plants at five sites -- Arkansas Nuclear One, Units 1 and 2; Grand Gulf, in Mississippi; River Bend Station and Waterford 3, in Louisiana; and Pilgrim, in Massachusetts. An Entergy affiliate is also overseeing the decommissioning of the Maine Yankee and the Millstone, Connecticut, plants.

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November 20, 2000

EA-00-179

Mr. John Groth
Senior Vice President - Nuclear Operations
Consolidated Edison Company of
New York, Inc.
Indian Point 2 Station
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A RED FINDING AND NOTICE OF VIOLATION AT INDIAN POINT 2 (NRC Inspection Report 05000247/2000-010)

Dear Mr. Groth:

The purpose of this letter is to provide you with the final results of our significance determination of the preliminary Red finding identified in the subject inspection conducted between March 7, 2000 and July 20, 2000. The inspection report was sent to you in a letter dated August 31, 2000. This inspection finding was assessed using the significance determination process and was preliminarily characterized as Red, an issue of high safety significance.

This finding involved deficiencies in the overall direction and execution of the 1997 steam generator (SG) inservice examinations at Indian Point 2. Specifically, Consolidated Edison did not identify and correct a significant condition adverse to quality, namely, the presence of primary water stress corrosion cracking (PWSCC) flaws in steam generator tubes, despite opportunities to do so. As a result, tubes with PWSCC were left in service following your 1997 SG inspection until one of these tubes failed on February 15, 2000, when the reactor was at 100% power. As noted in the subject inspection report, the specific opportunities to recognize degraded tubes included the identification of a PWSCC defect, indications of tube denting, and significant eddy current test signal interference. While there were no public health and safety consequences from the tube failure event itself, leaving the degraded tube in service following your 1997 SG inspections resulted in a significant reduction in safety margin during Operating Cycle 14 based on the increased probability of a steam generator tube rupture event.

Our August 31, 2000, letter also provided you an opportunity to attend a Regulatory Conference. The conference, which was open for public observation and transcribed, was held on September 26, 2000, to further discuss your views on this issue. During the conference, your staff discussed an analysis of the probability of a tube rupture, your assessment of the significance of the issue, and measures to prevent recurrence. Also, you indicated that your risk analysis characterized this issue as a Yellow finding, based on your plant-specific analysis of the degraded condition following the 1997 inspection. As a result of your presentation, the NRC requested additional information to support your contention. That additional information, as well as the transcript of the conference and your presentation, were issued by the NRC on October 24, 2000.

The NRC has evaluated the information developed during the inspection, as well as the information you presented during and subsequent to the conference. Based on that evaluation, although the NRC has lowered its calculation of the risk estimate in this case, the NRC revised risk estimate (Enclosure 2) remained above the threshold for classifying this finding as Red, an issue of high safety significance. The NRC recognizes that there is a wide band of uncertainty involved in such risk calculations and additional extensive review could possibly remove some of those uncertainties. Our risk estimate, which classifies the finding as Red, does include a sensitivity analysis that for certain assumptions shows a range of results at the Yellow/Red threshold. However, as noted in our October 10, 2000 letter, the Indian Point 2 facility has been found to have multiple degraded cornerstones. In response to deficiencies at the Indian Point 2 facility, the staff is following guidance in the NRC Action Matrix, which includes oversight of your performance improvement plan and conduct of a significant team inspection.

You have 10 business days from the date of this letter to appeal the staff's determination of significance for the identified Red finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 3.

The NRC has determined that your failure to identify and adjust or modify the inspection methods and analysis to account for significant conditions that affected the quality of the 1997 steam generator inspection is a violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, as cited in the enclosed Notice of Violation (Notice). The circumstances surrounding the violation were also 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 Red finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be 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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

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Sincerely,

/RA/

Hubert J. Miller
Regional Administrator
Region I

Docket No. 05000247
License No. DPR-26

Enclosures
1. Notice of Violation
2. NRC Significance Determination Analysis

cc w/encls:
A. Alan Blind, Vice President - Nuclear Power
J. Baumstark, Vice President, Nuclear Power Engineering
J. McCann, Manager, Nuclear Safety and Licensing
B. Brandenburg, Assistant General Counsel
C. Faison, Director, Nuclear Licensing, NYPA
J. Ferrick, Operations Manager
C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
P. Eddy, Electric Division, Department of Public Service, State of New York
T. Rose, NFSC Secretary
F. William Valentino, President, New York State Energy Research and Development Authority
J. Spath, Program Director, New York State Energy Research and Development Authority
County Clerk, West Chester County Legislature
A. Spano, Westchester County Executive
R. Bondi, Putnam County Executive
C. Vanderhoef, Rockland County Executive
J. Rampe, Orange County Executive
T. Judson, Central NY Citizens Awareness Network
M. Elie, Citizens Awareness Network
D. Lochbaum, Nuclear Safety Engineer, Union of Concerned Scientists

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NOTICE OF VIOLATION

Consolidated Edison Company of New York
Indian Point 2 Station

Docket No. 05000247
License No. DPR-26
EA-00-179

During an NRC inspection conducted from March 7 through July 20, 2000, 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 Actions," requires that measures shall be established to assure that conditions adverse to quality 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, despite opportunities during the 1997 Indian Point 2 refueling outage, Con Edison did not fully identify and correct a significant condition adverse to quality involving the presence of primary water stress corrosion cracking (PWSCC) flaws in four Row 2 steam generator tubes, in the small-radius low-row U-bend apex area. In conducting the 1997 steam generator inservice inspection, Con Edison did not adequately account for conditions that adversely affected the detectability of, and increased the susceptibility to, tube flaws. Specifically, while performing steam generator eddy current test (ECT) examination, during the 1997 outage:

- a PWSCC defect was identified for the first time, at the apex of one row 2 tube, signifying the potential for other similar cracks in the low-row tubes. However, Con Edison did not adequately evaluate the susceptibility of low-row tubes to PWSCC and the extent to which this degradation existed.
- indications of tube denting were identified for the first time in low-row tubes at the upper tube support plate (TSP) when restrictions were encountered as ECT probes were inserted into those tubes. Restrictions in 19 low-row tubes signified increased probability of deformed flow slots (hour-glassing) at the upper TSP. Hour-glassing of the upper TSP increases the stresses at the U-bend apex of

tubes. These stresses are a prime precursor for PWSCC. However, Con Edison did not adequately evaluate the potential for hour-glassing based on the indications of the low-row tube denting.

- significant ECT signal interference (noise) was encountered in the data obtained during the actual ECT of several low-row U-bend tubes. This significant noise level reduced the probability of identifying an existing PWSCC tube defect. However, the 1997 SG inspection program was not adjusted to compensate for the adverse effects of the noise in detecting flaws, particularly when conditions that increased susceptibility to PWSCC existed.

As a result, a minimum of four tubes (with PWSCC flaws in their small radius U-bends) were left in service following the 1997 inspection, until the failure of one of these tubes occurred on February 15, 2000 while the reactor was at 100% power.

This violation is associated with a Red SDP finding.

Pursuant to the provisions of 10 CFR 2.201, Con Edison 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 I, 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, (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/NRC/ADAMS/index.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 20th day of November 2000

Significance Determination Risk Assessment for Indian Point Unit 2 Steam Generator Inspection Findings - Review of Licensee Response to Initial Significance Determination and Final Staff Analysis

Result

The staff has reviewed the licensee's risk assessment and its supporting analyses, and has produced this final risk assessment. For the purpose of significance determination, the numerical result of the staff's assessment is in the "red" range on the basis of a LERF contribution that is greater than 1×10^{-5}/reactor-year. The bases for this result are documented, herein.

Background

The staff's initial risk assessment to support the significance determination process for the Indian Point Unit 2 steam generator inspection findings estimated the increase in the large early release frequency" (LERF) to be on the order of 10^{-4}/reactor-year. This supported an initial significance level of "red." In response," Consolidated Edison Company, the licensee for Indian Point Unit 2, presented results and some supporting information for its own revised risk assessment at the regulatory conference held on September 26, 2000.

The licensee's analysis made several changes to the earlier assessments. The steam generator failure initiating event was split into two parts, according to break flow rates, a Monte Carlo analysis was performed to estimate the frequency of the two break sizes, human error probabilities were reduced for events with the smaller break size, and 87% of the resulting core damage frequency (CDF) was removed from the LERF category on the basis of considerations regarding the path the radioactive materials would travel from the damaged core to the atmosphere. The licensee's analysis did not address the potential for additional LERF due to tube failure during a core damage accident that might be caused by some event unrelated to tube condition, such as a station blackout. The licensee's final result is a CDF contribution of 6.6×10^{-6}/ry and a LERF contribution of 3.6×10^{-6}/ry. If accepted by the staff, this would change the significance level to "yellow" on the basis of the LERF contribution.

Staff Response to Licensee's Analysis

The staff has reviewed the licensee's revised risk analysis and supporting material. Staff conclusions regarding each of the licensee's analytical modifications are discussed below, by topic. In the following section, the staff presents its final risk analysis incorporating those factors that it finds to be appropriate.

1. Split Initiating Event Frequency into Two Break-Size Categories

This technique for grouping initiating events is an appropriate and often-used technique in probabilistic risk assessments. It allows events that have different steps for the mitigation processes or substantially different probabilities for success of similar steps to be treated separately. The split used by the licensee puts tube breaks that exceed the flow of one charging pump but not full charging capacity into a different initiating event than the breaks that exceed full charging capacity. The licensee then reevaluated the human error probabilities for the increased time available at the maximum break flow for the smaller break size category. The staff finds that this is appropriate and can facilitate improved analysis. However, because reactor coolant system pressure can be maintained with breaks of the smaller size, there is still a potential for the break size to be increased during the event. Therefore, the logic for the smaller break size should account for the potential for operator error to create conditions that might increase the break size.

2. Tube Break Flow Rate for IP2 Event on February 15, 2000.

The licensee presented a mass-balance analysis which concluded that the flow through the tube break during the February 15, 2000 event was 109 gpm. Staff review of the licensee's analysis indicates that the flow rate was higher. (See the Augmented Inspection Team report 05000247/2000-002 dated April 28, 2000.) The licensee used its flow rate estimate and information on the crack length for this and other steam generator tube failures to demonstrate that apex cracking in tubes will result in lower break flow rates than those that occur for other types of tube cracks of the same length. Only six data points are used, so this provides little confidence regarding the maximum flow rates possible from apex cracking. The licensee presented metallurgical data to indicate that apex cracks should burst at higher pressures and open less at sub-burst pressures, compared to cracks in straight tube sections. Staff analysis concludes that burst is still possible for apex cracks, although it may be less likely. Because the licensee's revised analysis did not take credit for a reduced maximum break flow rate for its large tube break event category, the difference in leak flow assumptions does not significantly impact the staff's analysis.

3. Initiating Event Frequencies for Spontaneous Tube Ruptures

The licensee used a Monte Carlo analysis to estimate the frequency of occurrence for tube breaks of each size. While the staff agrees that Monte Carlo techniques are appropriate tools for combining widely varying parameters with complex interrelationships, it notes that the results must be checked for consistency with known information before the results are credited. The licensee's results and the actual occurrences are:

Leak Rate Range	Fraction of Results	Actual Events
< 0.1 gpm	< 0.1%	0 (0%)
0.1 gpm to 75 gpm	37.2%	0 (0%)
75 gpm to 225 gpm	55.0 %	2 (67%)
> 225 gpm	7.8%	1 (33%)

Because all of the real events had flow rates 134 gpm, it would have aided the consistency check if the licensee's results had a break point at that value. Even so, it is apparent that the ratio of the licensee's initiating event frequencies is substantially shifted from the actual experience. Given the one actual event with break flow above 225 gpm, the licensee's Monte Carlo analysis indicates that there should have been about 12 events with lesser flow rates, but only 2 have occurred. Similarly, a rough comparison can be made for events with flow rates above and below about 130 gpm by assuming that about half of the licensee's results for the 75 to 225 gpm range fall below 130 gpm. If so, then the Monte Carlo calculation predicts a 1:2 ratio of events with flows above 130 gpm to events with flows below 130 gpm. Because we have experienced 3 events with flows above 130 gpm, this would predict an additional 5 or 6 actual events with flows below 130 gpm. Therefore, the staff concludes that the licensee's Monte Carlo analysis does not provide an appropriate basis for estimating the ratio between the initiating event frequencies for the two tube break sizes. Accordingly, the staff evaluation used the 2:1 ratio from the actual experience.

A second issue with respect to the initiating event frequencies is that the licensee's risk assessment averaged the occurrence fractions given above over the entire two year operating period, effectively halving the initiating event frequencies. First, the staff notes that averaging is inconsistent with the licensee's Monte Carlo analysis, which placed approximately 90% of the failures in the first 90 days of the two year period that was modeled. This provides another indication that the licensee's Monte Carlo analysis is not appropriate for quantifying initiating events. More importantly, the staff notes that the continuing deterioration of the tubes over time makes the last part of the cycle contribute the most risk, especially when operation is terminated by a tube failure. Therefore, the staff based its significance determination on the increase in core damage and large early release frequencies calculated as the average over the last year of operation for this case.

4. Human Error Probabilities

In a teleconference on October 20, 2000, the licensee supplemented (and corrected some of) the information provided in the regulatory conference. Specifically, the conditional probabilities for core damage and large early release were provided for spontaneous and induced tube rupture sequences in each of the two tube break size ranges. These and their corresponding results are:

Sequence	Initiating Event Freq.	Conditional Probability	Result
SGTR >225	0.0385/yr	7.75×10^{-5}	2.98×10^{-6}/ry (CDF)
		1.0×10^{-5}	3.87×10^{-7}/ry (LERF)
SGTR 75-225	0.275/yr	2.90×10^{-6}	7.97×10^{-7}/ry (CDF)
		1.60×10^{-6}	4.4×10^{-7}/ry (LERF)
MSLB/SGTR >225	0.0076/yr x 0.0385	2.5×10^{-3}	7.31×10^{-7}/ry (both)
MSLB/SGTR 75-225	0.0076/yr x 0.275	1.0×10^{-3}	2.09×10^{-6}/ry (both)
			6.6×10^{-6}/ry (CDF)
			3.6×10^{-6}/ry (LERF)

This information provides some insight into the degree of mitigation credit taken in the licensee's analysis for the smaller break size. For spontaneous tube ruptures, it is about a factor of 27 reduction for CDF and about a factor of 6 for LERF. For the sequences with tube rupture induced by steam line break, the factors are only 2.5 for CDF and LERF. Although the overall human error probability appears to be very small (for an HEP) in the case of the spontaneous SGTR case with break flow below 225 gpm, the staff will use the licensee's HEP in the risk analysis[1].

5. Tube Ruptures Induced by Steam Line Break

As indicated in the table above, the licensee used the same numerical values it had derived for the initiating event frequencies of the two break size categories as if they were also the conditional probability of inducing those sizes of breaks by increasing the pressure differential with a main steam line break event. This raises two issues.

The first is that the staff does not agree it is proper to use initiating event frequencies as conditional probabilities. The staff estimates that, during the last year of operation, the largest flaw left in service in 1997 had a conditional probability of 1 that it would rupture if exposed to the higher pressure differential resulting from depressurization of the secondary side of the steam generator.

The second issue is the numerical split of the conditional failure probabilities between the break size categories. At the regulatory conference, the licensee stated that it had considered the pressure differential across the tube wall to be limited to 1800 psi during a main steam line break accident, due to the characteristics of the plant's safety injection pumps. However, without a break in a tube, RCS pressure can exceed the shut-off head of the safety injection pumps once the cooling effect of the steam line break is terminated, because the charging pumps are still running. Operator action is necessary to limit the pressure differential across the tubes. The emergency operating procedure guidelines (EPGs) for Westinghouse plants call for limiting the pressure difference to 1600 psid to minimize the potential for inducing tube ruptures. But, the EPGs also state that the tubes will be able to withstand full RCS pressure so long as the tube integrity has been maintained according to the licensing basis requirements. The problem is that the operators will always expect that the tubes are being maintained in accordance with the licensing basis. Consequently, a pressure differential approaching the normal RCS pressure level may occur. Compared to its behavior in a spontaneous rupture, a crack that fails at a higher pressure difference would be expected to open more, which would increase flow rate. However, operating experience data is not readily available to derive the frequency of steam line breaks with each maximum pressure differential nor is the flaw population size data available to estimate the probability distribution of final flow rates after a crack fails.

As a result, the staff estimated that the conditional probability of a tube break was about 1.0 if a steam line break event had occurred during the final year of operation. Although somewhat non-conservative (induced tube failures would be expected to result in a larger fraction being at higher leak rates), the staff did set the probability for resulting flow rates in the 75 to 225 gpm range at 0.67 and the probability for flow rates above 225 gpm at 0.33. These estimates were based on experience with spontaneous rupture events not for induced tube ruptures.

6. Number of Steam Generators Affected by Steam Line Break

In the staff's initial risk assessment, only one steam generator was assumed to be degraded to the extent that a tube would rupture in the event of a steam side depressurization event. This provided a reduction in the risk associated with those depressurizations by a factor of 0.25, because most depressurizations affect only one generator. The licensee did not take credit for this reduction in its risk assessment. It is most likely that U-bend apex cracking situations that result in the in-service rupture of a tube will show a "lead" generator in which the degradation is the worst. Hydro-testing of steam generator tubes in all four steam generators, following the tube failure, validated the "lead" generator assumption. Therefore, the staff has continued to apply the factor of 0.25 to the tube ruptures that are induced by steam side depressurizations.

7. Reduction of LERF from CDF

The licensee sorted its IPE core damage sequences due to spontaneous tube rupture according to whether the main steam line safety valve through which radioactivity is discharged to the atmosphere was modulating properly or was stuck open. If the valve was stuck open, the sequence was put in the LERF category; if it was modulating properly, the sequence was put in the category for successful containment. In its previous risk analyses, the staff has put high pressure core damage sequences with ruptured steam generator tubes in the large release category so long as the pressure was sufficient to open the steam line safety valves. The staff does not believe that the effects on radionuclide deposition in the secondary side of the steam generator due to the modulating valve would reduce the amount of radioactivity ultimately released sufficiently to make the event appear to be more like a contained core damage accident than an accident with a large early release. This is the

major effect stated by the licensee. However, the licensee also stated that the thermal-hydraulic calculations of core damage accidents performed to support its IPE showed that proper operation of the steam safety valves caused reactor coolant system (RCS) pressure to remain high enough that the RCS eventually burst by creep failure inside the containment. The staff believes that a break in the RCS boundary, if it occurs, could reduce the amount of radioactivity released to the atmosphere sufficiently to move an accident sequence out of the LERF category. However, the thermal-hydraulic calculations were performed some time ago with the MAAP computer code version 3.0B rev16, which the NRC staff has previously found to produce results that differ substantially from the results of current NRC codes for this type of analysis. It is also unclear that a steam safety valve that was modulating properly would continue to do so when the gas passing through it became very much hotter than its design temperature. Therefore, the staff analysis does not credit this factor for LERF reduction (LERF reduction factor assumed in ConEd's analysis ~ 0.13), but does consider it as an element in its sensitivity study.

8. Tube Ruptures During Core Damage Sequences with Causes Other Than Tube Degradation

As discussed in NUREG-1570, core damage sequences caused by events such as station blackout can be changed from the non-LERF to the LERF category by failure of degraded steam generator tubes during the accident sequence. There are two potential causes for tube failure. One is the potential for increased differential pressure to cause tube rupture if the steam side of a steam generator becomes depressurized while the reactor is still pressurized. Previous risk assessments have applied probabilities that a steam generator would depressurize due to a stuck-open safety valve on the steam line. In addition, because steam side leak tightness is not normally tested in pressurized water reactors, there is little assurance that a steam generator would remain pressurized, once it has evaporated all of its water inventory, even if all valves were nominally "closed." This also means that there is only anecdotal experience to provide data on the probability that a steam generator will depressurize when empty. Indian Point Unit 2 has provided some of the previous anecdotal experience. It also had some indication of leakage into the steam line during the February 15th event. And, after some valve work induced more leakage, IP2 was unable to pressurize the secondary side of the steam generator for a test during the outage. However, the licensee's risk assessment declined to add to the LERF category for these sequences, citing the probability of 0.018 used in NUREG-1150 for conditional tube failure probability during SBO core damage sequences. Considering the staff's estimate of 1.0 for the conditional probability of tube rupture in the event of an elevated pressure differential, the staff believes that this is a significant omission from the licensee's analysis.

The other potential cause for steam generator tubes to rupture during core damage sequences is that the tubes may be subjected to very high temperatures as the core melts. This would weaken the tube material and may lead to a rupture if the tube is sufficiently degraded. The conditions found by previous analyses to be necessary for this to occur are high reactor pressure and a dry, depressurized steam generator. These are called the "high/dry" core damage sequences, with depressurized secondary. This was discussed at the regulatory conference. The licensee's consultant stated that the short radius U-bend tubes are located in a region of the tube bundle that is not expected to experience the highest temperature during these accidents. The staff pointed out that, in one of 4 transient tests conducted in a 1/7th scale model, tubes in the region of the tube bundle that contains the tube that failed at IP2 on February 15th were in the portion of the tube bundle that received the hot flow. The licensee responded that, although they were in the hot gas flow path, they were substantially cooler than the hottest tubes and would not be expected to exceed 800 Kelvin (K) at the U-bend region where the crack was located. On that basis, the licensee concluded that the material would not weaken enough to result in tube failure. The staff has checked this assertion and estimates that the tube apex temperature could reach about 850 K, provided that the steam generator is depressurized and the flow pattern is as depicted in that particular transient test. This temperature would reduce the material strength by about 20%. Although the tube that failed on February 15th would be expected to fail if material strength was reduced by 20%, the depressurized condition of the steam generator that would allow that temperature increase by itself would have led to failure of that tube. Analyses with the steam generator still pressurized result in lower tube temperatures. Therefore, the staff agrees with the licensee that the potential for thermally-induced rupture is not a substantial consideration for this risk assessment.

So, the staff concludes that it is appropriate to disregard the potential for thermally-induced tube failures for sequences in the base CDF. But, it is necessary to consider the potential for pressure-induced failures to change some of the non-LERF base CDF sequences to LERF sequences, increasing the total LERF contribution associated with the tube degradation.

Final Staff Risk Assessment

Initiating Event Frequencies

The staff analysis is based on the position that the licensee's failure to identify the inadequacy of its tube inspection process for control of degradation by apex cracking would eventually lead to a tube failure event while in operation. For cases where an apex crack was found by inspection before one had failed in service, it is assumed that another cycle would begin without adequate inspection. Indeed, that was what occurred in 1997. Therefore, the probability that an in-service event would eventually occur is taken to be approximately one.

Thus, the issue becomes what the probabilities are for each type of potential in-service failure. As described in the previous section, the staff does not find the licensee's Monte Carlo analysis provides this information. The staff used the existing experience base to assign probabilities for the two different steam generator tube rupture leak rates. None of the in-service failures of apex cracks to date has been a leakage event that could be considered to "leak before break" in a manner that would allow the reactor operators to avoid the imminent break. One break has produced a flow rate above 225 gpm, and the other two have produced flow rates approximately in the middle of the 75 to 225 gpm range. Therefore, the probabilities utilized by the staff in its analysis are 0.33 for breaks above 225 gpm and 0.67 for breaks between 75 and 225 gpm.

The frequency that the staff considers in the significance determination process is the worst annualized frequency attained if the event is protracted and worsens over multiple years. Therefore, the staff has assumed a total frequency of one in-service failure event in the last year of operation.

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For the frequency of generator steam-side depressurization events that might induce a tube rupture, the licensee did not dispute the frequency the staff used in its initial risk assessment. However, the licensee did state that its analysis assumed the differential pressure in those events would not exceed 1800 psid. That would affect both the frequency of the higher pressure differential condition and the estimation of the period of operation during which a degrading tube would be susceptible to rupture. The staff has reviewed the issue and agrees that the combination of frequency and period of susceptibility used in our initial assessment is probably too conservative. Ideally, the probability distribution of events as a function of pressure differential should be combined with the rate of declining tube strength over time to arrive at a net frequency for burst due to those events. However, the currently available data base is not designed to facilitate that analysis. So, for this final analysis, the staff has chosen a simplified approximation using the frequency of high differential pressure events from NUREG-0844 and the period of susceptibility to that pressure.

Conditional Probabilities for Tube Rupture Induced by Steam Generator Depressurization

As discussed in the preceding section, the staff's final analysis will consider only the depressurization events that are expected to create differential pressure across the stream generator tubes near 2200 psid. Under those conditions, an apex flaw that would eventually fail within a year at normal service conditions is expected to be weak enough already to fail under the depressurization transient conditions assuming average values for the crack length and depth growth rates documented in the Condition Monitoring Operational Assessment (CMOA) report. Therefore, the staff sets the conditional rupture probability to unity for the last year of operation. Therefore, the probabilities utilized by the staff in its analysis are 0.33 for breaks above 225 gpm and 0.67 for breaks between 75 and 25 gpm.

Human Error Probabilities and Conditional Probabilities for Core Damage

The licensee's conditional probability for core damage given spontaneous tube rupture >225 gpm is in close agreement with the staff's value for 600 - 800 gpm events. In the staff's analysis, this value is dominated by human error probabilities. Therefore, it is reasonable to expect a substantial reduction for SGTR events below 225 gpm. However, the staff has no independent analysis to provide a value to quantify the risk. So, the licensee's value is used, with caution in interpreting the results.

For tube ruptures induced by steam line breaks, the conditional core damage frequency the staff used in its initial analysis was 1×10^{-2}, based on analyses described in INEL-95/0641 for MSLB events with 1 failed tube. The range of human error probabilities in that document is broad, and it does cover the licensee's values of 1 and 2.5×10^{-3}. That makes this part of the quantification very uncertain and subject to debate. However, the staff will continue to use its initial value for the large break case. For the smaller break case, the staff will adopt the licensee's value of 1×10^{-3}.

Staff Results

For spontaneous and MSLB-induced ruptures, the staff CDF contributions are:

for SGTR >225 gpm:	$0.33/yr \times 7.75 \times 10^{-5}$	$= 2.56 \times 10^{-5}/ry$
for SGTR between 75 and 225 gpm:	$0.67/yr \times 2.90 \times 10^{-6}$	$= 1.94 \times 10^{-6}/ry$
for MSLB with SGTR >225 gpm:	$0.001/yr \times 0.33 \times 0.25 \times 1 \times 10^{-2}$	$= 8.25 \times 10^{-7}/ry$
for MSLB with SGTR between 75 and 225 gpm:	$0.001/ry \times 0.67 \times 0.25 \times 1.0 \times 10^{-3}$	$= 1.67 \times 10^{-7}/ry$
		$2.85 \times 10^{-5}/ry$ total CDF

As in previous analyses, the staff estimates that the LERF contribution from these sequences is equal to this CDF contribution as described in item 7 above.

In addition, it is necessary to estimate the LERF that would result from steam generator depressurization-induced tube ruptures during other core damage accidents, such as those caused by station blackout events. Those events are estimated as the "high/dry" portion of the core damage frequency times the probability that the steam generator is depressurized. The total core damage frequency estimated in the Indian Point unit 2 IPE was $3.13 \times 10^{-5}/ry$, but the licensee has not tabulated the "high/dry" portion of their core damage frequency. Based on its experience with other pressurized water reactors, the staff expects the Indian Point unit 2 "high/dry" frequency to be in the range between $1 \times 10^{-5}/ry$ and $2 \times 10^{-5}/ry$.

Estimation of the fraction of these events with a depressurized generator is highly speculative. NUREG-1150 estimated probabilities that one or more generators would depressurize as 0.74 and 0.05 for Surry and Sequoyah, respectively, based on procedural differences. NUREG-1570 added the concern about depressurization of dry generators by leakage through nominally "closed" valves. But, because no leak rate tests are required for main steam isolation in pressurized water reactors, only anecdotal data is available from events where leakage was large enough to affect normal plant operations. As a sensitivity study, NUREG-1570 added 0.50 as the probability for one or more of the isolated generators depressurizing by leakage.

Severe accident management guidelines have been implemented to refill dry steam generators when core damage seems imminent. For the "high/dry" core damage sequences, feedwater is usually not available, so procedures require depressurizing the generators one-by-one and filing them with water from low pressure sources. This alone could bring the conditional probability of depressurization to near 1.0 for the high/dry sequences[2]. For this analysis, the staff assumed that only one generator is sufficiently degraded to burst if depressurized. The staff has not yet calculated the thermal-hydraulic response of the reactor coolant system caused by adding cold water to a dry steam generator

during a core damage accident. Phenomena such as condensing steam to depressurize the RCS, voiding the RCS loop seals by evaporation when the RCS depressurizes and allowing full-loop circulation of hot steam through the unfilled generators, and the repressurization effects when the accumulators discharge water onto hot RCS and core surfaces are too difficult to predict without detailed analysis. Therefore, it is not currently possible to predict what the effects would be on a still-pressurized generator with a severely degraded tube if one or more of the other generators was depressurized and successfully filled. But, there is a probability of 0.25 that the degraded generator would be the first to be depressurized, in which case the tube would fail. Therefore, the staff assumes that the conditional probability of the degraded generator becoming depressurized during "high/dry" core damage sequences is in the range 0.25 to 1.0.

When the conditional probability of depressurization for the degraded generator is applied to the expected "high/dry" frequency of 1×10^{-5}/ry to 2×10^{-5}/ry, the results are in the range 2.5×10^{-6}/ry to 2.0×10^{-5}/ry. This gives a total LERF estimate for the staffs analysis as:

LERF from additional CDF	=	2.85×10^{-5}/ry
LERF from "high/dry" base CDF	=	2.5×10^{-6}/ry to 2.0×10^{-5}/ry
		total LERF = 3.10×10^{-5}/ry to 4.85×10^{-5}/ry

This result is well above the "Red/Yellow" threshold value of 1×10^{-5}/ry threshold used in the significance determination process.

Sensitivity Study

As a sensitivity study, the staff also analyzed a case crediting the licensee's distinction between LERF and non-LERF sequences. Using the licensees conditional LERF probabilities, these results become:

for SGTR >225 gpm:

$$0.33/\text{yr} \times 7.75 \times 10^{-5} \times 0.13 = 3.32 \times 10^{-6}/\text{ry}$$

for SGTR between 75 and 225 gpm:

$$0.67/\text{yr} \times 1.60 \times 10^{-6} = 1.07 \times 10^{-6}/\text{ry}$$

for MSLB with SGTR >225 gpm:

$$0.001/\text{yr} \times 0.33 \times 0.25 \times 1 \times 10^{-2} = 8.25 \times 10^{-7}/\text{ry}$$

for MSLB with SGTR between 75 and 225 gpm:

$$0.001/\text{ry} \times 0.67 \times 0.25 \times 1.0 \times 10^{-3} = 1.67 \times 10^{-7}/\text{ry}$$

5.38×10^{-6}/ry LERF from CDF crediting licensee's reduction factors

So, if the staff also credits the licensee's basis for considering 83% of the CDF from spontaneous ruptures to create releases too low to be in the LERF category, then the sum of the LERF contributions for all sequences considered by the licensee would be below 1×10^{-5}/ry. However, as discussed above, the licensee did not include any consideration of the additional LERF that would result from steam generator depressurization-induced tube ruptures during other core damage accidents, such as those caused by SBO events. In the staff's base case analysis, above, that contribution was estimated in the range of 2.5×10^{-6}/ry to 2×10^{-5}/ry. Including that contribution, the corresponding sensitivity case LERF results is:

LERF from additional CDF	= 5.38×10^{-6}/ry
LERF from "high/dry" base CDF	= 2.5×10^{-6}/ry to 2.0×10^{-5}/ry
low sensitivity study total LERF	= 7.88×10^{-6}/ry to 2.54×10^{-5}/ry

Thus, the range of results for the sensitivity case include the numerical threshold for the "Red/Yellow" determination, with the larger portion of the range on the "Red" side. From this sensitivity case, the staff concludes that the question about the reduction in radiological releases created by a functioning steam line safety valve could be important when a plant is known to have a low "high/dry" component of its base CDF plus a high probability of maintaining the degraded steam generator secondary in a pressurized condition until the RCS fails inside the containment. However, the licensee did not address those factors in its response to the staff's initial risk assessment. Therefore, on the basis of the information available, the staff concludes that it is most probable that a LERF contribution above 1×10^{-5}/ry will occur for a year during which a steam generator is degrading severely enough to allow a tube to rupture during normal operation.

Staff Conclusion

The foregoing staff review and analysis has estimated that, when all contributions to LERF are considered, the condition being assessed is

most likely to remain in the "Red" category, with its LERF increment above the 1 x 10^{-5} threshold. This is true even when considerable credit is given for reduced human error probabilities for the smaller break size events and the licensee's rational is credited for taking much of the spontaneous rupture CDF contribution out of the LERF category. On this basis, the staff concludes that the result of its final risk evaluation is best quantified as a "Red" result.

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1. Also, it should be noted for clarity that the factor of 0.13 between CDF and LERF noted in the licensee's presentation on September 26th actually was applied by them only to the spontaneous ruptures with break flows >225 gpm. They applied a factor of 0.55 for spontaneous ruptures between 75 and 225 gpm, and credited no reductions for ruptures induced by steam side depressurization events.

2. It is of interest to note that the same procedure, if reliably applied to the sequences that reach core damage because of tube failure, might eliminate LERF from many of those sequences. This is because high pressure feedwater is available for many of those sequences. It only has been isolated from the ruptured generator in accordance with the emergency operating procedures. Consequently, consideration of the severe accident guidelines that were implemented after and not credited by the licensee's IPE, could change the situation with respect to which sequences would contribute the most to LERF, but still would be expected to produce a LERF contribution above 1 x10^{-5}/ry.

November 9, 2000

SDP/EA-00-137

Duke Energy Corporation
ATTN: Mr. W. R. McCollum
Vice President
Oconee Nuclear Station
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Seneca, SC 29672

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT 50-269/00-011, 50-270/00-011, AND 50-287/00-011, OCONEE NUCLEAR STATION)

Dear Mr. McCollum:

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 (i.e., an issue with low to moderate increased importance to safety, which may require additional NRC inspections). This White finding involved the reduced ability to provide reactor coolant makeup following tornados of such intensity that the 4160 volt electrical buses within the turbine building and the Borated Water Storage Tank are damaged (without damaging the Keowee hydro-electric units), accompanied by the failure of the Standby Shutdown Facility either through independent failure or due to the tornado. In the postulated dominant accident sequences, the licensee's analysis assumed that a High Pressure Injection (HPI) pump taking suction from the spent fuel pool (SFP) would provide a success path to prevent core damage. However, around 1990, a performance deficiency involving a design calculation resulted in Duke Energy Corporation's (DEC) failure to recognize that the SFP would be available for a time significantly less than the originally assumed mission time.

At your request, an open regulatory conference was conducted with you and members of your staff on September 7, 2000, to discuss your views on this issue. Enclosure 2 lists the attendees at the regulatory conference. Enclosures 3 and 4 contain copies of the material presented by DEC and the NRC at the regulatory conference, respectively. During the meeting, your staff described your assessment of the significance of the findings and detailed corrective actions, including the root cause evaluations. DEC's risk assessment determined that the increase in core damage frequency (CDF) for Oconee Unit 1 was approximately 3E-6 per year. The increase in CDF for Units 2 and 3 was approximately 5E-7 per year. DEC stated that the increase in risk for Units 2 and 3 was less because these units have a different reactor coolant pump (RCP) seal package that is less susceptible to a seal failure. DEC agreed that the pertinent design calculation was in error, but also stated that the Oconee design and licensing bases do not include consideration of a RCP seal loss of coolant accident (LOCA).

After considering the information developed during the inspection and the information you provided at the conference, the NRC has concluded that the inspection finding for Oconee Unit 1 is appropriately characterized as White. DEC's risk assessment is also consistent with this characterization. In addition, as you indicated at the regulatory conference, there is a difference in the type of RCP seals for Units 2 and 3. The NRC accepts DEC's position that this difference results in a reduced core damage frequency to a value approximated by DEC at the conference. As such, this issue is appropriately characterized as Green (i.e., an issue of very low safety significance) for Units 2 and 3.

You have ten 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 3.

The NRC has also determined that your failure to adequately consider design inputs to assure that the design basis was translated into specifications, drawings, procedures, and instructions for the HPI system using the SFP as a suction source following a tornado is a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The violation is cited in the enclosed Notice of Violation (Notice), and the circumstances surrounding it are described in detail in the subject inspection report. The NRC accepts DEC's position that a tornado induced RCP seal LOCA is outside the design basis of the facility. However, scenarios exist within the design basis of the facility which would require the SFP as a suction source for the HPI pump. As such, DEC's failure to adequately consider thermal-hydraulic design inputs into design calculations represents a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions - May 1, 2000," NUREG-1600 (Enforcement Policy), the Notice is considered escalated enforcement action because it is associated with a White finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in DEC's presentation materials provided at the regulatory conference (Enclosure 3). Therefore, you are not required to respond to this letter unless the description herein 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 issue 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 assessing this issue, the NRC recognizes that the vulnerability that resulted in the increased risk existed before 1990 is not reflective of DEC's current performance, the vulnerability was identified by your staff in 1998, and actions have been or will be implemented by DEC to address the vulnerability. Nonetheless, these factors do not change the overall risk significance of the issue and are not mitigating under the current Reactor Oversight Program (ROP). You should be aware, however, that at the end of the first year of implementation of the

ROP, the program will be evaluated for lessons learned from initial implementation, including the appropriateness of the various assessment process outcomes.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures, and your response (if you choose to provide one), 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). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR and PARS without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Luis A. Reyes
Regional Administrator

Docket Nos. 50-269, 50-270, 50-287, 72-04
License Nos. DPR-38, DPR-47, DPR-55, SNM-2503

Enclosures:
1. Notice of Violation
2. List of Attendees
3. Presentation material presented by DEC
4. Presentation material presented by NRC

cc w/encls:

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NOTICE OF VIOLATION

Duke Energy Corporation
Oconee Nuclear Station
Units 1, 2 and 3

Docket Nos. 50-269, 50-270, 50-287, 72-04
License Nos. DPR-38, DPR-47, DPR-55, SNM-2503
SDP/EA-00-137

During an NRC inspection conducted on June 28, 2000, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions - May 1, 2000," NUREG-1600, the violation is listed below:

10 CFR 50, Appendix B, Criterion III, Design Control, states in part that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions, and that design control measures shall be applied to items such as stress, thermal, hydraulic and accident analysis.

Updated Final Safety Analysis Report, Section 3.2.2, states in part that a sufficient supply of primary side makeup water is assured during a tornado initiated loss of offsite power by several backup systems.

a. The SSF Reactor Coolant Makeup Pump can take suction from the Spent Fuel Pool. The pump can be supplied power from the SSF Diesel.

b. A High Pressure Injection Pump can take suction from either the Borated Water Storage Tank or the Spent Fuel Pool. Either the "A" or "B" High Pressure Injection Pump can be powered from Keowee via the Auxiliary Service Water Pump Swithchgear.

Duke Topical Report 1-A states that the quality assurance program meets the requirements of ANSI 45.2.11 - 1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants."

ANSI 45.2.11, Section 3.2.4 states in part that design inputs include design conditions such as pressure and temperature and, Section 3.2.11 states that hydraulic requirements such as pump net positive suction head, allowable pressure drops, and allowable fluid velocities are design inputs.

Contrary to the above, as of April 1, 2000, measures had not been adequately established to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions, in that pressure, temperature or hydraulic requirements had not been adequately considered as design inputs for calculation OSC 3873, "Hydraulic Model of High Pressure Injection System with Suction from the Fuel Pool." (01013)

This violation is associated with a White SDP finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in an enclosure to this letter transmitting this Notice of Violation (Notice). 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 II within 30 days of the date of the letter transmitting this 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.

Because any 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/NRC/ADAMS/index.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 9th day of November 2000

LIST OF OPEN REGULATORY CONFERENCE ATTENDEES

NUCLEAR REGULATORY COMMISSION:
L. Reyes, Regional Administrator, Region II (RII)
B. Mallett, Deputy Regional Administrator, RII
C. Casto, Director, Division of Reactor Safety, RII
V. McCree, Deputy Director, Division of Reactor Projects (DRP), RII
R. Borchardt, Director, Office of Enforcement, OE
A. Boland, Enforcement Officer, RII
S. Sparks, Senior Enforcement Specialist, RII
C. Ogle, Chief, Branch 1, DRP, RII
W. Rogers, Senior Reactor Analyst, DRS, RII
M. Shannon, Senior Resident Inspector, Oconee, DRP, RIIL
C. Evans, Regional Counsel, RII
R. Carroll, Jr., Project Engineer, Branch 1, DRP, RII
J. Lenahan, Senior Reactor Inspector, Engineering Branch, DRS, RII
R. Schin, Senior Reactor Inspector, Engineering Branch, DRS, RII
D. Nelson, Senior Enforcement Specialist, OE (teleconference)
V. Ordaz, Senior Enforcement Coordinator, NRR, (teleconference)
D. LaBarge, Senior Project Manager, Office of Nuclear Reactor Regulation (NRR) (teleconference)
R. Emch, Chief, Project Directorate II, Division of Licensing Project Management, NRR, (teleconference)
P. Koltay, Inspection Program Branch, NRR, (teleconference)
P. Wilson, Probablistic Safety Assessment Branch, NRR, (teleconference)

DUKE ENERGY CORPORATION:
W. McCollum, Oconee Vice President
M. Nazar, Engineering Manager
L. Nicholson, Regulatory Compliance Manager
E. Burchfield, Design Supervisor
D. Brewer, Risk Analysis Manager

Enclosure 2

ENCLOSURE 3
PRESENTATION MATERIAL PRESENTED BY DEC
Is ADAMS accessible from the NRC Web site at
http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).
The accession number is ML 003768658.

OPEN REGULATORY CONFERENCE

OCONEE NUCLEAR STATION

SEPTEMBER 7, 2000, 1:00 P.M.
NRC REGION II OFFICE, ATLANTA, GEORGIA

I. OPENING REMARKS AND INTRODUCTIONS
L. Reyes, Regional Administrator

II. NRC REGULATORY CONFERENCE POLICY
A. Boland, Enforcement Officer

III. EMPHASIS ON MEETING INTENT
L. Reyes, Regional Administrator

IV. STATEMENT OF THE ISSUE WITH CURRENT RISK & VIOLATION PERSPECTIVE
C. Casto, Director, Division of Reactor Safety

V. LICENSEE RISK & REGULATORY PERSPECTIVE PRESENTATION

VI. BREAK / NRC CAUCUS
L. Reyes, Regional Administrator

VII. CLOSING REMARKS
L. Reyes, Regional Administrator

Enclosure 4

STATEMENT OF APPARENT VIOLATION

10 CFR 50, Criterion III, Design Control, states in part that "Measures shall be established to assure that applicable regulatory requirements and design basis ... are correctly translated into specifications, drawings, procedures, and instructions ... Design control measures shall be applied to items such as ... stress, thermal, hydraulic and accident analysis ... "

Updated Final Safety Analysis Report, section 3.2.2, states in part that "... a sufficient supply of primary side makeup water is assured during a tornado initiated loss of offsite power by several backup systems.

c. The SSF Reactor Coolant Makeup Pump can take suction form the Spent Fuel Pool. The pump can be supplied power from the SSF Diesel.

d. A High Pressure Injection Pump can take suction from either the Borated Water Storage Tank or the Spent Fuel Pool. Either the "A" or "B" High Pressure Injection Pump can be powered from Keowee via the Auxiliary Service Water Pump Swithchgear."

Duke Topical Report 1-A states that the quality assurance program meets the requirements of ANSI 45.2.11 - 1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants."

ANSI 45.2.11, section 3.2.4 states in part that design inputs include design conditions such as pressure and temperature and, section 3.2.11 states that hydraulic requirements such as pump net positive suction head, allowable pressure drops, and allowable fluid velocities are design inputs.

As of April 1, 2000, measures had not been adequately established to assure that applicable regulatory requirements and design basis were correctly translated into specifications, drawings, procedures, and instructions in that pressure, temperature or hydraulic requirements had not been adequately considered as design inputs for calculation OSC 3873, "Hydraulic Model of High Pressure Injection System with Suction from the Fuel Pool."

Note: The apparent violation discussed at this Regulatory Conference is subject to further review and is subject to change prior to any resulting enforcement action.

October 27, 2000

EA-00-163

Tennessee Valley Authority
ATTN: Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: NOTICE OF VIOLATION AND EXERCISE OF ENFORCEMENT DISCRETION, BROWNS FERRY NUCLEAR PLANT (NRC OFFICE OF INVESTIGATIONS REPORT NO. 2-1999-028 AND INSPECTION REPORT NOS. 50-259/00-03, 50-260/00-03, 50-296/00-03)

Dear Mr. Scalice:

This is in reference to an investigation conducted by the Nuclear Regulatory Commission's (NRC) Office of Investigations (OI) between September 21, 1999 and June 15, 2000, and an inspection completed on June 24, 2000. The purpose of the OI investigation and inspection was to review the circumstances involving an individual formerly employed by the Tennessee Valley Authority's (TVA) Browns Ferry Nuclear Plant (BFN), who failed to perform measuring and test equipment (M&TE) nonconformance evaluations in accordance with Technical Specification required site procedures.

The results of the OI investigation and inspection were formally transmitted to you by letter dated July 27, 2000. Our letter also provided you the opportunity to either respond to the apparent violation in writing or request a predecisional enforcement conference. By letter dated August 25, 2000, you responded to the apparent violation and addressed the root causes and your corrective actions to prevent recurrence. We have reviewed the information you provided and have concluded that sufficient information is available to determine the appropriate enforcement action in this matter.

Based on the information developed during the investigation and our review and follow-up of the information provided in your response of August 25, 2000, the NRC has determined that a violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation (Notice), and the circumstances surrounding it are described in detail in the subject inspection report. The violation involves the failure to adhere to TVA procedures as required by Technical Specification 5.4.1, related to out-of-tolerance M&TE. During a self-assessment in June 1999, your staff identified that certain procedurally required actions had not been taken in response to numerous out-of-tolerance M&TE. Site procedures required that upon being informed (in this case, by TVA's Central Laboratory Field Testing Services) that M&TE was out-of-tolerance, the M&TE Program Administrator was required to issue and/or disposition non-conformance evaluations for those plant components tested or inspected using the out-of-tolerance M&TE. The purpose of a nonconformance evaluation is, among other reasons, to initiate the site review process to ensure that plant components have not been negatively affected by the out-of-tolerance M&TE, and to initiate action to address plant components that have been affected. Your review determined that from June 1997 to June 1999, approximately 500 nonconformance evaluations were not properly issued and/or dispositioned. After TVA identified the full scope of unprocessed nonconformance evaluations, your staff's reevaluation of all nonconformance evaluations determined that plant component operability, and in particular, safety related component operability, was unaffected. The NRC's OI investigation concluded that the failure of the M&TE Program Administrator to issue and/or disposition nonconformance evaluations for the out-of-tolerance M&TE was deliberate. Based on the deliberate aspects of this matter, the violation has been categorized at Severity III, in accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions - May 1, 2000" (Enforcement Policy), NUREG-1600.

In accordance with the Enforcement Policy, a base civil penalty in the amount of $55,000 is considered for a Severity Level III violation. Because your facility has not been the subject of escalated enforcement action in the past 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 letter of August 25, 2000, described your corrective actions in response to this issue, which included: (1) an evaluation of the impact of the unprocessed nonconformance evaluations on plant equipment; (2) an overall assessment of the M&TE program to examine compliance with internal governing documents and policies; (3) increasing the awareness of M&TE issues at the site by a weekly review of nonconformance evaluations during the Plan-Of-The-Day meeting; (4) a briefing of senior site managers on the circumstances of this particular violation; (5) the development of a new training course for first line supervisors to improve their awareness of the potential for the type of behavior encountered in this particular instance, and methods of detection; and (6) inclusion of the M&TE program in the BFN Programs and Process Core Assessment program, which will re-assess the site M&TE program in June of 2001.

As stated in our July 27, 2000 letter, the NRC initially was concerned that management oversight of the M&TE program failed to detect this situation during the two year period it was occurring, and requested that TVA address any management oversight aspects of this issue. Your response referenced, as background information, the performance of M&TE periodic status reports. The purpose of these status reports, as you indicated, was to provide feedback to management as a means to ensure that all nonconformance evaluations were being properly investigated in a timely manner. As stated in your letter, four status reports were performed by the M&TE Program Administrator from 1997 to 1999 and were submitted to his management superiors. Based on our subsequent review and follow-up of the four audit reports, we have identified that the audit reports contained only one month of nonconformance evaluation data. Based on the NRC's discussions with TVA staff, it appears that the intent of these audit reports was to provide six months of nonconformance data to plant supervision. However, this discrepancy was apparently not recognized by TVA supervision at the time the audits were provided to them, nor was the discrepancy identified during TVA's

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review of this matter. The NRC has concluded that BFN supervisory review of the audit reports during the time of the M&TE Program Administrator's deliberate misconduct was inadequate, in that the review failed to consider that a significant quantity of nonconformance evaluation data was not submitted for review. An adequate review of the audit reports and the work of the M&TE Program Administrator may have provided management a possible opportunity to identify the violation earlier.

More importantly, your response of August 25, 2000, indicated that an overall assessment of the M&TE program was conducted to examine compliance with internal governing documents and policies. However, the omitted data issue was not addressed by TVA. The failure to recognize the omitted data supports a conclusion that your review of the factors stemming from the violation was not adequate to fully identify management oversight deficiencies. As such, this management oversight failure was not considered in the development of TVA's corrective actions for this issue. Therefore, the NRC determined that credit is not warranted for the factor of *Corrective Action*.

This assessment normally would result in a proposed civil penalty of $55,000. However, the NRC may refrain from proposing a civil penalty for a Severity Level III violation in accordance with Section VII.B.6 of the Enforcement Policy. In this case, the NRC has concluded that the underlying safety significance involving the unprocessed nonconformance evaluations was low, because safety related plant component operability was unaffected. Therefore, with approval of the Director, Office of Enforcement, and in consultation with the Deputy Executive Director for Reactor Programs, I have been authorized to propose that no civil penalty be assessed in this case. However, significant violations in the future could result in a civil penalty. Please note that an Order Prohibiting Involvement in NRC Licensed Activities (Effective Immediately) has been issued to the former employee responsible for the deliberate misconduct. A copy of the Order is enclosed for your information.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures, TVA's letter dated August 25, 2000, and your response to the Notice 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). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR and PARS without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

If you have any question regarding this matter, please contact Paul Fredrickson, Chief, Projects Branch 6 at 404-562-4530.

Sincerely,

/RA/

Luis A. Reyes
Regional Administrator

Docket Nos. 50-259, 50-260, 50-296
License Nos. DPR-33, DPR-52, DPR-68

Enclosure: Notice of Violation

cc w/encl:

Karl W. Singer
Senior Vice President
Nuclear Operations
Tennessee Valley Authority
Electronic Mail Distribution

Jack A. Bailey, Vice President
Engineering and Technical Services
Tennessee Valley Authority
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John T. Herron
Site Vice President
Browns Ferry Nuclear Plant
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R. J. Adney, General Manager
Nuclear Assurance
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General Counsel
Tennessee Valley Authority
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Ashok S. Bahtnager, Plant Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
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Mark J. Burzynski, Manager
Nuclear Licensing
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Timothy E. Abney, Manager
Licensing and Industry Affairs
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

State Health Officer
Alabama Dept. of Public Health
RSA Tower - Administration
Suite 1552
P. O. Box 303017
Montgomery, AL 36130-3017

NOTICE OF VIOLATION

Tennessee Valley Authority
Browns Ferry Nuclear Plant

Docket Nos.: 50-259, 50-260, 50-296
License Nos. DPR-33, DPR-52, DPR-68
EA-00-163

During an investigation conducted by the Nuclear Regulatory Commission's (NRC) Office of Investigations (OI) between September 21, 1999 and June 15, 2000, and an inspection completed on June 24, 2000, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions - May 1, 2000," NUREG-1600, the violation is listed below:

> Technical Specification 5.4.1 requires written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A, Section 8, specifically addresses procedures for control of measuring and test equipment. Appendix A further states that procedures of a type appropriate to the circumstances should be provided to ensure that tools, gauges, instruments, controls, and other measuring and test devices are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy.
>
> Browns Ferry Site Standard Practice Procedure (SSP)-6.7, Control of Measuring and Test Equipment (M&TE), Revision 8A, Effective May 27, 1997 through June 1, 1998, Step 3.14.A, states that nonconformance evaluations shall be issued for the following conditions: lost M&TE or standards, out-of-tolerance M&TE or plant standards, damaged or otherwise defective M&TE or plant standards, and disassembled M&TE or plant standards. Step 3.14.E states that all nonconformance evaluations should be completed within 30 calendar days of the site receipt of the initiating document. An extension of up to ten calendar days may be approved by the Plant Manager, or designee, using Appendix H.
>
> Tennessee Valley Authority Standard Programs and Processes Procedure (SPP)-6.4, Measuring and Test Equipment, Revision 0, Effective May 29, 1998, Step 3.15.1, states that nonconformance evaluations shall be issued to determine the validity and acceptability of previous work for the following conditions: lost M&TE or standards, out-of-tolerance M&TE or plant standards, damaged or otherwise defective M&TE or plant standards, and disassembled M&TE or plant standards. Step 3.15.6 requires all nonconformance evaluations be completed within 30 calendar days of the site receipt of the initiating document.
>
> Contrary to the above, during the period from June 2, 1997, to June 14, 1999, SSP-6.7 and SPP-6.4 were not implemented, in that approximately 500 nonconformance evaluations were either not issued or completed for measuring and test equipment which had been identified as out-of-tolerance or otherwise meeting the criteria for evaluation. (01013)

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

Pursuant to the provisions of 10 CFR 2.201, Tennessee Valley Authority 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 II, and a copy to the NRC Resident Inspector at the Browns Ferry Nuclear Plant, 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: (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 previously 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/NRC/ADAMS/index.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 27th day of October 2000

ACRS BRIEFING
ON
THE POWER UPRATE PROGRAM

December 6, 2000

T. J. Kim, Lead Project Manager, NRR

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AGENDA

- INTRODUCTION ..... R. BARRETT, NRR
- PROGRAM OVERVIEW & STANDARD REVIEW PLAN ..... T. J. KIM, NRR
- APPLICATION OF RISK-INFORMED DECISION MAKING ON POWER UPRATES ..... M. RUBIN, NRR
- USE OF MORE "REALISTIC" ANALYSES IN SUPPORT OF POWER UPRATES ..... R. CARUSO, NRR
- POTENTIAL SYNERGISTIC EFFECTS:
  - HIGH BURN-UP FUEL/UPRATE CONDITIONS ..... R. CARUSO, NRR
  - ACCELERATED EROSION/CORROSION DUE TO UPRATE/AGING ..... E. CARPENTER, NRR
- RES PERSPECTIVES ON EXTENDED POWER UPRATE ..... F. ELTAWILA, RES
- SUMMARY/CONCLUSION ..... R. BARRETT, NRR

POWER UPRATE PROGRAM - OVERVIEW

- CAPACITY RECAPTURE FOR THE POL PLANTS
- 5% "STRETCH" POWER UPRATES
- "EXTENDED" POWER UPRATES OF 6-8%
- "EXTENDED" POWER UPRATES OF 15-20%
- MAINE YANKEE LESSONS LEARNED
- STANDARD REVIEW PLAN SECTION ON POWER UPRATES

POWER UPRATE PROGRAM - CAPACITY RECAPTURE FOR THE POL PLANTS

- HADDAM NECK (24%) IN 1969
- OYSTER CREEK (14%) IN 1971
- PALISADES (15%) IN 1977
- GINNA (17%) IN 1984
- MAINE YANKEE (8%) IN 1978, FOLLOWED BY (2.5%) IN 1989
- INDIAN POINT 2 (11%) IN 1990

POWER UPRATE PROGRAM - 5% "STRETCH" UPRATES

	BWRs	PWRs
1977		CALVERT CLIFFS 1&2
1979		ROBINSON MILLSTONE 2
1980		FORT CALHOUN
1981		ST. LUCIE 1
1985	DUANE ARNOLD	ST. LUCIE 2
1986		SALEM 1 *(2%)* NORTH ANNA 1 & 2
1988		CALLOWAY TMI 1 *(1.3%)*
1992	GE 5% TOPICAL REPORT FERMI ***(ACRS REVIEW)***	
1993		VOGTLE 1 & 2 WOLF CREEK
1994	SUSQUEHANNA 1& 2 PEACH BOTTOM 2 & 3	
1995	LIMERICK 2 WNP2 NMP2 HATCH 1 & 2	SURRY 1 & 2
1996	LIMERICK 1	TURKEY POINT 3 & 4 PALO VERDE 2 & 3 *(2%)*
1997	BRUNSWICK 1 & 2 FITZPATRICK BROWNS FERRY 2 & 3	FARLEY 1 & 2
1999	LASALLE 1 & 2 PERRY	
2000	RIVER BEND	DIABLO CANYON 1 *(2%)*

POWER UPRATE PROGRAM - EXTENDED POWER UPRATES OF 6-8%

BWRs	PWRs
▸ GE TOPICAL REPORT (UP TO 20%)* - 1998	
▸ MONTICELLO (6.3%)* - 1998	**NONE.**
▸ HATCH (8%)* - 1998	

* GE TOPICAL REPORTS "ELTR-1" AND "ELTR-2" ADDRESSING POWER UPRATES OF UP TO 20%, AS WELL AS MONTICELLO & HATCH UPRATES WERE REVIEWED BY THE ACRS.

POWER UPRATE PROGRAM - EXTENDED POWER UPRATES OF 15-20%

▸ DUANE ARNOLD	(15%)	APPLICATION RECEIVED ON 11/17/2000*
▸ DRESDEN 2 & 3 QUAD CITIES 1 & 2	(17%)	APPLICATION EXPECTED IN JANUARY 2001
▸ BRUNSWICK 1 & 2	(15%)	APPLICATION EXPECTED IN MID-2001
▸ CLINTON	(20%)	APPLICATION EXPECTED IN MID-2001

* THE LICENSEE HAS REQUESTED COMPLETION OF STAFF REVIEW BY MID-2001 TO ACCOMMODATE THE LICENSEE'S IMPLEMENTATION SCHEDULE.

POWER UPRATE PROGRAM - MAINE YANKEE LESSONS LEARNED:

- ENSURE APPROPRIATE USE OF ANALYTICAL METHODOLOGIES AND COMPUTER CODES
- CONSISTENCY IN REVIEW AREAS COVERED:
  - HUMAN FACTORS
  - GRID STABILITY
  - FUEL POOL COOLING
  - BOP EQUIPMENT
  - MOVs
- CONSIDER DEVELOPING A SRP SECTION FOR POWER UPRATE REVIEWS
- STAFF SAFETY EVALUATIONS FOR THE FARLEY UPRATE IN LATE 1997 AND THE MONITCELLO UPRATE IN 1998 HAVE FULLY INCORPORATED THE MAINE YANKEE LESSONS LEARNED ISSUES, AND HENCE FORMED TEMPLATES FOR THE SUBSEQUENT UPRATE REVIEWS.
- THE STAFF HAS RECOMMENDED THAT THE COSTS OF DEVELOPING A SRP SECTION FOR POWER UPRATE REVIEWS OUTWEIGHS THE POTENTIAL BENEFITS AT THIS POINT IN THE PROGRAM.

RISK-INFORMED REVIEW CONSIDERATIONS OF EXTENDED POWER UPRATE APPLICATIONS

by

Mark P. Rubin
Probabilistic Safety Assessment Branch
Division of Systems Safety & Analysis
Office of Nuclear Reactor Regulation

General Perspective on Extended Power Uprates

- Extended power uprate applications are not requesting relaxation of any deterministic requirements.
- Licensee requests for power uprates are required to meet all deterministic requirements.
- Nevertheless, any request to operate at power significantly beyond "stretch power" represents the potential for reduction of plant margin and increase in risk.
- Therefore, for Monticello and Hatch submittals, staff believed it would be prudent to evaluate risk to see if the risk profile of the plant is changed in any significant way or if new vulnerabilities are introduced.
- Decisions on considering risk impact for future uprates:
  - Will be guided by policy in SECY 99-246
  - Will consider size of uprate, baseline CDF/LERF of plant and insights from deterministic evaluation such as DBA margins reduction, setpoint changes and fluid conditions, operational insights available from previous uprates, and risk of expected activity release.

Risk-Informed Assessment for Monticello and Hatch

- RG 1.174 provided a sound framework to assess power uprate impacts.

- Areas that were addressed by licensees for Monticello and Hatch reviews

    1. Initiating event frequency
    2. Equipment/component failure rates
    3. Operator error probability
    4. Success criteria

- Areas 3 and 4 were amenable to modeling and assessment in the licensees' risk model. Changes in timing available for operator actions due to power uprates and changes to success criteria were considered and reflected in modified CDF/LERF estimates. Only minor impacts were observed.

- More difficult to assess potential impact on initiating event frequencies and failure rates.
  -- For Monticello and Hatch uprates, these areas were qualitatively assessed to have minimal impact.

- Staff did not identify any related concerns that revised risk insight.

Potential for Synergistic Effects Beyond Operator Timing and Success Criteria

- Staff conclusion on Monticello and Hatch uprates:
  - Solid deterministic basis
  - First order (synergistic) risk impacts well understood and modeled explicitly (timing and success criteria)
- Potential exists for some "unexpected" impacts in areas in initiating frequency and SSC failure rates. Difficult to predict them in absence of operational data.
- These will likely be relatively small secondary impacts
  - Acceptable deterministic analysis for SSCs to operate in uprate conditions
  - Significant changes in initiating frequencies would be self-revealing
  - Significant changes in unavailability for normally operating equipment would be self-revealing. (Recirc pump vibration induced failure).
  - Standby equipment, which may not reveal availability reductions, will often not be subjected to full uprate conditions (partial isolation from RCS and secondary), therefore may not be as subject to degradation.

Challenges for Incorporation of Broader Synergistic Impacts

- Risk models do not currently include provision for assessing SSC reliability impact for changes in operating condition, or impacts on IE frequencies.
- No models for passive system degradation, impacts on pipe break frequencies.
- Though uncertain of causality, examples of uprate-induced impacts have been reported.

Initial Thoughts on Significance of Unmodeled Synergistic Effects

- Adherence to Maintenance Rule will provide reasonable feedback and corrective action. Performance problems for active SSCs will be identified.
- Passive system/component degradation may be identified in maintenance rule, but with somewhat less confidence than active components.
- Degradation of standby components may not be as readily identified due to lower challenge frequency, but may be less impacted due to reduced exposure to uprated parameters (temps, pressures, flows, vibration)
- Due to focus and sensitivity on pipe failures, degradation mechanisms related to power uprates will likely be fully explored and appropriate corrective actions identified and/or augmented inspection activities implemented.
- Due to sensitivity to plant level transients, mechanisms and degradation that lead to plant trips get significant attention and have root cause explored with corrective actions taken as necessary.

Initial Thoughts on Significance of Unmodeled Synergistic Effects
(Continued)

- With emphasis on utilization of as-built as-operated PRAs, staff would expect utilities implementing risk-informed activities to reflect operational data that justifies changes in initiating frequencies and SSC availabilities, however the update cycle will result in delay of assessing the impact of these data changes.

- Plants not involved in risk-informed activities may not be assessing data and incorporating into updates; however, plants pursuing extended uprates will likely be active in risk-informed activities.

Preliminary Conclusions

- Current scope of risk evaluations for extended power uprate provides adequate insights to support extended power uprate evaluations.
- Absent operational data showing reasonably significant degradations on SSC availabilities and reliability and initiating event frequencies, it may not be productive to attempt to perform risk assessments reflecting full "possibility" of synergistic effects.
- We are considering whether operational data should be assessed specifically for plants with extended power uprates.

EXTENDED POWER UPRATE REVIEWS AND STAFF ANALYSIS

by

Ralph Caruso
Reactor Systems Branch
Division of Systems Safety & Analysis
Office of Nuclear Reactor Regulation

EXTENDED POWER UPRATE ANALYSES AND STAFF REVIEWS

- The staff is mindful of the potential for reduction in plant margin and increase in risk associated with proposed extended power uprates.
- The staff has a process in place to review BWR extended power uprates
  - NEDC-32424P "Generic Guidelines for GE BWR Extended Power Uprate" (ELTR1)
  - NEDC-32523P "Generic Evaluations of GE BWR Extended Power Uprate" (ELTR2)
  - SRP
  - SERs for previously approved power uprates (up to 5%)
- Monticello and Hatch extended power uprates were reviewed and approved in accordance with this guidance.
- Extended power uprate applications are required to meet all deterministic requirements.
- The staff works closely with licensees to identify potentially significant issues/phenomena relevant to power uprates.

AUDIT CALCULATIONS

- Methodologies are approved by the staff on a generic basis.
- The staff has the capability to perform independent audit calculations for each application when deemed appropriate.
- The staff verifies that methodologies are being used in accordance with conditions of approval.
- There is nothing about power uprate that causes the staff to believe that the currently approved methodologies are not valid for this application.
- The staff plans to audit selected power uprate calculations.

BEST-ESTIMATE METHODS

- There are currently no approved "best-estimate" evaluation methods for BWRs.
- The use of "best-estimate" analyses may be necessary for future power uprates.
- The staff supports the use of "best-estimate" analyses for power uprates with prior staff review and approval of the methodology.

HIGH-BURNUP FUEL

- Staff has briefed ACRS several times on high-burnup fuel issues:
  - April 23, 1998
  - June 4, 1998
  - March 10, 1999
- No significant developments since last briefing, no change in policy.
- Agency program plan for high-burnup fuel is in place.
- Staff continues to meet with industry.
- Approval has not been granted to exceed rod average burnup limit of 62 GWd/t.
- Future approval for extensions in burnup above the present limit will require data to demonstrate that the fuel design acceptance criteria will be met.
- No new phenomena associated with power uprate, if new phenomena are identified, industry will gather data and include in analyses.

POTENTIAL EFFECTS OF EROSION/CORROSION ASSOCIATED WITH EXTENDED POWER UPRATE APPLICATIONS

by

C. E. Carpenter
Materials & Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

POTENTIAL EFFECTS OF EROSION/CORROSION ASSOCIATED WITH EXTENDED POWER UPRATE APPLICATIONS

- Staff Has Concluded That BWR Power Uprate Will Not Cause an Adverse Increase in Flow-Induced Erosion/Corrosion Damage to Reactor Coolant System Piping.
- Licensees Are Required to Reexamined Their Inspection Programs in Light of Plant-Specific Uprate Concerns, and to Evaluate the Effect on Postulated Existing Flaws.
- Licensees Verify That BWR Power Uprate Would Have No Significant Effects on the Potential for Flow-induced Erosion/Corrosion in Those Systems Which Might Be Susceptible to the Phenomenon.

CONCLUSION

There is a need for alternative repair criteria

CONCLUSION

plants will be operated with flaws in the steam generator tubes and this need not be risk significant

CONCLUSION

The general features of the procedures that the staff has established to limit the number and size of flaws left in operating steam generator tubes are adequate.

RECOMMENDATION

Risk analyses that the staff considers need to account for progression of damage to steam generator tubes in a more rigorous way.

- dynamic phenomena associated with depressurization

- plate movement

CONCLUSION

Analyses of steam generator tube performance under severe accident conditions are not adequate.

CONCLUSION

Analyses of human performance errors during design basis events appear consistent with the current state of the art

RECOMMENDATION

Data bases for 7/8" tubes need to be greatly improved to be useful

RECOMMENDATION

Staff should establish a program to monitor the predictions of flaw growth for systematic deviations from expectations

RECOMMENDATION

Staff should develop a more technically defensible position on the treatment of radionuclide release to be used in safety analyses of design basis events

(3a)

12/6/00

W Shack — reclused himself on DPO COI

1st Handout for DPO (Staff)

CREDIT FOR TUBE SUPPORT PLATES IN GL 95-05 METHODOLOGY
STEAM GENERATOR DIFFERING PROFESSIONAL OPINION ISSUES
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

December 6, 2000

Kenneth J. Karwoski, Division of Engineering, NRR (301) 415-2752

3b

GL 95-05: TECHNICAL OVERVIEW

GL 95-05 provides a methodology for an alternate tube repair criteria for predominantly axially oriented outside diameter stress corrosion cracking (ODSCC) at tube support plate elevations

Degradation occurs in crevice between tube and tube support plate - tube support plate is 0.75-inch thick

Crevice can be packed with corrosion products (e.g., magnetite)

2 fundamental goals of repair criteria: Ensure adequate structural and leakage integrity

Correlations were developed relating an inspection parameter (i.e., voltage) to the burst pressure and leakage from this type of degradation (i.e., ODSCC)

STRUCTURAL AND LEAKAGE INTEGRITY CORRELATIONS

Correlations include data from 2 sources: Tubes removed from operating steam generators and specimens produced in model boiler facilities (i.e., laboratory facilities)

Destructive examinations typically performed:

- Leak Testing
  - Used in leakage correlations
- Burst Testing
  - Used in burst pressure correlation
- Metallurgical examination
  - Ensure degradation mode is consistent with those from other sources/plants

PERFORMANCE OF BURST AND LEAK TESTING FOR GL 95-05

Burst and leak testing of model boiler and pulled tube specimens used in the burst pressure and conditional leak rate correlations was performed without the presence of the tube support plates (i.e., they were performed assuming the tube support plate moved infinitely away from the degradation)

All degradation at the support plate was exposed

If the tube support plates move during a depressurization event, the GL 95-05 methodology will provide an appropriate determination of the conditional probability of burst and the postulated accident conditions

Supporting data did not take into effect the presence of the tube support plates

If the tube support plates do not move (or do not move enough to expose all of the degradation) during a depressurization event, the GL 95-05 methodology will provide a conservative estimate of the conditional probability of burst and the postulated accident conditions

Presence of the tube support plate will limit leakage and will prevent tube from bursting

INDUSTRY PROPOSAL ON DISPLACEMENT OF TUBE SUPPORT PLATES

A utility submitted a report assessing the potential for tube support plate displacement during a postulated steam line break

Their conclusion was that the plates are essentially locked in place due to corrosion product buildup in the tube-to-tube support plate crevice

NRC had a number of issues with the report as documented in a letter to the utility

- Large forces are required to move a tube past a packed and dented tube support plate intersection

(4)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
478th	4.0	4-1

Title

SUBCOMMITTEE REPORT - NOVEMBER 13-14, 2000 T/H PHENOMENA SUBCOMMITTEE MEETING

Authors: V. SCHROCK/G. WALLIS/N. ZUBER

List of Documents Attached

1. Report of ACRS Consultant V. Schrock, dated November 28, 2000
2. Report of ACRS Consultant N. Zuber, dated November 25, 2000 (Proprietary Material Attached)
3. Comments by G. Wallis: "TRACG Model Description", undated (ACRS Internal Use Only)

4

From Staff Person

P. BOEHNERT

PROPRIETARY MATERIAL ATTACHED
"ACRS INTERNAL USE ONLY" MATERIAL ATTACHED

REPORT TO THE ACRS FROM THE PLANT SYSTEMS SUBCOMMITTEE CHAIRMAN ON THE TOPICAL REPORTS FOR ABB/CE AND SIEMANS DIGITAL APPLICATIONS

December 6, 2000
Briefing by
Dr. Robert E. Uhrig

G:\SINGH\I&C.wpd

5

Instrumentation and Control Systems Failures in Nuclear Power Plants

Paper Presented by

Robert W. Brill

At

International Symposium on

Software Reliability Engineering

San Jose, California

October 9, 2000

Review of Current Status

By

Steven A. Arndt

Office of Nuclear Regulatory Research

December 6, 2000

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- Examination of the Licensee Event Report database provides a snapshot of instrumentation and control (I&C) impact on plant safety.

- The LER database consists of all reportable events that could effect the safety of nuclear power plants.

- There were 6681 LERs Between 1994-1998, with 385 of those LERs involving digital anomalies.

- The analysis also showed that approximately 8% of all LERs, from 1994-1999, contained digital failure, and 9% of reactor trips for those years were attributed to digital I&C failures.

- The data base study shows that I&C systems including digital I&C systems, have a noticeable impact on nuclear power plants

INSTRUMENTATION AND CONTROL SYSTEM FAILURES IN NUCLEAR POWER PLANTS[1]

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KEYWORDS: Instrumentation, Control, Digital, License Event Report

ABSTRACT

Examination of the Licensee Event Report (LER) database, by the Office of Nuclear Regulatory Research, provides a snapshot of instrumentation and control (I&C) impact on plant safety. The LER database consists of all reportable events that could affect the safety of Nuclear Power Plants. The LER database study uncovered digital I&C vulnerabilities in nuclear power plants from operational experience. This study considered digital-related LERs for a five-year period, starting in 1994. The LER study places LERs in three categories: hardware, software, and human/system interface (HSI). Analysis showed an nearly equal distribution of events in each of the three categories. The analysis also showed that approximately 8% of all LERs, from 1994 to 1999, contain digital I&C failures, and 9% of reactor trips for those years are attributed to digital I&C failures. Detailed examination of the digital I&C failures emphasizes that a significant percentage of the failures occurs as a result of failures in the requirements and Verification and Validation life-cycle stages. This database study shows I&C systems, including digital I&C systems, have a noticeable impact on nuclear power plant safety.

INTRODUCTION

Instrumentation and control (I&C) systems are vital to nuclear power plant operation and safety. I&C systems provide operators with important plant information, and they send commands to plant systems. With the introduction of digital technology, I&C systems are now embedded in plant components such as transformers, valves, motor control centers, and circuit breakers. As the U.S. Nuclear Regulatory Commission moves toward a risk-informed, performance-based regulatory environment, a major question arises:

- What is the impact of digital technology on nuclear power plant safety?

The review of the Licensee Event Report (LER) database was instituted to find answers to this question. This study, provides some insight into the vulnerability of digital I&C systems and results of the LER study will help guide future research and regulatory developments regarding digital I&C systems.

NOMENCLATURE

I&C	Instrumentation and Control
LER	Licensee Event Report
V&V	Verification & Validation

LER DATABASE STUDY

The LER database consists of reports from the licensees for types of reactor events and problems that are believed to be significant and useful to the NRC in its effort to identify and resolve threats to public safety. It is designed to provide the information necessary for engineering studies of operational anomalies and trends and patterns analysis of operational occurrences. This database is stored in the Sequence Coding and Search System web site[1].

A study of these LERs was undertaken to determine whether there was sufficient operational experience that could be used to uncover digital I&C system vulnerabilities in nuclear power plants. This examination covered all LERs during the years 1994-1998 and included both digital failures and external events causing digital I&C systems to malfunction. An example of an external event affecting a digital I&C system is a case in which the control room operators received annunciations indicating that nonessential loads from the 600-volt bus had been de-energized. This event was caused by an arcing ground on a freight elevator brake solenoid, which resulted in a trip of the nonessential load lockout logic on the 600-volt bus. The ground also caused a trip of the RPS motor-generator feeder breaker. (The affected breaker

[1] The views expressed in this paper are those of the authors and should not be construed to reflect the U. S. Nuclear Regulatory Commission position.

is equipped with a microprocessor-based trip unit.) The ground affected the trip unit such that its microprocessor actuated the breaker, which in turn tripped the reactor.

The initial analysis placed the selected LERs in three categories: hardware, software, and human/system interface (HSI). (A significant number of the LERs include human errors that did not result in inappropriate operator actions. These are included in the category HSI.) In a number of LERs, the reported problem fell into multiple categories. For example, in one LER, a sudden trip of the main turbine generator resulted in a reactor trip. The reason for the turbine trip included:

- a hardware failure in a digital feedwater control card,
- a software error in the main turbine trip logic allowing a single failure to trip the turbine, and
- an HSI error in which the redundant turbine trip relays were connected in parallel rather than in series.

1.1 LER Analysis Summary

There were 6681 LERS between 1994 -1998, with 385 of those LERs involving digital anomalies. Figure 1 shows the percentage of LERs involving digital anomalies on a per year basis. With the exception of 1994, the number of digital LERs is relatively constant. A possible explanation of the high number of digital-related LERs in 1994 is that it was a year in which utilities performed a number of digital upgrades and startup and learning problems occurred.

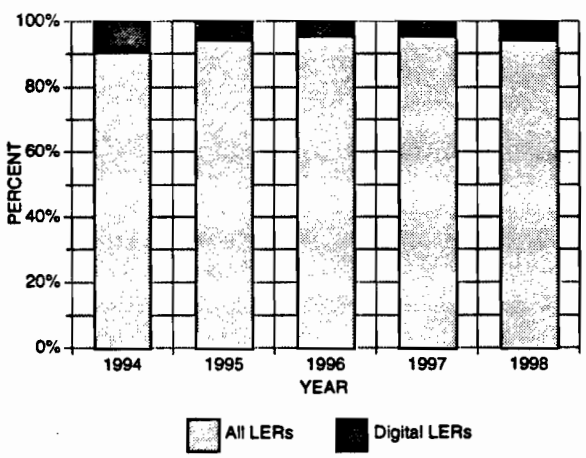

Figure 1. LER Percentages.

There were 484 reactor trips from 1994 - 1998, with digital anomalies contributing to 60 of these. As shown in Figure 2, the percentage of all trips caused by digital anomalies is relatively constant over the time period. Approximately 13% of all digital-related LERs involved a reactor trip.

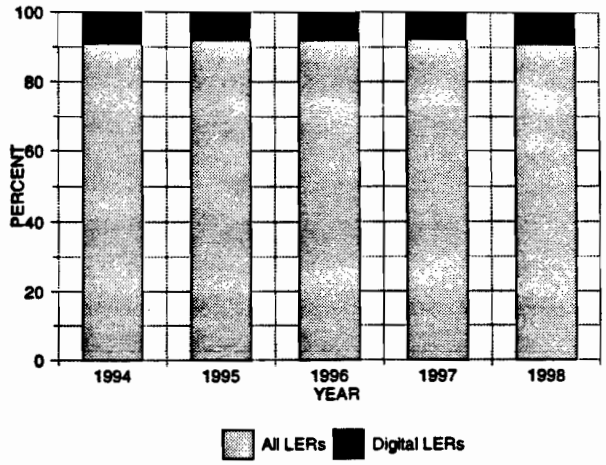

Figure 2. Trip percentages.

The number of digital LERs per category (hardware, software, and HSI) is almost evenly distributed, as shown in Figure 3. A number of the 385 digital events fit into more than one category. For example, there may have been both an HSI failure and a software failure reported in a single LER. Thus the allocation of failures to more than one category accounts for the sum of each type of failure to be greater than the total number of digital failures found in the LERs.

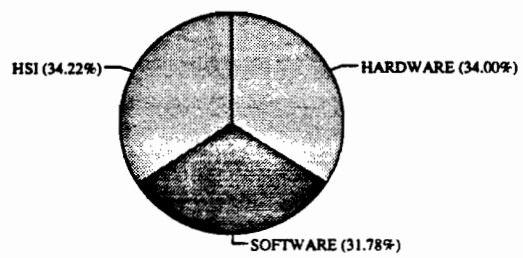

Figure 3. Digital Anomaly Categories

Figure 4 presents the analysis by system type. Digital anomalies in three safety/risk-significant systems (reactor protection, feedwater, and reactor coolant system) contributed to nearly 29% of the LERs. The largest single contributor to the LERs was the plant computer at 28%.

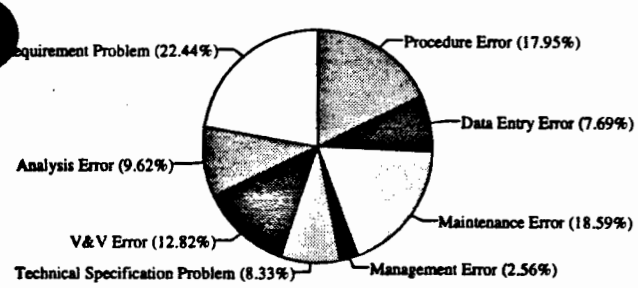

Figure 7. Digital HSI Errors

Examination of the data to determine where problems with surveillances occurred provides the results shown in Figure 8. According to the data, of the 93 surveillance errors 14 occurred in digital hardware, 21 in the software, and 58 in the HSI.

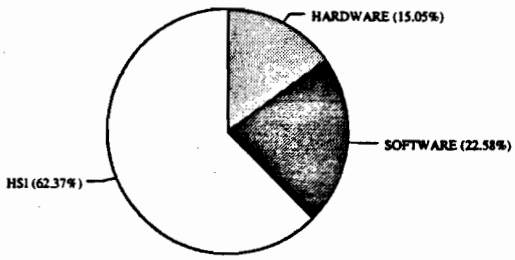

Figure 8. Digital Surveilance Problems

The types of surveillance problems that occurred were broken down into five different categories: 1) test error; 2) design error; 3)Function or box partially not tested 4) Occurred during

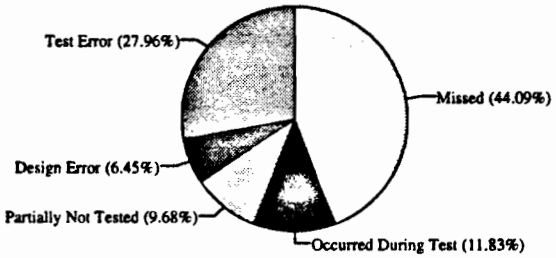

Figure 9. Surveilance Problem Breakdown

test, and 5) surveillance missed. The results are shown in Figure 9.

CONCLUSIONS

This section presents observations made from this database study.

The first observation indicates that some I&C components in non-safety systems have risk-significance. The second observation suggests a closer look at I&C components embedded in safety systems. Often, the scope of safety I&C components is limited to reactor protection systems and engineered safety features actuation systems. However, many safety systems and components, such as pumps, valves, and diesel generators, depend upon I&C components to function correctly. The third observation points to the possible risk-significance of embedded I&C components in breakers, inverters, and other required power supply components for both safety and non-safety systems. The final observation shows design and maintenance errors having as much impact on I&C reliability as component failure.

Based upon the analysis of the LER database, the failure of digital systems affects plant performance and safety. The analysis has shown that digital systems are involved with approximately 9% of the events reported in LERs and contribute approximately 13% of the trips. Analysis of the LER database reveals the types of problems occurring with digital I&C system installation and usage. However, the data is of insufficient depth to perform a definitive risk analysis. Perhaps one of the most significant observations is that with the use of computers, the problems encountered are caused by poor design, incomplete implementation of the system requirements and the human tendency to believe what the computer shows them. The other major failure category is in the V&V, both in incompleteness at the requirements level and during the V&V process. A significant number (93) of the LERs were attributed to missed surveillances. In many of these LERs, the surveillance scheduling computer programs were written in such a way that they did not alert the user to the critical dates.

REFERENCES

1. USNRC, 1999, *Sequence Coding and Search System Database*, http://scss.ornl.gov/scss/default.htm This is arstricted site, and permission to access the site must be obtained from Contact Dale Yeilding (301-415-6355)at the NRC with questions or comments concerning the SCSS Web site.

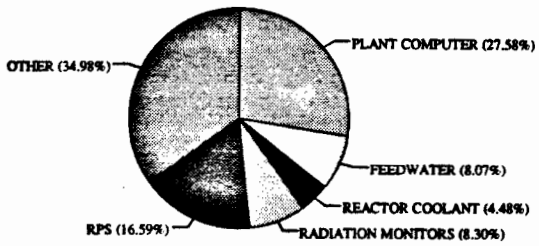

Figure 4. Digital LERS by System

Examination of the hardware LERs shows the distribution of 153 digital hardware failures. This distribution resembles that which would normally occur in an analog system. Figure 5 shows this distribution.

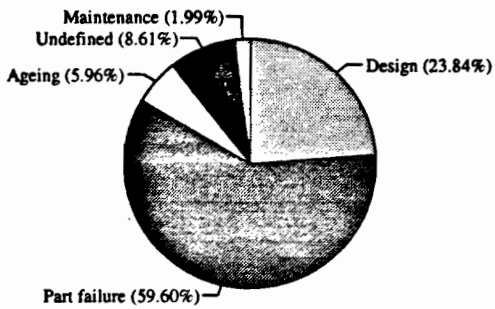

Figure 5. Digital Hardware Failures

Figure 6 presents the analysis of the 143 software events which were found in the LER's. For the analysis, the following definitions were used:

- Requirements error is an inherent error in the procedures, technical specifications, etc. that is replicated in the software.
- Software incomplete is an error that, if software had been designed correctly, with appropriate diagnostics, would not have occurred, this is an error in the requirements.
- V&V error is an error that would have been detected if the requirements, procedures, and software program had been checked properly.
- Software development is an error that occurred because software was written incorrectly by the programmer.
- Logic error is the case when the logic written into code was incorrectly.
- Undefined means there was insufficient information to categorize the source of the problem, and Miscellaneous includes data input errors where personnel input incorrect data. Software could have been written to detect problem but didn't, this is a improper analysis of the requirements.

As can be seen in the figure, the largest category is requirements errors, followed by software being incomplete, which is a form of requirements error. Together the two categories contribute over 53% of the software errors found in LERs.

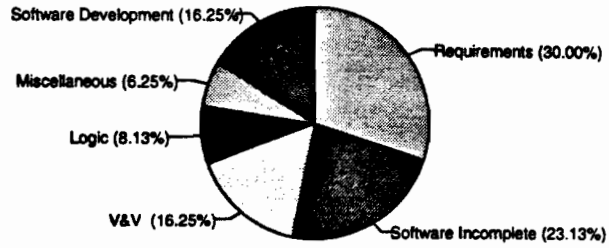

Figure 6. Software Errors

Figure 7 is an analysis of the digital LERs pertaining to human system interface. As can be seen in the figure, for the 154 incidents, approximately 58% of the total consists of problems can be attributed to problems in the requirements. Problems that in the requirements category include: analysis errors, technical specification problems, and procedure errors. The root cause of these is generally caused by inconsistent, ambiguous, and incomplete requirements. The second largest category are maintenance (maintenance and data entry errors) problems. This category contributes 26% of the total. The following definitions were used for this analysis:

- Procedure Errors are procedures for performing the required function that are incomplete, inaccurate, or incorrect.
- Data Entry Errors entail data that was incorrectly input.
- Maintenance Errors entail procedures that were not followed.
- Management Errors are cases in which management made an improper decision.
- Technical Specification Problems entail technical specifications that were incomplete, confusing, or conflict with each other.
- V&V Errors entail procedures that were not thoroughly reviewed against requirements and technical specifications or verified.
- Analysis Errors are failures in analyzing the requirements, resulting in incomplete or incorrect designs and procedures.
- Requirement Problems are problems with requirements such as inconsistentency or incompleteness.

by licensees, submittals that are now substantially out of date. But ACRS added that Lochbaum could not use any updated PRAs because those "are not publicly available."

At a commission briefing Oct. 6 by the ACRS, Commissioner Edward McGaffigan again raised the issue of publicizing the risk analyses being used by the NRC staff to judge various licensee submittals.

McGaffigan asked how could NRC get the public "to buy into the notion that a PRA is good enough?" And he raised the possibility that information from industry PRA peer reviews may somehow have to be included in the public record so that "somebody who is an interested member of the public can understand the staff's thought processes and why they thought that a PRA in this instance was good enough for the application." ACRS member George Apostolakis agreed that "this is definitely a crucial issue."

McGaffigan also alluded to a commission briefing on Sept. 29 in which he asked Nuclear Energy Institute Senior Vice President Ralph Beedle whether UCS' Lochbaum would be welcome on one of the industry PRA peer review visits. Beedle responded by saying that "I think we could probably talk somebody into letting Mr. Lochbaum in the front gate."

Lochbaum followed up with an Oct. 10 letter to Beedle: "I would welcome the opportunity to participate as an observer in the peer review process for at least one nuclear plant site."

An NEI staffer said that there are a "bunch" of PRA peer reviews coming up in the near future and that Lochbaum will likely be given a choice of which review he wants to observe.

The staffer said that NEI believes the issue of public access to the utility information used to support risk-informed NRC decisions is a "key issue." He said the industry has come up with a number of ideas on how best to do that. He said those ideas will be discussed by an NEI working group in November.

ACRS Comments On UCS Report

As for the UCS report "Nuclear Plant Risk Studies: Failing the Grade," ACRS offered a number of observations including the following:

— PRA has "reached a level of maturity that allows it to be used to identify unnecessary regulatory burden, as well as additional safety improvements. It is unfortunate that the two uses of PRA (to impose burden when necessary, and to remove it when unnecessary) have been separated by time, because this may create the false impression that burden reduction is the primary use of risk information."

— The UCS report is misleading when it just lists the number of regulatory violations and design problems. ACRS said that a recent NRC study of design basis violations showed that in 1990 about 8% had some safety significance (i.e. could potentially result in a change in core damage frequency on the order of several events per million reactor years). In 1998, NRC found that only about 1% of design basis violations had safety significance.

— "We agree that standards establishing minimum requirements for PRA quality are necessary to reduce [the NRC] staff effort required to assess the quality of PRAs used for risk-informed decision-making." ACRS added that while the development and expansion of PRA technology is needed, current PRAs can certainly be used for certain applications. "Most practitioners know that one does not always need a 'perfect PRA' to gain important insights regarding plant safety," ACRS said. "There are definite benefits to society from the use of risk information in the regulation of nuclear reactors, and it would be a disservice to the nation if the agency ignored the valuable insights that this technology provides," the ACRS said.

—Michael Knapik, Washington (mknap@mh.com)

NRC STAFFER RAISES ISSUES ABOUT DIGITAL I&C SYSTEMS

In a study of the impact of digital instrumentation and control (I&C) systems on nuclear plant safety, Robert Brill, an NRC staffer in the Office of Nuclear Regulatory Research (RES), found that some I&C components in nonsafety systems have risk-significance and that there is possible risk-significance of embedded I&C components in breakers, inverters, and other required power supply components for both safety and nonsafety systems. In a study of licensee event reports (LERs) between 1994 and 1998, Brill said he found design and maintenance errors have as much impact on I&C reliability as component failure. Brill cautioned, however, that the data were of insufficient depth to perform a definitive risk analysis.

Brill presented his findings Oct. 9 to the International Symposium on Software Reliability Engineering in San Jose, Calif. The paper is available on NRC's Adams electronic recordkeeping system (ML003757315).

Digital I&C issues were among the research issues that NRC Chairman Richard Meserve noted in a meeting with staffers in RES Oct. 19. Meserve said later that those issues could become even more important as the agency grants more plant license extensions, given that utilities are having to replace worn-out analog systems with digital systems. And a source indicated that the agency is likely to allocate more research money in this area in its fiscal 2002 budget.

Brill looked at 6,681 LERs between 1994 and 1998 and found that 385 involved digital anomalies. Brill also

found that of the 484 reactor trips from 1994-1998, digital anomalies contributed to 60 of them. The number of digital LERs by category—hardware, software, and human/system interface (HSI)—is almost evenly distributed, he found, with a number of the 385 digital events fitting into more than one category.

According to Brill, digital anomalies in three safety/risk-significant systems—reactor protection, feedwater, and reactor coolant—contributed to nearly 29% of the LERs. The largest single contributor to the LERs was the plant computer at 28%.

An examination of the 153 digital hardware-related failures, Brill said, indicated that about 60% were due to a part failure and about 24% to some design problem. Brill said this distribution of hardware failures is similar to what would occur in analog systems.

An examination of the 143 software events found that about 30% were due to a requirements error, that is, an error that is inherent in the procedures, technical specifications, etc. that is replicated in the software. About 39% of the events were due to either incomplete or incorrect software.

Of the 154 digital events where HSI was a cause, 58% of the total can be linked with requirements problems, inducing analysis errors, technical specification problems, and procedure errors. About 26% of the events were related to problems with maintenance and data entry.

Brill also said that 93 of the LERs could be attributed to missed surveillances. He said that with many of these LERs, "the surveillance scheduling computer programs were written in such a way that they did not alert the user to the critical dates."

Brill concluded with this observation: "With the use of computers, the problems encountered are caused by poor design, incomplete implementation of the system requirements, and the human tendency to believe what the computer shows them."

There will be a presentation on digital I&C issues Tuesday, Oct. 24, at NRC's 28th annual Water Reactor Safety Information Meeting at the Bethesda, Md. Marriott Hotel.

—Michael Knapik, Washington (mknap@mh.com)

INTERNATIONAL REGULATION . . .

SOUTH KOREAN REGULATORS SUCCESSFULLY FIGHTING FOR FUNDS, INFLUENCE

Senior safety officials from the Republic of Korea (ROK) explained to their NRC counterparts this month that they are winning their battle to obtain more influence over the South Korean nuclear energy program as well as government funding. Both the Directorate-General for Nuclear Safety at the Ministry of Science & Technology (MOST), the ROK regulatory agency, and its affiliated technical consultant body, the Korea Institute for Nuclear Safety (KINS), have increased nuclear safety funding since the Korean 1997 economic crisis.

Since 1997, nuclear safety experts at NRC, at the IAEA, and in Asia have warned that lack of funds, market deregulation, and over-emphasis on economics would lead to an erosion in nuclear safety in ROK and elsewhere in the region. Sources said then that the relative position of MOST vis-a-vis the ROK Ministry of Commerce, Industry & Energy (Mocie) and Korea Electric Power Corp. (Kepco), the national utility, was weak, and would be further challenged by Korea's economic crisis.

That is no longer the case, Korean officials told NRC this month. While the ROK government since 1997 has continued to cut its budget in absolute terms, funding for both the MOST safety directorate and KINS has been increasing.

Korean officials reported this month that safety problems which have beset Japan, most recently last year's criticality accident at the JCO Ltd. uranium conversion plant, also served as a wake-up call to Korean leaders, who are now supporting MOST and KINS in "making sure that something like that never happens" in Korea, one official said. The safety directorate at MOST is now "very firmly addressing safety issues" in Korea's nuclear program, one U.S. official said this month.—*Mark Hibbs, Bonn (mhibb@mh.com)*

FRENCH APPLYING 'SAFETY DOMAIN' FOR RIA, PENDING NEW CRITERION

Electricite de France (EDF) has proposed a new "safety domain" it says could be used as a regulatory stopgap pending international agreement on a new criterion for reactivity-insertion accidents (RIA).

The utility's idea is to define boundary conditions that guarantee fuel will not rupture and disperse after a sudden, sharp deposition of energy such as could conceivably occur if a control rod or rods were ejected. EDF recognizes that its technical dossier isn't solid enough to propose a single new criterion to replace the existing clad rupture criterion that assumes fuel would fail at an energy deposition of 200 calories per gram uranium. But it has asked French regulators to agree to the more flexible "safety domain" approach on a case-by-case basis until a new criterion, or more probably a new set of criteria, is set in regulations.

The safety domain is a set of four criteria, based on clad corrosion, width of power pulse, enthalphy value,

These are questions that I provided to Commissioner Diaz

(1)

Some Questions on Research

- To design a useful research program, it is necessary to have some view of when it is that the NRC must conduct an independent assessment rather than just reviewing licensee submittals and proposals. Could the commissioner provide his views on when the NRC staff should do independent assessments?

- RES justifies some research programs, at least in part, as a mechanism for preserving 'core competencies' in particular fields. Could the Commissioner provide his views on the core Competencies RES should preserve?

- RES has opportunities to join in international cooperative research programs. Often these programs are not perfectly aligned with agency needs. Could the Commissioner provide his views on far afield RES should go to preserve its participation in international cooperative research activities?

- The Commission is moving toward a more risk-informed regulatory process. Yet the agency has rather little risk information. Most of the risk information is just about plants during normal operations. Even this little information comes from studies of representative plants that are now almost 11 years old or from the IPE submittals that are widely regarded as crude or imperfect sources of risk information about specific plants. Could the Commissioner provide his views on the breadth and depth of risk information that the staff should have available for specific plants to facilitate the regulatory process?

- Should the NRC subject its risk assessment codes to a public peer review?

- NRC is allowing 'best estimate' analyses for thermal hydraulic assessments involved in regulatory analyses of plant safety. If the codes work in the sense that they match limited sets of applicable data well, is it also important that these codes have firm technical foundations?

- Human performance is likely to remain a key aspect of plant safety and important to evaluate in the assessment of risk posed by plants. Much of human performance technical community is persuaded that the safety culture of a plant has an important effect on human performance. Should there be an effort to include measures of safety culture in probabilistic risk assessments?

South Texas Project Insights into Option 2 and the Proto-type Pilot Approach

Meeting of the ACRS

December 7, 2000

Background

- Graded Quality Assurance SER received 11/97
- GQA SER did not provide expected and required flexibility during implementation. Obstacles included:
  - ASME - Seismic
  - Class 1E - Equipment Qualification
- 'Request for Exemption to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulation' submitted 07/99
- Approach represents the prototypical Option 2 pilot per SECY 98-300 for risk-informing 10CFR Part 50

Background (cont.)

- SECY-98-0300 states:
  - 'Under [Option 2], SSCs of low safety significance (from a risk-informed assessment) would move from 'special treatment' to normal industrial (sometimes called 'commercial') treatment, but would remain in the plant and be expected to perform their design function but without additional margin, assurance, or documentation'
- RAIs received 01/18/00 and responded to on 05/22/00
- Revised Exemption Request submitted 08/31/00
- Draft Safety Evaluation Report received 11/15/00

Future target dates

- 01/15/01 - STP to respond to the Draft SER
- 04/15/01 - Final SER to be granted

SECY-98-0300 Option 2

- Per SECY 98-300, Option 2 would:
  - adjust the scope of SSCs to which special treatment requirements apply
  - adjust SSC scope without changing the regulations

  Low safety significant SSCs would:
  - remain in the plant
  - move from special treatment to normal commercial treatment
  - be expected to perform their design function but without additional margin, assurance, or documentation associated with high safety significant SSCs
- Adjustments to the regulation content (distinct from the scope) would be addressed under Option 3
- Safety-related SSCs would remain safety-related (not reclassified)

STP Foundational Principles in Piloting an Option 2 Approach

- Emphasis is placed on proper categorization of SSCs
- Current commercial practices are sufficient for LSS/NRS safety-related SSCs
- Details of commercial practices are unimportant to safety given the low safety significance of LSS/NRS SSCs
- No additional controls required over LSS/NRS commercial practices
- HSS/MSS SSCs would continue to be governed by existing regulations and non-safety related HSS/MSS SSCs would be evaluated for enhanced treatment
- Focus on procurement benefits to be gained from ASME, EQ, Seismic, and 1E exemptions

What Does the Exemption Request Ask For?

- Exemption seeks to exclude Low Safety Significant (LSS) and Non-Risk Significant (NRS) SSCs from the scope of special treatment requirements of the following regulations
  - 10CFR Part 21 (Defect notification)
  - 10CFR50.34 (Appendix B treatment)
  - 10CFR50.49 (EQ)
  - 10CFR50.54 (QA Program)
  - 10CFR50.55a (ASME)
  - 10CFR50.59 (Change Evaluation)
  - 10CFR50.65 (Maintenance Rule)
  - Appendix A; GDCs 1,2,4,18 (QA, Seismic, EQ, 1E)
  - Appendix B (QA Program)
  - Appendix J (RCB Leak Testing)
  - 10CFR Part 100 (Seismic)

Feedback From the Draft SER

- Indication where Special Treatment Requirements can be relaxed pending resolution of open items:
  - 10CFR Part 21 - no issues
  - Appendix B -some restrictions apply
  - Appendix J - no issues
  - Maintenance Rule - some restrictions apply
  - 10CFR50.59 - partially granted
  - Inservice Testing - some additional restrictions placed on testing and inspections

Feedback From the Draft SER (cont.)

- Indication where Special Treatment Requirements cannot be relaxed:
  - Inservice Inspection
  - Repair and Replacement of ASME with non-ASME
  - Changes to the QA Program
- Indication where it is indeterminate if Special Treatment Requirements can be relaxed (potential denials):
  - Equipment Qualification
  - Seismic
  - Class 1E

More Insights Into Draft SER

- Additional Treatment may be Imposed on LSS/NRS SSCs

  DSER:

  - Current STP commercial treatment needs additional requirements
  - Additional requirements to provide confidence
    - Commit to National Consensus Standards
    - Testing at design basis conditions
    - Perform pre-service inspections/pre-operational testing
    - Engineering analyses for procurement

  Possible Impact:

  - Separate commercial programs for LSS/NRS safety related SSCs
  - Impose additional burden onto existing program
  - Certain testing and inspections are not warranted or feasible
  - May be unable to use commercial grade items in LSS/NRS applications

More Insights into Draft SER (cont.)

- Categorization Process

  DSER:

  - STP categorization process does not fully address onsite and off-site dose consequences
  - Insights from containment systems that mitigate latent fatality risks should be added into the categorization process

  Possible Impact:

  - revision to the existing NRC-approved categorization process
  - potential reevaluation of SSCs

More Insights into Draft SER (cont.)

- Restrictive Commitment Change Process

  DSER:

  - extensive detail requested for inclusion into FSAR
  - stringent change control processes which mirror controls imposed on Tech Specs

  Possible Impact:

  - prior NRC approval would be required for any changes in either categorization or treatment (including beneficial changes)
  - flexibility not given to permit changes to account for feedback
  - unable to adjust treatment (MOV, AOV, snubber testing scope) without additional NRC approval

More Insights into Draft SER (cont.)

- Reliance on Deterministic Criteria

  DSER:

  – deterministic basis relied on for evaluating exemption request

  Possible Impact:

  – risk informed arguments are not given appropriate consideration (for example, ISI, Seismic):
    - LSS/NRS SSCs currently receiving special treatment only due to the deterministic definition of safety-related
    - industry data demonstrates that failure rates of safety-related and non-safety related SSCs are essentially the same
    - an assumed 10x increase in failure rates of all LSS/NRS SSCs results in only a 2% increase in CDF and LERF
    - if an LSS/NRS SSC fails, no significant impact on safety

More Insights into Draft SER (cont.)

- Some Special Treatment Requirements are viewed as Design Requirements

  DSER:

  - Fracture toughness impact tests viewed as design requirement
  - Other ASME requirements (including allowable stress limits and hydrostatic testing) may also be viewed as design requirements

  Possible Impact:

  - These special treatment requirements may not be available for inclusion into an Option 2 approach
  - Unable to replace ASME components with non-ASME
  - Resolution may be deferred until revised Code case is approved or other Option 3 method is considered

More Insights into Draft SER (cont.)

- HSS/MSS SSCs Require Additional Evaluations

  DSER:

  - HSS/MSS SSCs require a documented engineering evaluation to determine if a beyond design basis function exists and is properly treated

  Possible Impact:

  - Additional burden associated with preparing an engineering evaluation for every HSS/MSS SSC
  - SSCs will perform no unique function under beyond design basis conditions that are not already capable of being performed under design basis conditions

Concluding Remarks

- Draft SER represents a significant effort
- SECY-98-0300 Option 2 is a challenging task
- Important to focus on the adequacy of categorization such that the current commercial controls can be used
- Benefits are expected in procurement of seismic, EQ, ASME, and 1E replacement components
- NRC staff and STP are working to achieve a mutually acceptable result

United States
Nuclear Regulatory Commission

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NRC DRAFT SAFETY EVALUATION

SOUTH TEXAS PROJECT REQUESTED EXEMPTIONS FROM SPECIAL TREATMENT REQUIREMENTS

John A. Nakoski
Senior Project Manager
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

December 7, 2000

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AGENDA

1. TIMELINE
2. STAFF REVIEW PROCESS
3. CATEGORIZATION PROCESS
4. TREATMENT PROCESSES
5. CONTROLLING CHANGES
6. PRELIMINARY ASSESSMENT OF EXEMPTIONS

TIMELINE

- *7/13/99* *Exemption Request Submitted*
- *8/31-9/1/99* *Meeting on Exemption Requests*
- *10/5-6/99* *Meeting on Exemption Requests*
- *1/18/00* *Request for Additional Information Issued*
- *4/10-11/00* *Meeting on Categorization*
- *6/20-21/00* *Meeting on Treatment*
- *7/19/00* *Draft Review Guidelines Issued to STP*
- *7/24-25/00* *Meeting on Commercial Practices*
- *8/31/00* *Revised STP Exemption Request Submitted*
- *11/15/00* *Draft Safety Evaluation Issued*
- **12/7/00** **ACRS Briefing on Draft Safety Evaluation**
- **2/15/01** **Open Items from Draft SE Resolved**
- **3/8/01** **Commission Paper Due**
- **3/15/01** **Final Safety Evaluation Due**
- **3/30/01** **Commission Briefing**
- **4/15/01** **Issue Final SE and Exemptions**

STAFF REVIEW PROCESS

Risk-Informed Decisionmaking - Key Principles

1. **Meets Current Regulations** (Unless Exemption or Rule Change)
2. **Consistent with Defense-in-Depth Philosophy**
3. **Maintains Sufficient Safety Margins**
4. **Increases in CDF/Risk Are Small** (consistent with the intent of the Commission's Safety Goal Policy Statement)
5. **Impact Monitored** (Using Performance Measurement Strategies)

Review Guidelines Developed for Assessing STPNOC Exemption Requests (Provided to STPNOC on July 19, 2000)

CATEGORIZATION PROCESS

- ☞ **Generally Acceptable to Define Risk Significance for Exemptions** (subject to open item resolution)
- ☞ **Tied to Confidence of Functionality Provided by Treatment Processes**

Open Items on Categorization Process:

- ★ **Equations for PRA Importance Measures on Common Cause Failure Contribution** (Open item 3.1)
- ★ **Criteria for Use of Fussell-Vesely Importance Measure to Categorize SSCs as HSS** (Open item 3.2)

CATEGORIZATION PROCESS (con't)

Open Items on Categorization Process: (con't)

- ★ Qualification Attributes for Expert Panel and Working Group Members in FSAR (Open item 3.3)
- ★ Risk Significance of SSCs that Mitigate Consequences of Accidents by Maintaining Containment Integrity (Open item 3.4)
- ★ Support for ASME Section XI ISI Exemption (Passive Functions) (Open item 3.5)
- ★ Use of "General Notes" (Open item 3.6)

TREATMENT PROCESSES

Treatment of HSS/MSS Structures, Systems, Components (SSCs), and Functions

☞ **Provides Confidence in Functionality** (subject to open item resolution)

- ★ **HSS/MSS Remain Within Scope of Existing NRC Special Treatment Regulations**
- ★ **Open Item 4.1 - Process Attributes for Determining Treatment Applied to HSS/MSS Functions Not Currently Covered by NRC Required Programs** (Safety- and Non-safety-Related)

TREATMENT PROCESSES

Treatment of LSS/NRS SSCs and Functions

- ☞ **Generally Provides Confidence in Functionality**
  (subject to open item resolution)

LSS/NRS Open Items

- ☞ **Open Item 4.2: Process Attributes to Include in FSAR**

  ✓ **<u>Procurement</u>: (1) Item Received is Item Ordered; (2) Vendor Recommendations Considered; (3) Engineering Evaluations Provide Confidence That Replacement SSCs Meet Design-Basis Inputs; and (4) National Consensus Standards Used**

TREATMENT PROCESSES

☞ **Open Item 4.2: Process Attributes to Include in FSAR (con't)**

- ✓ **<u>Installation</u>: (1) Provide Preoperational/Preservice Testing and Evaluation; and (2) National Consensus Standards Used**
- ✓ **<u>Maintenance</u>: (1) Vendor Recommendations Considered; (2) Implements Corrective Action Process; (3) Proper Maintenance and Acceptable Operation Demonstrated; and (4) National Consensus Standards Used**

TREATMENT PROCESSES

☞ **Open Item 4.2: Process Attributes to Include in FSAR** (con't)

- ✓ **<u>Inspections, Tests, and Surveillances</u>: (1) Vendor Recommendations Considered; (2) Conducted, or Compared to Performance, at Design-Basis Conditions; and (3) National Consensus Standards Used**
- ✓ **<u>Management and Oversight</u>: (1) Training and Qualification per Vendor Recommendations and National Consensus Standards; and (2) Surveillance Equipment Controlled** (Including Attribute on Confidence of Functionality After Failure of Post-Calibration Checks)

TREATMENT PROCESSES

LSS/NRS Confirmatory Items

Confirmatory item 4.1: Licensee Will Resolve Implementation Inconsistencies with Program Description

- ✓1E Component Not Fully Qualified Isolated from 1E Circuitry Without Discussing Functional Capability
- ✓SSCs Exceeding Qualified Life Assumed Capable of Functioning and Not Replaced Unless Separate Reason
- ✓Functional Requirements Envelope the Credible Design Basis Conditions Expected
- ✓Designed to Function in the Installed Environment

TREATMENT PROCESSES

LSS/NRS Confirmatory Items (con't)

Confirmatory Item 4.2: Confirm Commitment to Follow NRC-Endorsed NEI Guidance on Commitment Management

- STPNOC Requested 10 CFR 50.59 Exemption to Extend to Other Special Treatment Provisions in FSAR
- NRC Cannot Support an Open Ended Exemption from 10 CFR 50.59
- As Committed, Follow NEI 99-04, "Guidelines for Managing NRC Commitments"

CONTROLLING CHANGES

- ✓ Exemption Requests Based on Processes
- ✓ NRC Must Have Confidence that Changes to Processes do not Invalidate Bases for Exemptions
- ✓ STPNOC Must Have Flexibility to Change Implementing Procedures as Experience Gained
- ✓ Processes to be Documented in FSAR (10 CFR 50.59 Alone is not Adequate to Control Changes)
- ✓ No Changes to FSAR Description of Processes Allowed without NRC Approval (Open item 5.1)

PRELIMINARY ASSESSMENT OF EXEMPTIONS (con't)

Deny: (Absent Additional Information From Licensee)

- ★ 10 CFR 50.34(b)(6)(ii) - App. B Information Included in FSAR
- ★ 10 CFR 50.54(a)(3) - Changes to QA Program (Open item 7.1)
- ★ 10 CFR 50.55a(g) - Section XI Repair/Replacement & Inspection (Open items 10.1 and 10.2)

More Information Needed:

- ? 10 CFR 50.34(b)(11) - Related to SSE and OBE (Part 100, App. A)
- ? 10 CFR 50.49(b) - Electrical Equipment Important to Safety (Open item 8.1)
- ? 10 CFR 50.55a(h) - IEEE 279 Section 4.4 (Open item 11.1)
- ? 10 CFR Part 100, App. A, VI, (a)(1) & (2) - SSE and OBE (Open item 18.1)

PRELIMINARY ASSESSMENT OF EXEMPTIONS

Approve: (Subject to Resolution of the Open/Confirmatory Items in the DSE)

- ✓10 CFR 21.3 - Definition of Basic Component
- ✓10 CFR 50.55a(f) - ASME Section XI Inservice Testing
- ✓10 CFR 50.59 - Changes, Tests, & Experiments - (Limited)
- ✓10 CFR 50.65(b) - Maintenance Rule Scope (Open item 13.1)
- ✓10 CFR Part 50, App. B - Quality Assurance Criteria
- ✓10 CFR Part 50, App. J - Type C Containment Leak Testing

Not Necessary: (Therefore, should deny)

- ☞ GDC 1 - Quality Standards and Records
- ☞ GDC 2 - Protection Against Natural Phenomena
- ☞ GDC 4 - Environmental and Dynamic Effects
- ☞ GDC 18 - Inspect/Test Electrical Power Systems

10

Overview of Revised NEI 99-03

NEI

1

Purpose of NEI 99-03

- Provide an approach assuring adequate protection of CR operators during abnormal events (Radiological and Toxic gas)
- Obtain generic NRC agreement with that approach

NEI

2

Key Elements

- CR in-leakage (baseline test and periodic assessment)
- Toxic gas (reassessment and periodic evaluation)
- Smoke infiltration -- qualitative assessment
- Uses existing licensing basis
- Limiting design basis accident assessment
- CR as-built configuration and operating procedures assessment
- Considers current radiological dose analysis methods (TID and AST)
- Program to maintain CRH

Achieves reasonable assurance of CR operator protection

NEI

Baseline Testing

- Baseline test to determine air in-leakage
- Baseline Test Attributes
  - Comprehensive
  - Reflects accident configuration lineup(s)
  - In accordance with recognized standards
- Acceptable baseline test methods
  - Integrated tracer gas testing
  - Component test method
  - Alternative test method(s)

NEI

Tracer Gas Method
ASTM E741

- Valid for all CR designs
- Recommended for Non-pressurized CR
- Factors affecting accuracy
  - Uniform concentration throughout CR volume
  - Determination of CR volume
  - Environmental effects (wind, temperature, pressurization flows)

NEI

Component Test Method

- Procedure to determine total in-leakage
  - Demonstrate that CR spaces are at positive pressure to all adjacent spaces
  - ID potential in-leakage sources (vulnerable)
  - Measure in-leakage of vulnerable components
- Only for pressurized CRs
  - Key -- CR positive pressure to all adjacent areas
  - Likely use -- CRs with small number of potential in-leakage vulnerabilities

Acceptable for baseline test

NEI

Reasons for Component Test

- Recommended for positive pressure CRs with few in-leakage pathways:
  - Large pressurization air flow can lead to significant uncertainties in tracer gas results
    - Neutral <10%
    - Pressurized 30% to 60%
- Benefits
  - Component test methods are routinely used by plant staff
  - Measurement uncertainty <5%

Reasonable assurance of accurate in-leakage measurement

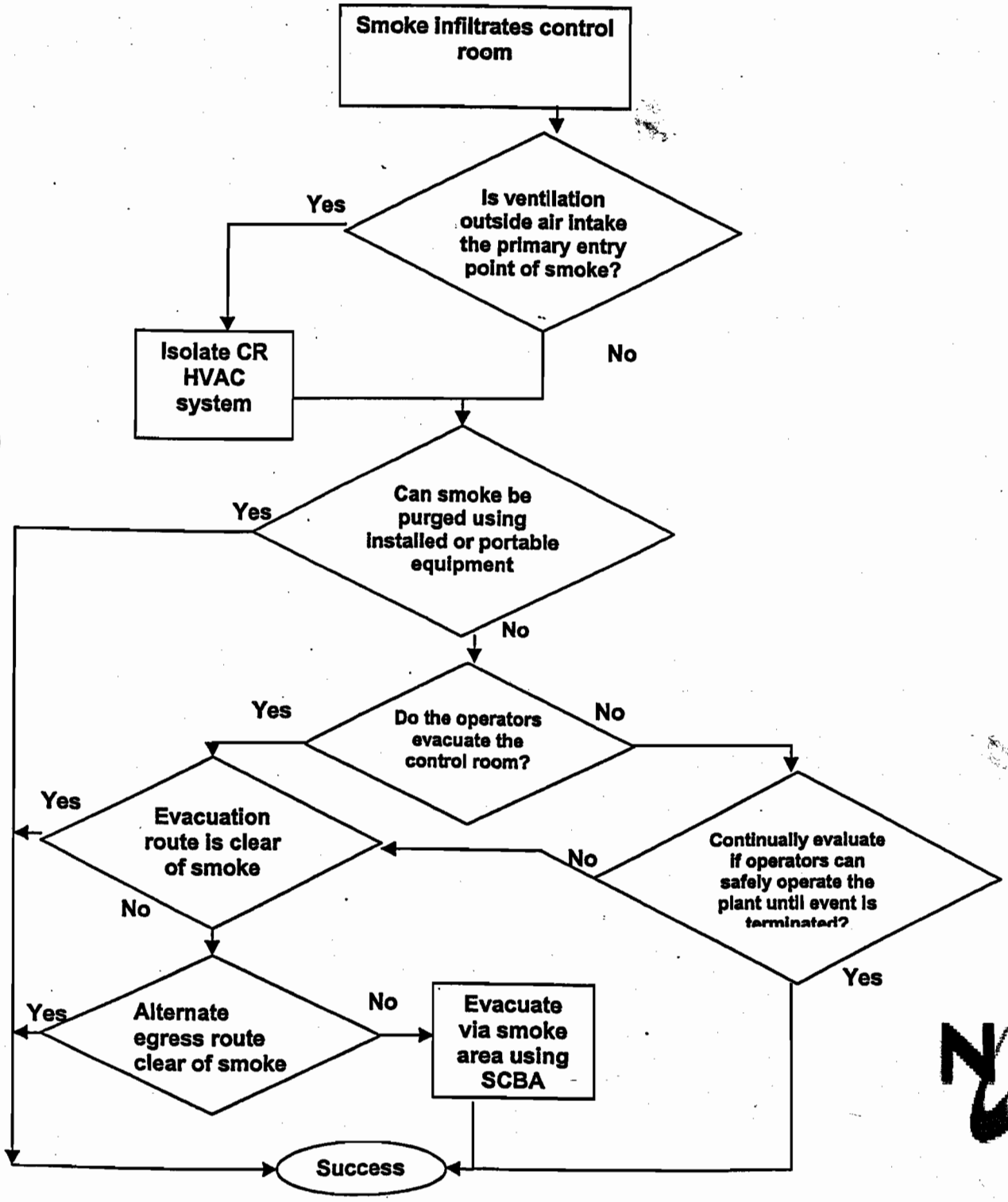
NEI

CRH Program

- Periodic assessment
  - System material condition
  - In-leakage challenges
  - Toxic gas challenges
- Configuration control
  - CRE barrier control
  - Procedure control
  - Design change control
  - Analysis change control
- Training
- Factors of assessment/test frequency
  - Number of in-leakage sources
  - Differential pressure margins
  - Margin of measured to design in-leak
- Test when appropriate

NEI

CR Smoke Infiltration



NEI

ANALYSIS IMPROVEMENT

- Alternative Source Term (AST)
  - 5 Rem TEDE
- AST insights with TID source term
  - 50 Rem Thyroid Limit
- Meteorology and dispersion modeling

NEI

AST CONSIDERATIONS VALID FOR TID

- **Fuel handling accident**
  - **Revision to Isotopic Gap Fractions**
  - **Decontamination Factor of 200 for 23 ft of water**
- **SGTR spiking factor of 335**
- **SGTR and MSLB**
  - **Spike duration less than 8 hrs**
- **Elimination of ECCS passive failure**
  - **50 gpm for 30 minutes at 24 hours after DBA LOCA**
- **Lower BWR containment leak rates at time greater than 24 hours**
- **Enhanced containment mixing rates**

NEI

METEOROLOGY AND DISPERSION MODELING

- **Use of ARCON96 acceptable for χ/Q calculations**
  - **Advanced Modeling Technique**
  - **Critical Component for Analysis**
  - **Does Not Handle All Release Scenarios**
    - **Free Standing Stack Releases**
    - **High Energy Steam Line Valve Releases**
- **MSSV and ADV χ/Q should consider elevated release**
- **ARCON96 upgrade**
  - **Joint NRC/NEI Effort**
  - **NRC Contribution**
    - **Enhancement of ARCON96**
  - **NEI Contribution**
    - **Technical Inputs**
    - **Benchmarking**

NEI

Technical Specifications

- Options considered
  - No change
  - Commitment to a CRH Program
  - Admin TS committing to a CRH program
  - Moving CR HVAC TS requirements into a TRM and adoption of periodic in-leakage assessment requirements through a CRH program
  - Adoption of an in-leakage surveillance in the current TS
- Observations
  - Commitment to reassess CRH in-leakage periodically is the right thing to do and is recommended by NEI 99-03
  - Current TS surveillances adequately address the operability of CR HVAC systems
  - Technical Specification should focus on parameters and indications observable and controllable by the operator
  - It is sufficient to maintain CRH in accordance with Appendix B criteria
  - TSTF requested to examine issue

NEI

Future NEI Actions

- Develop alternatives to NRC positions on:
  - Test options -- survey of tested CRs
  - Technical specifications (need, AOT and test frequency) -- TSTF
  - Change to common licensing basis - accidents analyzed -- Backfitting
  - Elimination of excessive conservatisms -- continued discussion with NRC technical staff
- NRC staff comments on NEI 99-03 -- December 31, 2000
- Issue NEI 99-03 to Industry for review -- Targeted for January 2001
- Develop comments on draft RG
- Issue final NEI 99-03 and conduct industry workshop -- Summer 2001?

NEI

UNCERTAINTY AND REPEATABILITY OF TRACER GAS INLEAKAGE MEASUREMENTS

Peter L. Lagus, Ph.D., CIH

December 7, 2000

Lagus Applied Technology, Inc.

San Diego, CA

TRACER GAS MEASUREMENTS OF AIR INLEAKAGE PERFORMED IN 16 NUCLEAR POWER PLANT CRE's

(Representing 22 reactors)

- In practice a single value of inleakage is obtained for each CREVS operating mode
  - For a pressurization CREVS, each Inleakage value is obtained from between 10 and 20 individual data points
  - For a recirculation CREVS, the volume-normalized Air Inleakage Rate is determined from at least one hundred data points

ANSI/ASME STANDARD PTC 19.1 "MEASUREMENT UNCERTAINTY"

- Combines both Bias or Systematic Uncertainties (Errors) of the measurement equipment with Random Uncertainties (Errors) of the actual measurement data
- Provides 95% Confidence Limits
- Substitutes a calculational format for subjective "engineering judgment" uncertainty analysis
- Used extensively, for instance, in Steam Turbine Industry

PROBLEM:

OUT OF OVER SIXTY INDIVIDUAL DATA SETS, ONLY FOUR HAVE BEEN REPEATED

HENCE, A STATISTICALLY DEFENSIBLE METHOD IS REQUIRED TO ASSESS UNCERTAINTY

ANSI/ASME ROOT SUM UNCERTAINTY, Urss IS GIVEN BY

$$U_{rss} = \pm\left[(B)^2 + (t_{95} \bullet S)^2\right]^{1/2}$$

- **B= Systematic Uncertainties in Measurement Apparatus**
- **S= Random Uncertainties in Measured data**
- **t95= Student's "t" distribution value**

SYSTEMATIC UNCERTAINTIES (BIAS)

- Injection Gas Conc (for Press CREVS)
- Injection Gas Flowmeter (for Press CREVS)
- Analyzer Calibration Gases
- Analyzer Response
- Volume of CRE (for Recirc Systems)

RANDOM UNCERTAINTIES

- Variability in the Measured Data Set
  - Plant Configuration & Operation
  - HVAC Configuration & Operation
  - External Influences
  - Gas Sampling
  - Incomplete Mixing
  - Tracer not at steady state (for Pressurized CREVS)

UNCERTAINTY

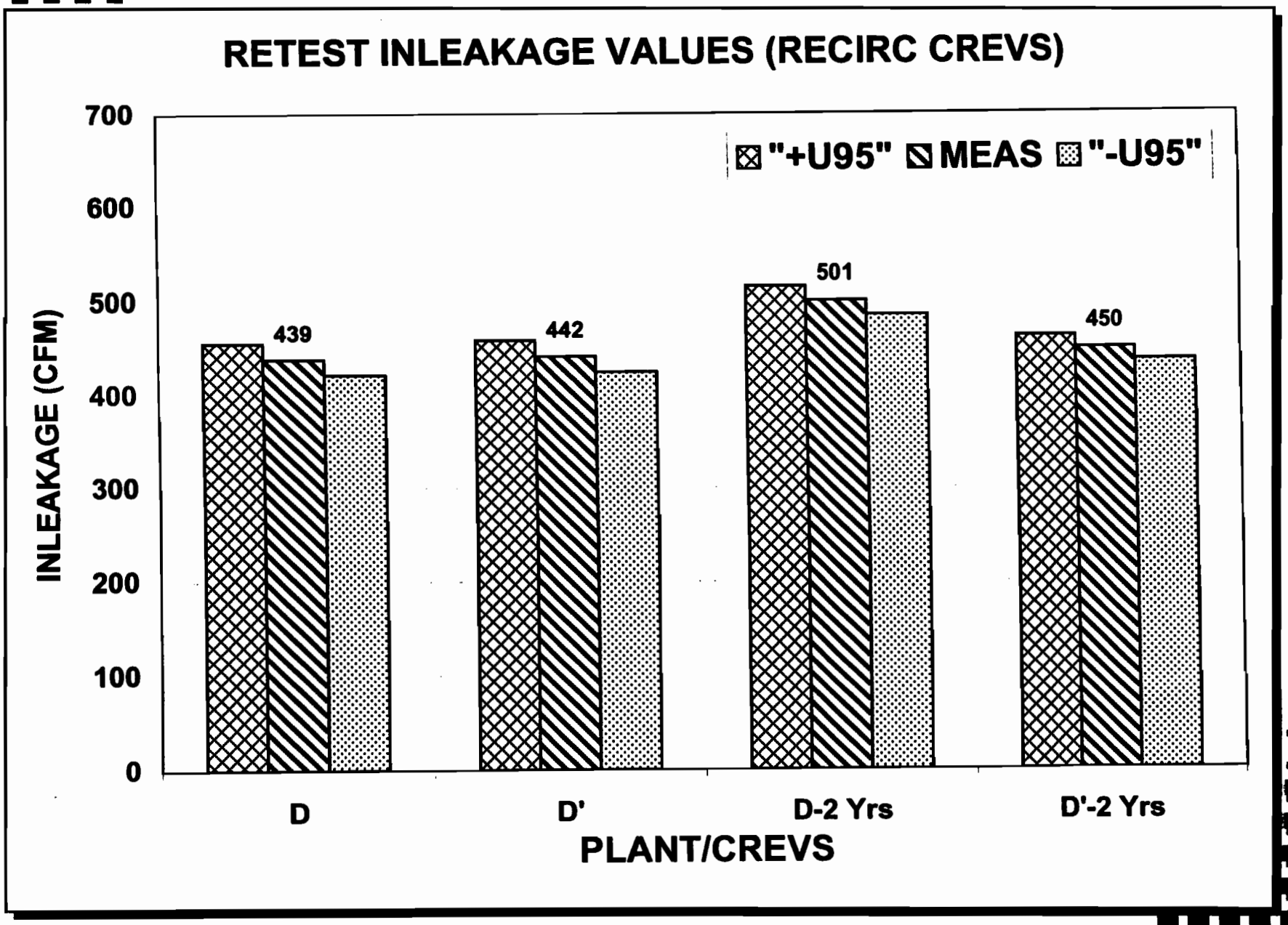
SIXTY INDIVIDUAL DATA SETS FOR PRESS & RECIRC CREVS

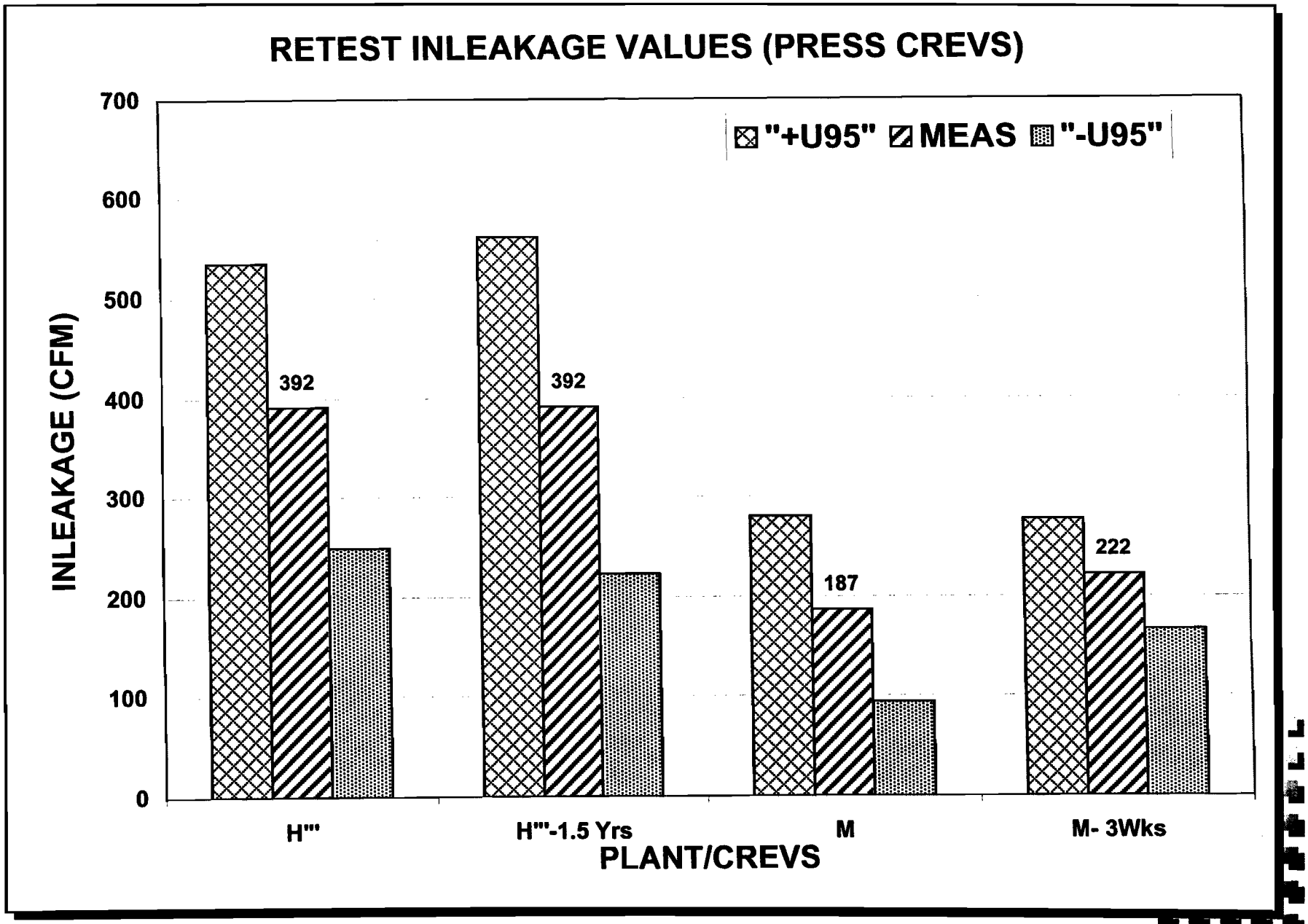
- Random Uncertainties are *larger* than the Systematic Uncertainties for Press CREVS
- Urss for Press CREVS average approx.. ± 40 %
- Random Uncertainties are *smaller* than the Systematic Uncertainties for Recirc CREVS
- Urss for Recirc CREVS average approx.. ± 5% (assumes ± 2% in CRE Volume)

REPEATABILITY

FOUR RETEST CREVS DATA SETS

- No Modifications to CRE or CREVS
- Tests with Same Ventilation & Physical Plant Lineup
- Two Pressurization CREVS in different plants
- Two different Recirculation CREVS in same CRE (High dP induced across CRE)


RETEST INLEAKAGE VALUES (RECIRC CREVS)
"+U95"
MEAS
"-U95"
INLEAKAGE (CFM)
700
600
500
400
300
200
100
0
439
442
501
450
D
D'
D-2 Yrs
D'-2 Yrs
PLANT/CREVS



LAGUS APPLIED TECHNOLOGY, INCORPORATED

REPEATABILITY

- Is better than the 95% Urss limits for Two Pressurization CREVS
- Is within the 95% Urss limits for three of four Recirculation CREVS

HOW TO DEFINE AN ACCEPTABLE INLEAKAGE VALUE?

- OSHA inspectors face same problem when sampling an atmosphere for potential violations
- Inspectors use a one-sided confidence limit test on airborne concentration values
  - Any measured value is compared to the PEL and a confidence limit based on a standard Sampling and Analysis error
- Crudely speaking, any value that lies below the Upper Confidence Limit is not in violation

ACCEPTABLE INLEAKAGE VALUE (Cont'd)

- For inleakage, NRC could propose to accept a similar Confidence Limit Value concept
  - NRC wants UCL, Industry wants LCL
- In Control Room Habitability context, need to establish "Standard Errors" for test methods
  - Need to precisely define test performance, instrumentation, calibration, sampling and analysis protocols
  - These definitions are necessary to control Confidence Limit Value Excursions

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

CALCULATION OF NEUTRON FLUENCE REGULATORY GUIDE

Dr. Nilesh Chokshi, Chief
Materials Engineering Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research

301-415-6013

U.S. NRC Office of Nuclear
Regulatory Research
Division of Engineering Technology
Materials Engineering Branch

Purpose of Presentation

- To Provide Overview of the Final Version of the Regulatory Guide
- To Discuss Disposition of Selected Public Comments
- To Seek ACRS Approval to Issue the Final Regulatory Guide

Need for Regulatory Guide

- Provide Standardized Methods and Procedures for Fluence Calculations and Use of Vessel Dosimetry
- Wide Variety of Application of Existing Methods for Determining Fast Neutron Flux
- Comply with Regulations on Reactor Pressure Vessel Integrity

Status of Regulatory Guide

- Initial Efforts Started in 93
  - Publication of 2 Drafts
  - 3 Rounds of Public Comments
  - 3 Public Meetings
  - 5/00 - Latest Round of Comments Received
- 11/00 - Appropriate Comments Incorporated
- NRR and OGC Concurrence
- ACRS Subcommittee on Materials and Metallurgy Presentation 11/16/00
- Plan to Publish Early 2001

U.S. NRC Office of Nuclear
Regulatory Research
Division of Engineering Technology
Materials Engineering Branch

Presentation Outline

▪ Overview of the Guide	William Jones, NRC
▪ Guide Technical Details	Dr. John Carew, BNL
▪ Summary and Schedule	William Jones, NRC

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

CALCULATION OF NEUTRON FLUENCE REGULATORY GUIDE

William Jones
Materials Engineering Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research

301-415-7558

U.S. NRC Office of Nuclear
Regulatory Research
Division of Engineering Technology
Materials Engineering Branch

Introduction

- Objectives of Regulatory Guide
- Contents of Regulatory Guide

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Division of Engineering Technology
Materials Engineering Branch

OBJECTIVES OF REGULATORY GUIDE

PROVIDE:

- METHODS FOR FLUENCE CALCULATION AND USE OF DOSIMETRY FOR DETERMINING RPV NEUTRON FLUENCE THAT SATISFY REGULATIONS.
- DESCRIPTION OF FLUENCE CALCULATION METHODS
- USE OF DOSIMETRY MEASUREMENTS FOR VALIDATION OF CALCULATIONS
- PROCEDURES FOR QUALIFICATION OF CALCULATIONS & DETERMINATION OF UNCERTAINTY
- MEASUREMENT PROCEDURES

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Regulatory Research
Division of Engineering Technology
Materials Engineering Branch

CONTENTS SUMMARY

- FLUENCE CALCULATION METHODS
  - Input Data
    - Materials & Geometry
    - Cross-Sections
    - Neutron Source
  - Fluence Calculation
  - Qualification & Uncertainty

- DOSIMETRY
  - Procedures
  - Dosimeters
  - Measurement Uncertainties
  - Validation
  - Fluence Estimates Based on Measured Data

CONTENTS SUMMARY

(CONTINUED)

- REPORTING
  - Methods
  - Calculation vs Measurement
- IMPLEMENTATION

Summary

- Detailed Information on Fluence Calculation and Measurement Methods
- Benchmark Calculations in NUREG/CR-6115
- Extensive Review and Substantial Public Participation
- Believe Regulatory Guide is Ready for Final Publication
- ACRS Materials and Metallurgy Subcommittee Met 11/16/00
- Publish Early 2001

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
DG-1053 CALCULATION OF NEUTRON FLUENCE

Presented by

John Carew

Energy and Nuclear Technology Division

Department of Energy Science and Technology

Brookhaven National Laboratory

December 7, 2000

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
DG-1053 CALCULATION OF NEUTRON FLUENCE

Presented by

John Carew

Energy and Nuclear Technology Division

Department of Energy Science and Technology

Brookhaven National Laboratory

December 7, 2000

Background

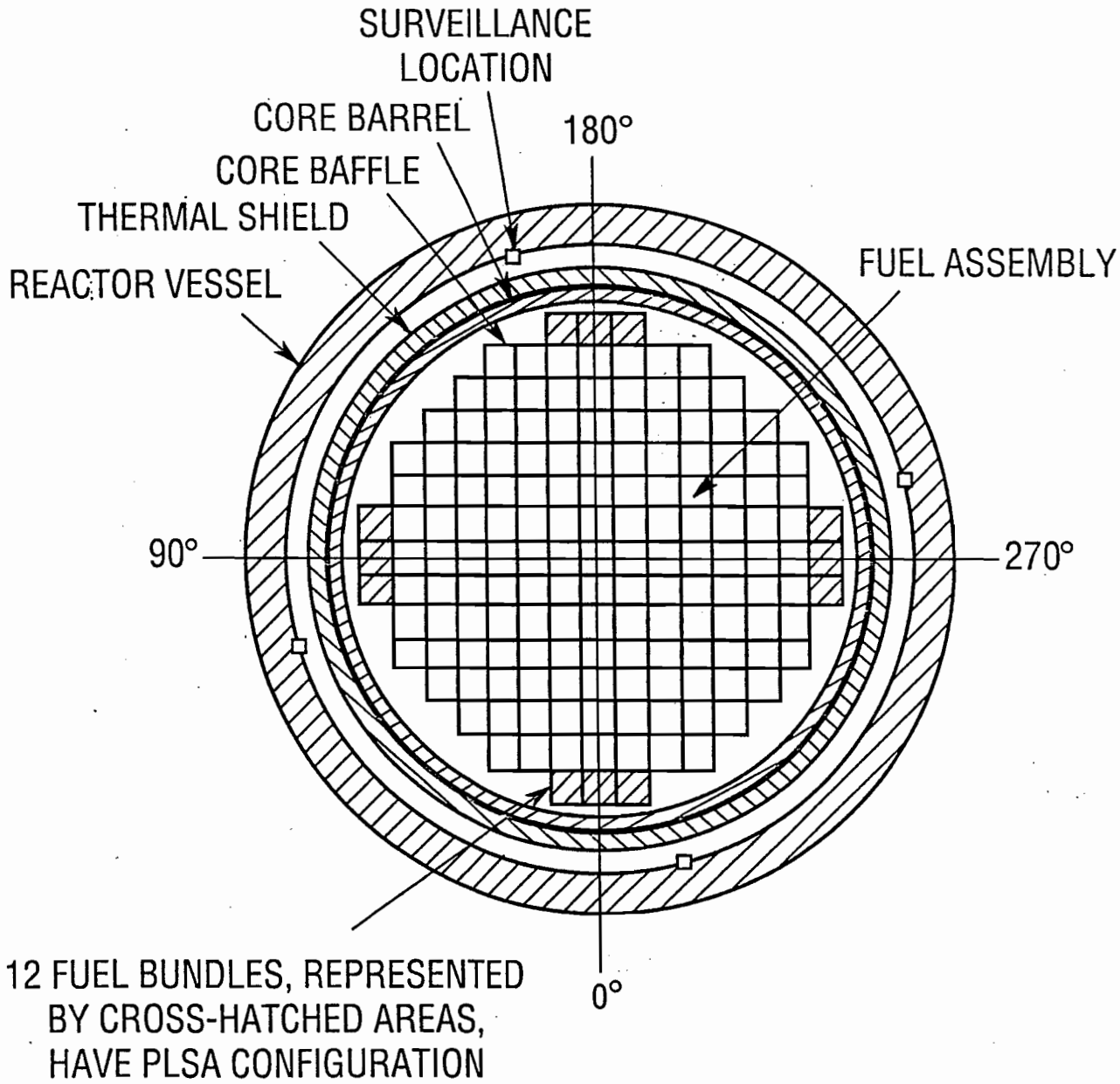
- Reactor Pressure Vessel Fluence is Required for Determination of Vessel Embrittlement and Lifetime

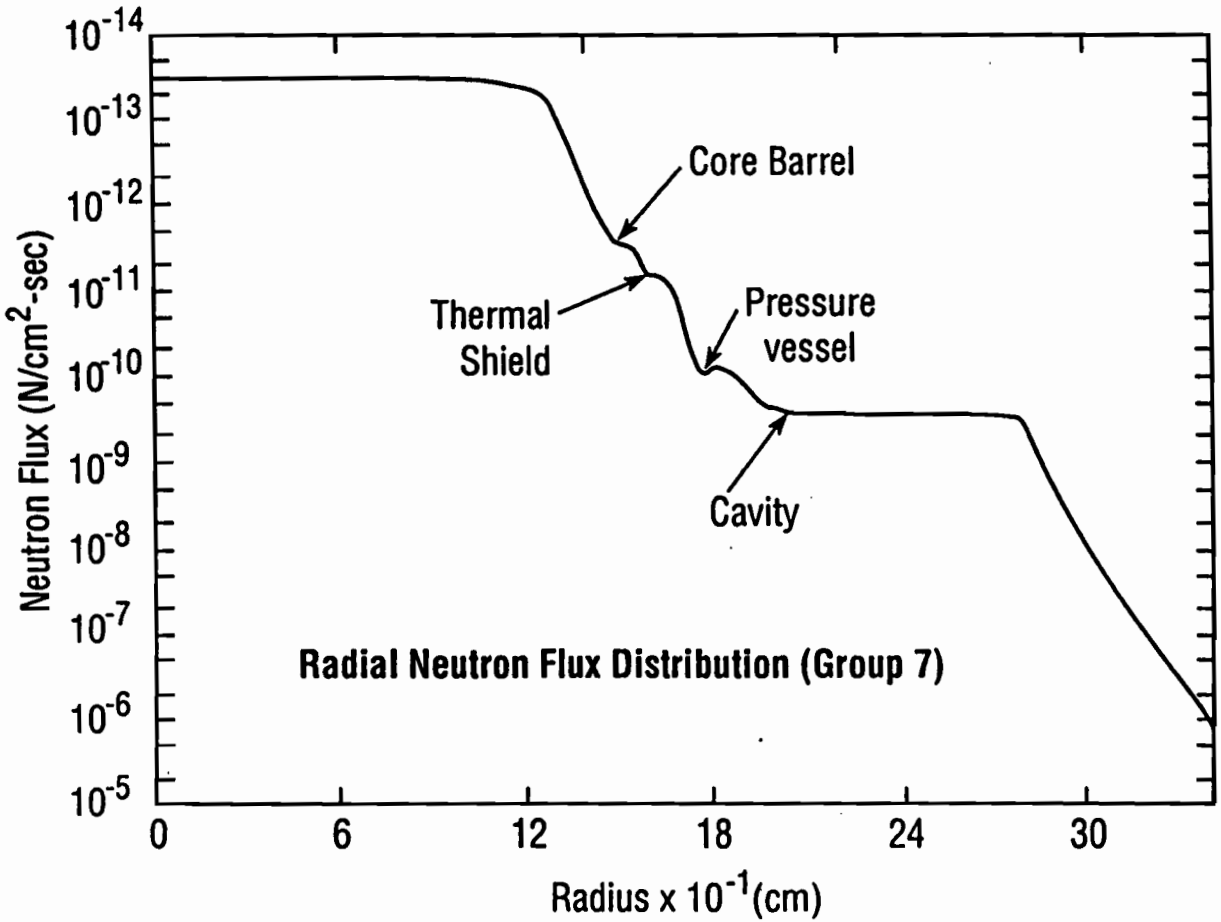
- Vessel Fluence is used to Determine the Adjusted Reference Temperature for the Nil-Ductility Transition RT_{NDT}

- The "PTS Rule", 10 CFR PART 50.61, Requires the Determination of the Fluence for RT_{PTS}

Background

- Neutron Fluence Undergoes Several Decades of Attenuation Between the Core and Vessel
- Fluence Calculation is Therefore Sensitive to
  - Material and Geometry Representation of the Core and Vessel Internals
  - Space/Energy Neutron Source
  - Transport Calculation Numerical Schemes
- Detailed Multigroup/Multidimensional Analysis is Required for an Accurate Fluence Estimate


SURVEILLANCE LOCATION
CORE BARREL
CORE BAFFLE
THERMAL SHIELD
REACTOR VESSEL
180°
FUEL ASSEMBLY
90°
270°
0°
12 FUEL BUNDLES, REPRESENTED BY CROSS-HATCHED AREAS, HAVE PLSA CONFIGURATION


Neutron Flux (N/cm2-sec)
10-14
10-13
10-12
10-11
10-10
10-9
10-8
10-7
10-6
10-5
0
6
12
18
24
30
Radius x 10-1(cm)
Core Barrel
Thermal Shield
Pressure vessel
Cavity
Radial Neutron Flux Distribution (Group 7)

Background

- Wide Range of Fluence Methods are used: Cross Section Sets, Physics Approximations (Source and Axial Treatment) and Codes
- Limited Number and Uncertainty of Capsule Benchmark Data
- For Certain Vessels Limited EOL Margin to RT_{PTS} Limits

Presentation Overview

- Review Regulatory Guide DG-1053
- Resolution of Industry Comments on DG-1053
- PWR and BWR Pressure Vessel Benchmark Problems - NUREG/CR-6115

Regulatory Guide DG-1053

- Developed
  - For - Materials Engineering Branch
    Division of Engineering Technology
    Office of Nuclear Regulatory Research
  - By - Brookhaven National Laboratory
    - National Institute of Standards and Technology
    - Oak Ridge National Laboratory

Summary of Regulatory Guide DG-1053

- Purpose
  - Document calculation and measurement methods for determining pressure vessel fluence that are acceptable to NRC

- Scope
  - Vessel fluence determination for input to CFR 50.61 RT_{PTS}, input to Regulatory Guide 1.99 and Appendix-G

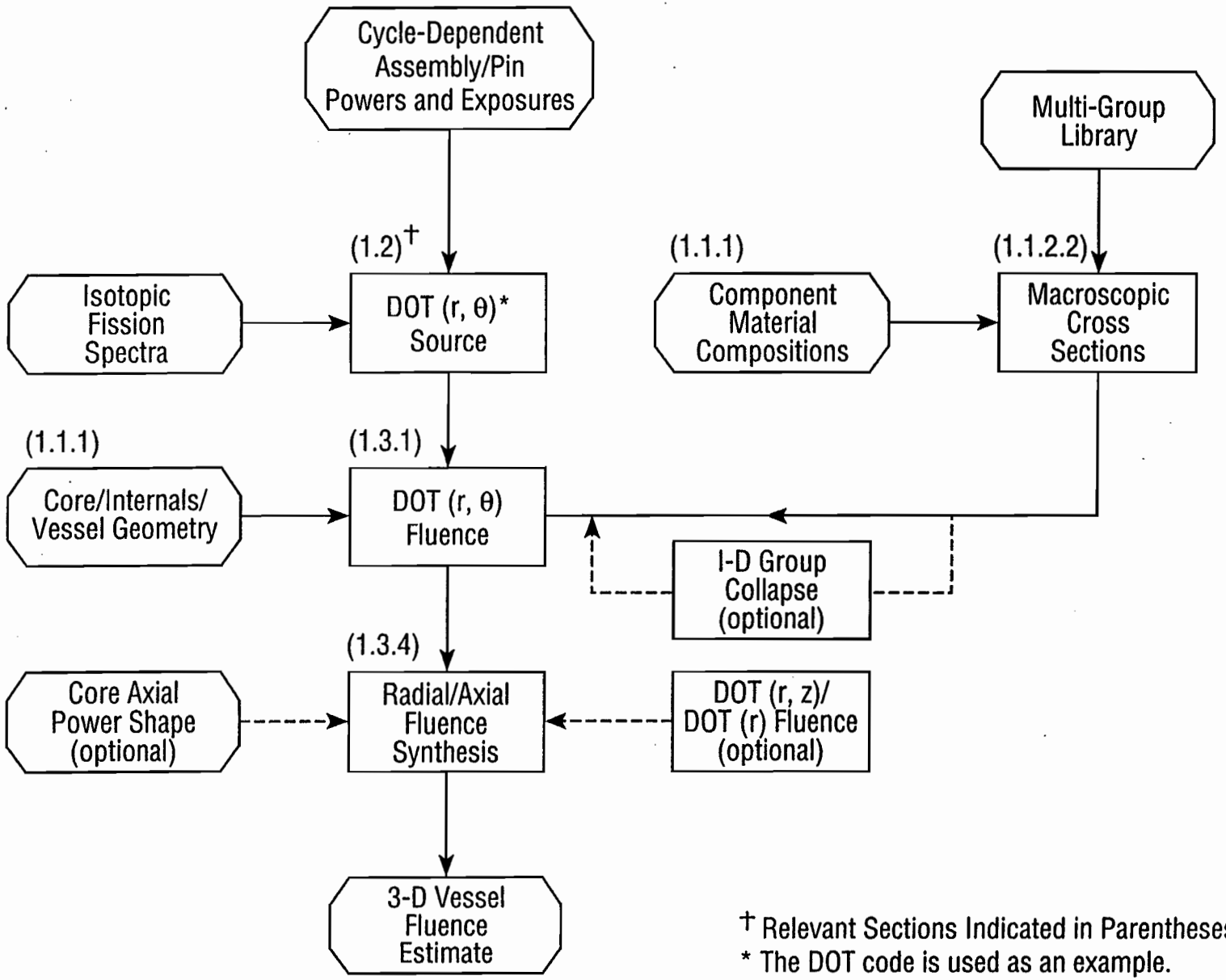
Summary of Regulatory Guide DG-1053

- Includes
  - Detailed description of fluence calculation and measurement methods
  - Procedures for qualification of calculations and measurements
  - Table of specific modeling, dosimetry,qualification and reporting requirements
  - Requires calculation of NUREG/CR-6115 pressure vessel fluence benchmark problems for methods qualification

Summary of Regulatory Guide DG-1053

- Primary Calculational Tasks
  - Determination of Geometrical and Material Composition Data
  - Determination of the Core Neutron Source
  - Transport Theory Calculation of the Neutron Flux from Core to Vessel and Cavity

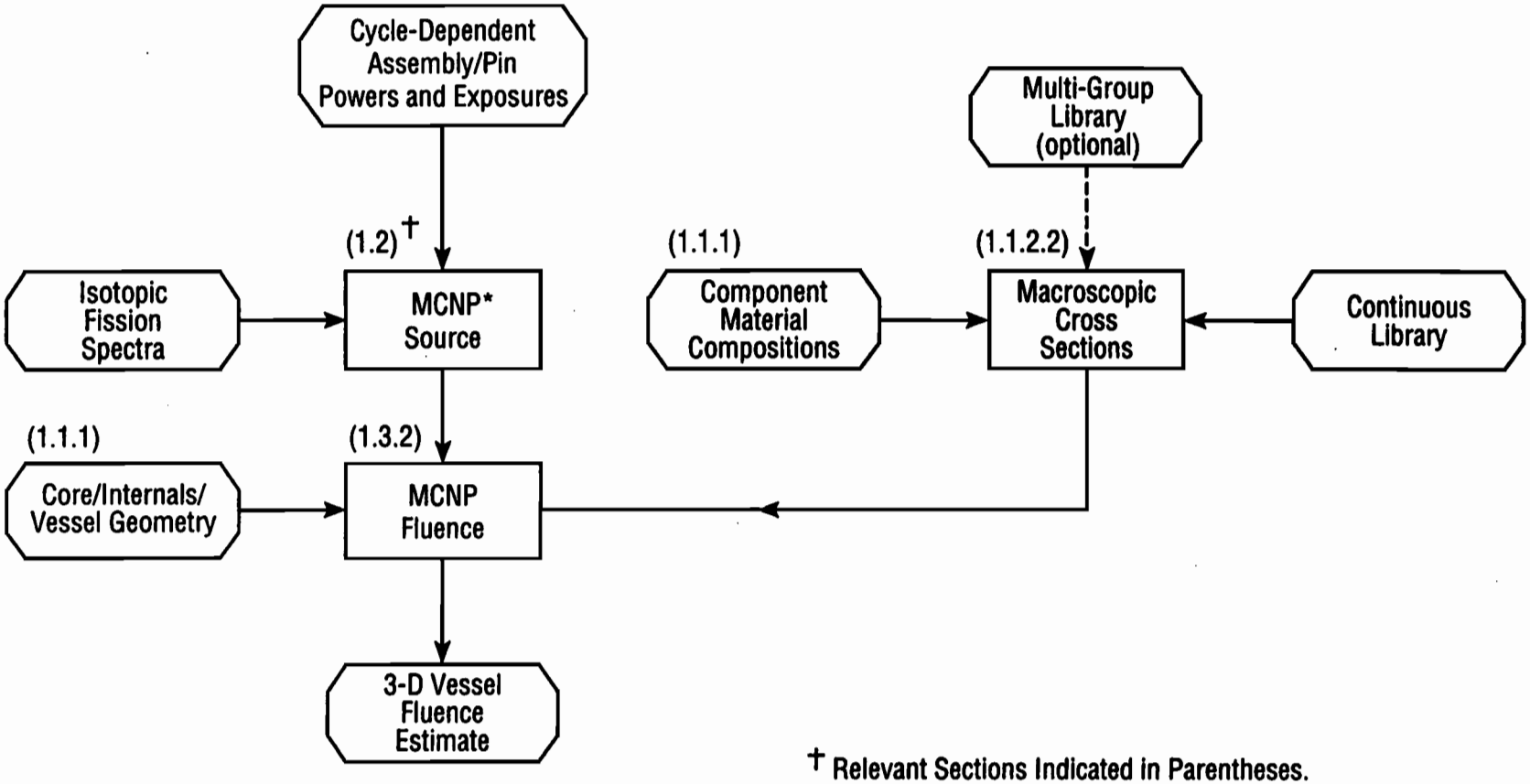
Discrete Ordinates Calculation Methodology



† Relevant Sections Indicated in Parentheses.
* The DOT code is used as an example.

FIGURE 1

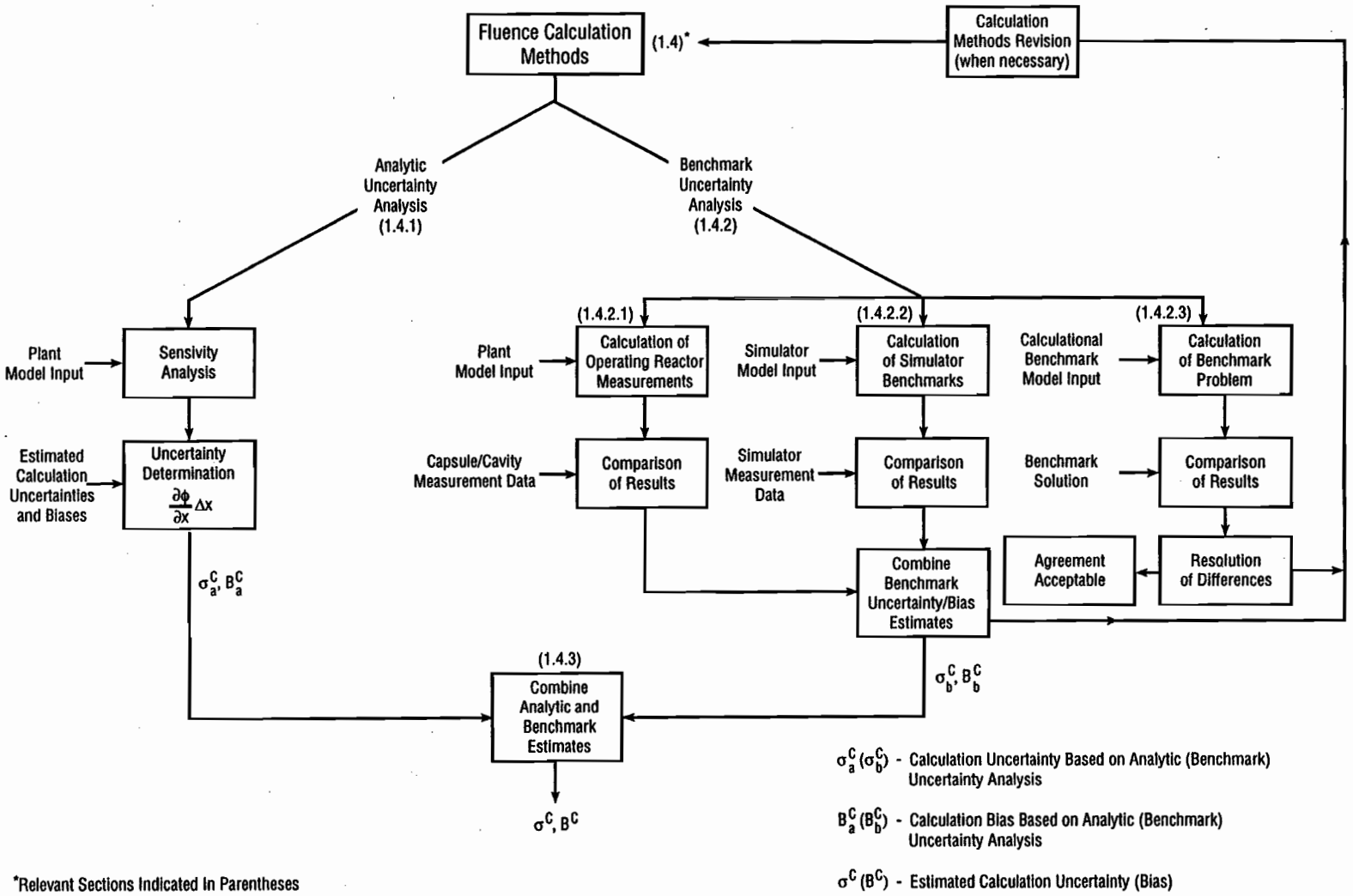
Monte Carlo Calculation Methodology



† Relevant Sections Indicated in Parentheses.
* The MCNP code is used as an example.

FIGURE 2

Calculation Method Qualification Procedure



$\sigma_a^C (\sigma_b^C)$ - Calculation Uncertainty Based on Analytic (Benchmark) Uncertainty Analysis

$B_a^C (B_b^C)$ - Calculation Bias Based on Analytic (Benchmark) Uncertainty Analysis

$\sigma^C (B^C)$ - Estimated Calculation Uncertainty (Bias)

*Relevant Sections Indicated In Parentheses

FIGURE 3

Fluence Determination

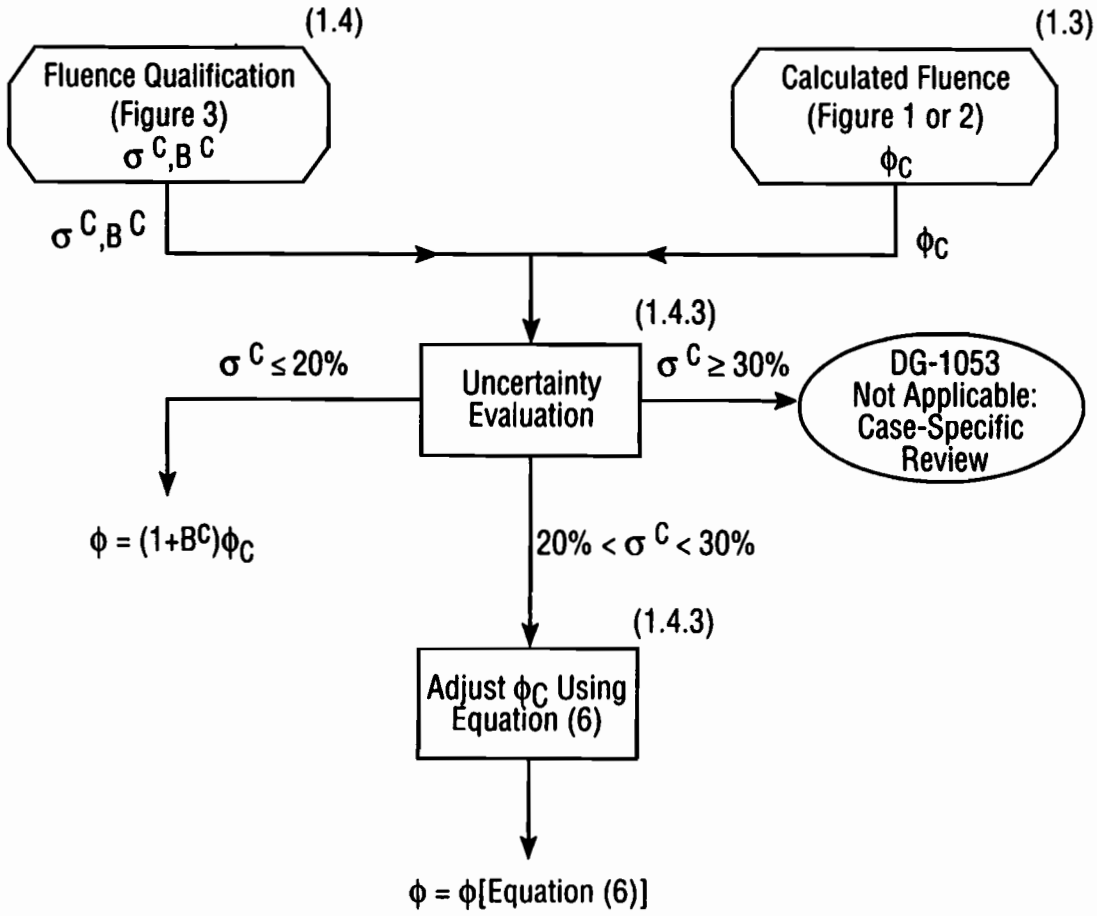
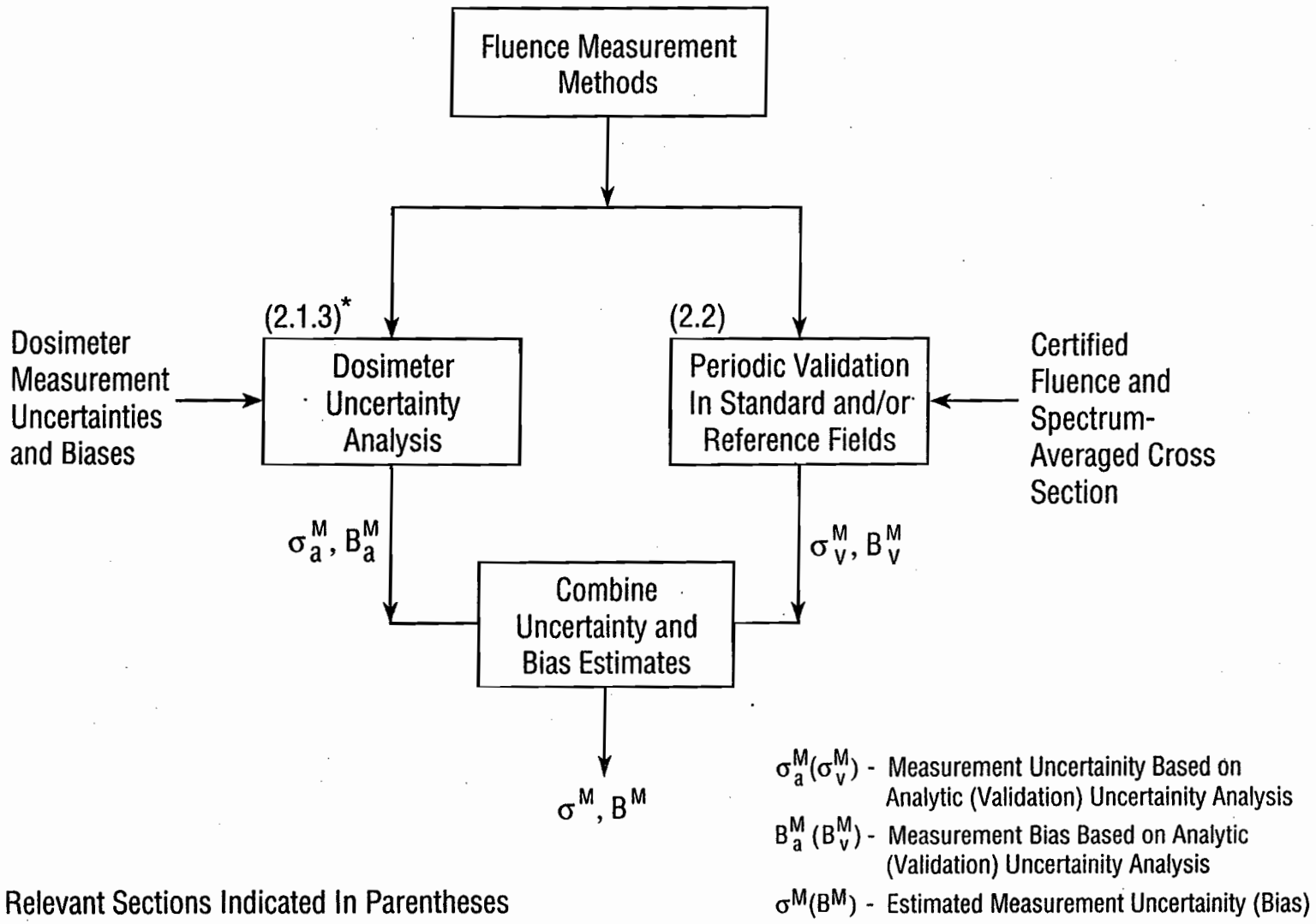


FIGURE 4

Measurement Qualification Procedure



*Relevant Sections Indicated In Parentheses

Figure 5

Summary of Regulatory Guide DG-1053

- Fluence Calculational Methods
  - Best-Estimate Rather Than Bounding Approach
  - Provides Accuracy of < 20% (1-σ)
  - Energy Range from 15 MeV to 0.1 MeV
- Qualification Via Benchmarking and Uncertainty Analysis

Summary of Regulatory Guide DG-1053

- Applicability
  - Fluence Input for Appendix-G and Reg. Guide 1.99
  - Both PWR and BWR Core/Vessel Geometries and Fuel Designs
  - Vessel Fluence Reduction Designs (PLSAs, Low-Leakage Cores, etc.) and Life Extension Calculations

Status of Regulatory Guide DG-1053

- NRC Pre-Release Reviews (1993)
  - ACRS Subcommittee
  - ACRS Committee
  - CRGR
- NRC Release for Comment
- Formal Review Meeting with Industry
- Round-1 Industry Comments Evaluated and Incorporated where Appropriate

Status of Regulatory Guide DG-1053

- September 18, 1996 Meeting with Industry to Review DG-1053
- Round-2 Industry Comments Evaluated and Incorporated where Appropriate
- Incorporation of Monte Carlo Transport Methods and BWR Benchmark Problem in DG-1053 and NUREG-6115
- September 28, 1999 NRC/Industry Meeting to Review DG-1053

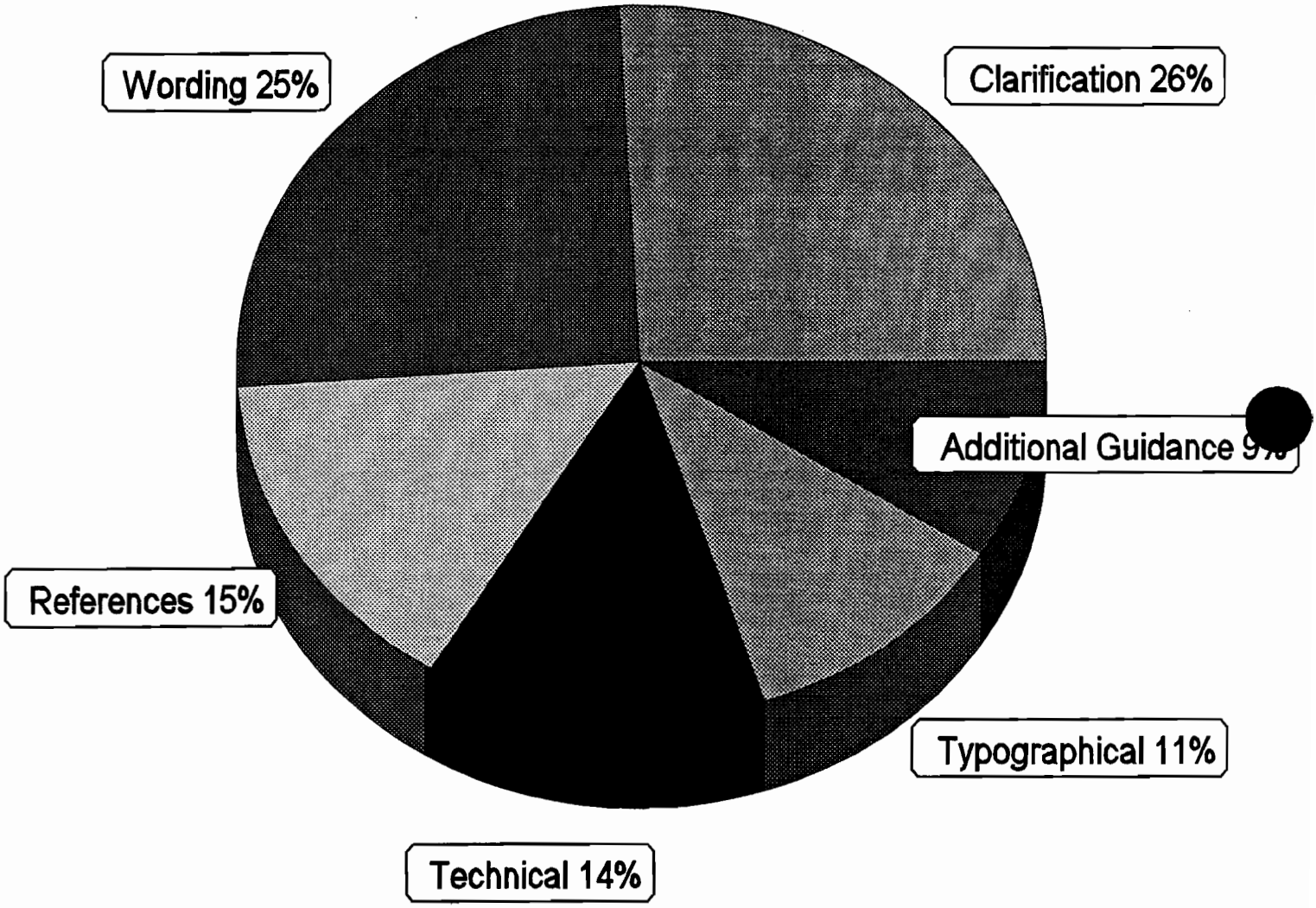
Status of Regulatory Guide DG-1053

- September 28, 1999 - Industry Requests Resolution of Comments Document (RCD) which Provides Basis for Resolution of Previous Comments
- March 2000 - Resolution of Comments Document Provided for Stakeholders Review
- May 2000 - Industry Comments on DG-1053 Received
- August 2000 - Evaluation of Industry Comments Completed and Appropriate Changes Incorporated in DG-1053
- Early 2001 - Final Release of Regulatory Guide

Resolution of DG-1053 Comments

- Resolution of Comments
  - Comments from: NEI, PSU (A. Haghighat), Bechtel Power (W. C. Hopkins) and DWM
  - Concerned editing, organization, methods and qualification
  - Includes recommendations to both relax and tighten requirements
  - Response to <u>all</u> comments and decision basis
  - Response based on Team consensus
  - Resulting changes and additions included in DG-1053

COMMENT TYPES



Resolution of DG-1053 Comments

- Comment Resolution Approach
  - All comments thoroughly evaluated for possible inclusion
  - Requirements for inclusion
    - Consistent with scope and purpose
    - Technically valid
    - Significant part of fluence determination
    - Consistent with presently accepted methods
    - Level of detail not overly prescriptive and should allow sufficient freedom to apply engineering judgement

Resolution of DG-1053 Comments

- Comment - The DG-1053 methods are not applicable to BWRs because of the three-dimensional spatial dependence of the void and fuel composition in BWR cores.

- Response -

  - BWRs and PWRs have similar spatial variability in fuel compositions which is accounted for in the DG-1053 methodology.

  - DG-1053 requires accounting for the axial and radial dependence of the core void fraction.

  - The application of the DG-1053 methodology to an operating BWR configuration is provided in NUREG/CR-6115.

Resolution of DG-1053 Comments

- Comment - DG-1053 should include guidance on the selection of the weighting factors (w_i) used to determine the fluence calculational uncertainty.

- Vessel fluence calculational uncertainty determined by a weighted mean -

$$\sigma = W_A \sigma_A + W_O \sigma_O + W_S \sigma_S$$

  - σ_A - Based on analytic uncertainty propagation.
  - σ_O - Based on comparisons with operating reactor surveillance measurements.
  - σ_O - Based on comparisons with simulator experiments.

Resolution of DG-1053 Comments

- In practice, the weighting factors (w_A, w_O, w_S) depend on application-specific details which can vary widely. For example,
  - Accuracy and completeness of the input uncertainties used in the analytic uncertainty propagation (e. g., availability of vessel diameter measurements and core neutron source uncertainty).
  - Reliability and completeness of the benchmark measurement data (e. g., surveillance capsule data).
  - Level of Quality Assurance for each uncertainty estimate.

Resolution of DG-1053 Comments

- Values of the weights (w_i) are intentionally not specified in the Guide in order to
  - Allow sufficient flexibility to accommodate all cases
  - Allow the application of engineering judgement to accommodate plant-specific cases
- The Guide has been revised to provide a typical example of the weight selection, illustrating the factors to be considered and the determination of the weights

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE

27) Comment: (a) Additional guidance should be provided indicating how the uncertainty estimates based on the operating reactor measurements and the simulator benchmarks are to be combined to determine the fluence uncertainty estimate.	27) Response: (a) "Because the weighting of the analytic and benchmark uncertainty estimates depends on the details of the specific application which can vary widely, it is not possible to specify a practical and generically valid prescription for determining the weights. However, the following example illustrates factors that should be considered. In the case where as-built measurements of the vessel diameter are available and reasonable estimates of the core neutron source and other input uncertainties can be determined, the analytic uncertainty estimate should be reliable and have an uncertainty of ~ 15% (1-sigma). Assume that there are a statistically significant number of accurate ($\sigma < 5\%$) operating reactor measurements and the uncertainty estimate based on this data has an uncertainty of ~ 20%. The uncertainty estimate based on the vessel simulator measurements is assumed to be less certain and has an uncertainty of ~ 25%. Using a weighted mean in which the weight is inversely proportional to the square of the standard deviation of the estimate (i.e., σ^{-2}), the weights are $w_A = 0.5$, $w_O = 0.3$ and $w_s = 0.2$.

NUREG/CR-6115 Pressure Vessel Fluence Benchmark Problems

- Purpose
  - Insure accurate fluence predictions and quantify uncertainty
  - Standardize vessel fluence methods
  - Streamline licensing process

Application of Benchmark Problems

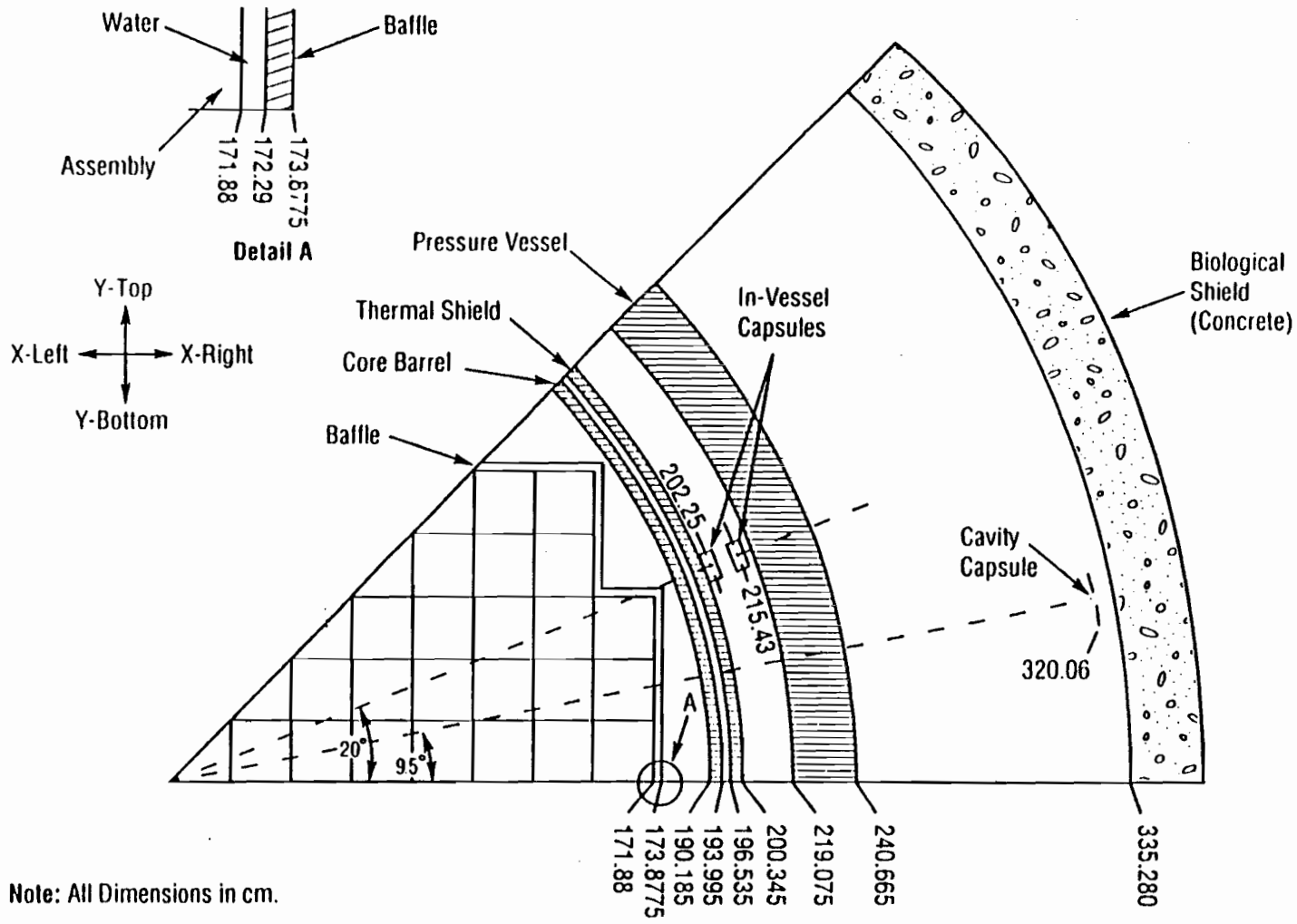
- Fluence Methods Provided in DG-1053
- NUREG/CR-6115 Provides Problem Definitions and Reference Solutions
- Licensee Calculates the Benchmark Problems
- Comparisons to the Reference Solutions Provided to NRC
- Fluence Methods Accepted (in part) Based on Agreement with Reference Solutions

Problem Definition

- Core Types Include
  - PWR - Standard Core, Low-Leakage Core (LL), and Partial Length Shield Assembly (PLSA) Core
  - BWR - Standard Core
- Detailed Description of Problem Materials, Geometry and Pin-Wise Source
- Typical Operating Reactor Geometry and Materials
- Complete Fluence Analysis Involving the Execution of Steps Required for the Determination of RT_{PTS} Input

PWR Standard Core Benchmark Problem

- Geometry
  - Typical core-to-cavity dimensions and materials including a thermal shield and biological shield
- Core Neutron Source
  - Pin-wise power distribution with fuel isotopics vs burnup
- Complete and Detailed Set of Calculation Input Provided
- Dosimeters Located in Standard Wall Capsule and in Cavity
- ENDF/B-VI Dosimeter Cross Sections Provided



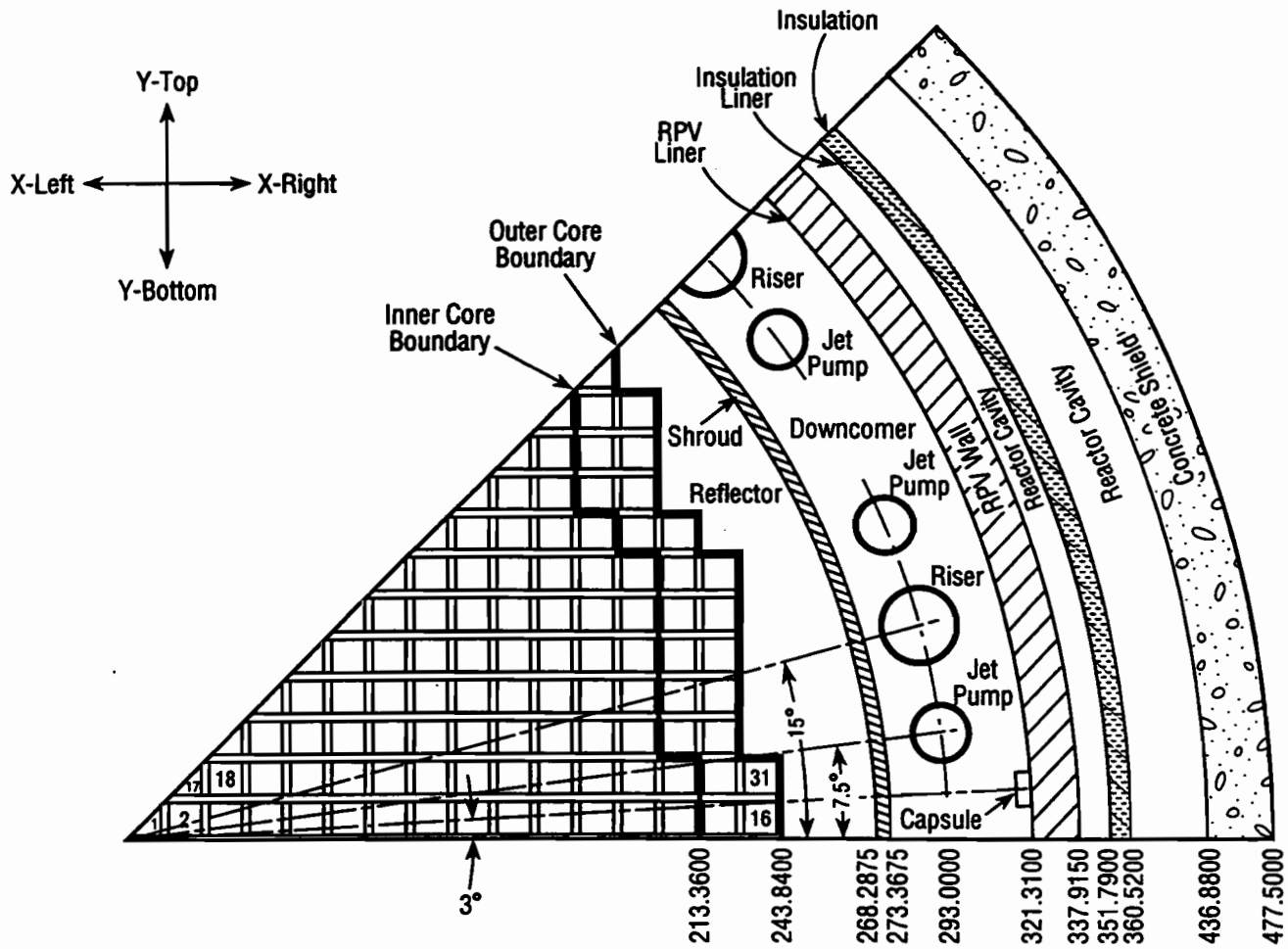
Note: All Dimensions in cm.

Location of Surveillance Capsules

Fig. 2.2.3

Figure 2.2.2.1

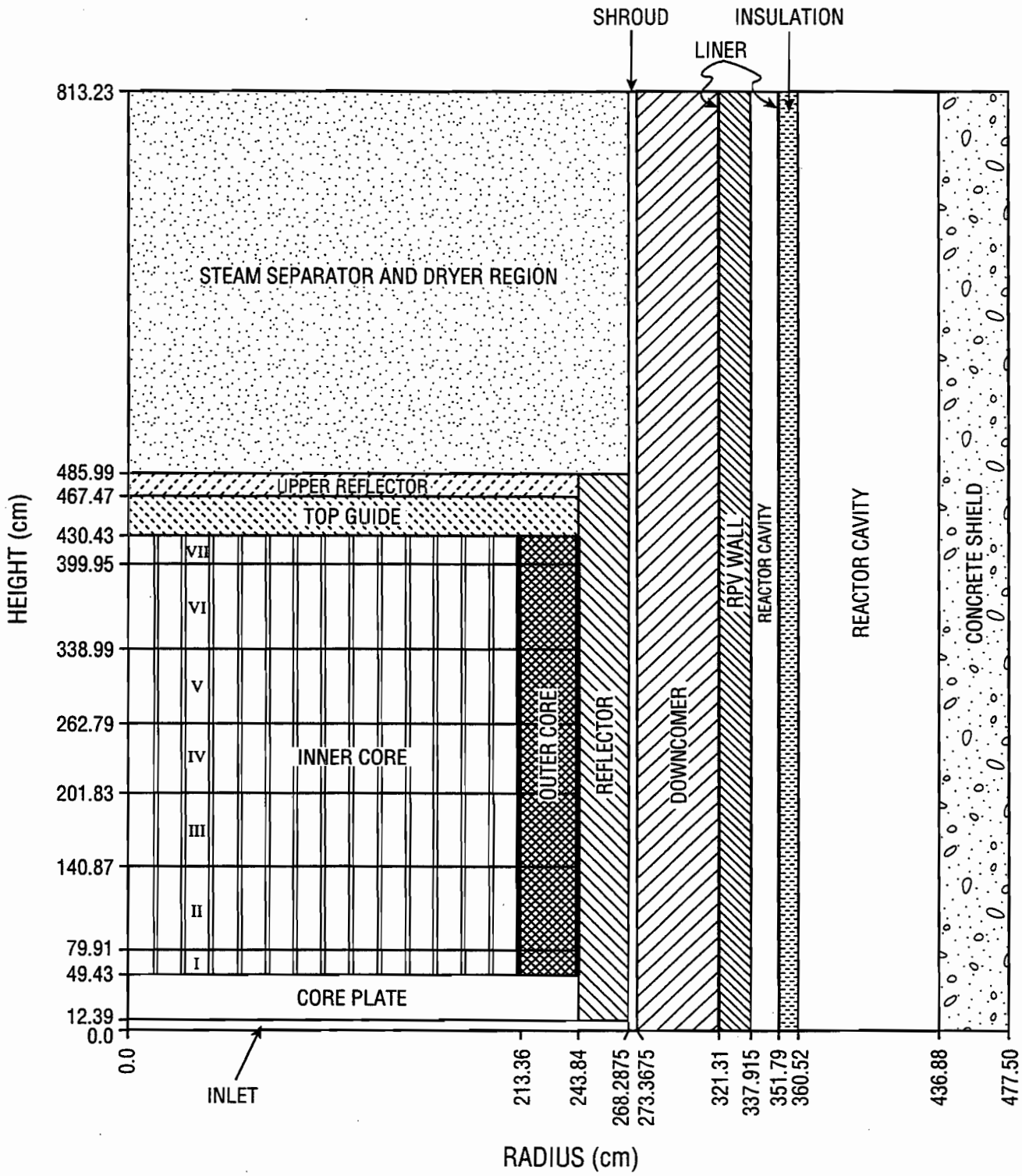
BWR Planar Geometry



NOTE: All Dimensions in cm

Figure 2.2.2.2

BWR Axial Geometry



NOTE: All Dimensions in cm

Table 2.2.1

Standard Core Loading

Basic Design Data

Reactor		Material
Thermal Power	2527.73 MW(TH)	--
Core Inlet Temp.	536°F	--
Core Operating Pressure	2010 psia	--
Baffle Thickness	1.5875 cm	SS-304
By-Pass	--	H_2O (560°F 2010 psia)
Inner Radius of Core Barrel	190.185 cm	--
Barrel Thickness	3.81 cm	SS-304
Inner Inlet Thickness	2.54 cm	H_2O (536°F 2010 psia)
Inner Radius of Thermal Shield	196.535 cm	--
Thermal Shield Thickness	3.81 cm	SS-304
Outer Inlet Thickness	18.095 cm	H_2O (536°F 2010 psia)
Inner Radius of Liner Clad	218.440 cm	--
Vessel Liner Clad Thickness	0.635 cm	SS-304
Vessel Thickness	21.59 cm	SA-302B
PV Insulation Air Thickness	1.835 cm	Air
PV Insulation Thickness	10.16 cm	PV Insulation
Cavity Thickness	82.62 cm	Air
Inner Radius of Biological Shield	335.280 cm	--
Bio-Shield Liner Thickness	0.635 cm	SA-302B
Bio-Shield Thickness	213.36 cm	Concrete

Table 2.2.3

Standard Core Loading

Design Specification Material Compositions

Mixture	Component	Atom Densities (atom/b*cm)
Core	H	2.82500E-02
	B	2.69200E-05
	O	1.41200E-02
	C	2.58100E-05
	O (fuel)	1.28500E-02
	Al	3.56000E-04
	Cr	1.60800E-05
	Fe	2.20500E-05
	Ni	3.87700E-05
	Zr	5.44800E-03
	U-235	1.12000E-04
	U-238	6.03000E-03
	Pu-239	2.20000E-05
	Pu-240	7.12000E-06
Baffle (SS-304)	Cr	1.85300E-02
	Mn	1.75200E-03
	Fe	5.80700E-02
	Ni	8.57400E-03
Bypass+ (560 F)	H	4.92900E-02
	O	2.46400E-02
	B	4.90000E-06
Core Barrel (SS-304)	Cr	1.85300E-02
	Mn	1.75200E-03
	Fe	5.80700E-02
	Ni	8.57400E-03
Inlet Water Gap (536 F)	H	5.09600E-02
	O	2.54800E-02
	B	5.10000E-06
Thermal Shield (SS-304)	Cr	1.85300E-02
	Mn	1.75200E-03
	Fe	5.80700E-02
	Ni	8.57400E-03
RPV Liner (SS-304)	Cr	1.85300E-02
	Mn	1.75200E-03
	Fe	5.80700E-02
	Ni	8.57400E-03
Cavity (Air)	N	3.20000E-05
	O	8.00000E-06

+This composition is also to be used in the water region between the fuel assembly and the core baffle (see Figure 2.2.1).

Benchmark Problem Solution

- Tabulated Solution Based on Standard Fluence Analysis Predictions
- Problem Solution Includes
  - Fluence > 1-MeV, > 0.1 MeV, dpa and spectrum
  - Accelerated and wall capsules and vessel internal and cavity locations
  - Dosimeter reaction rates at capsule and cavity locations
  - Fluence sensitivity calculations

Benchmark Problem Solution

- Calculational Methods Based on DG-1053
  - DORT S_8 P_3 transport
  - (r, θ) / (r,z) synthesis
  - BUGLE-93 / ENDF/B-VI cross sections
  - ENDF/B-VI dosimeter cross sections provided
  - ENDF/B-VI fission spectrum

Table 4.2.1

Standard Core Loading

Flux (E>1.0 MEV) At Pressure Vessel

z	125.488cm	125.488cm	129.186cm	129.186cm	140.282cm
θ	0-T	1/4 T	1/2 T	3/4 T	T
1	3.14799E+10	1.78931E+10	8.69114E+09	4.00968E+09	1.63867E+09
2	3.14939E+10	1.78948E+10	8.69181E+09	4.01204E+09	1.63590E+09
3	3.15515E+10	1.79309E+10	8.71023E+09	4.01916E+09	1.63546E+09
4	3.18090E+10	1.80707E+10	8.77347E+09	4.04608E+09	1.64438E+09
5	3.22184E+10	1.82947E+10	8.87711E+09	4.09152E+09	1.65943E+09
6	3.27523E+10	1.85911E+10	9.01599E+09	4.15139E+09	1.67925E+09
7	3.34669E+10	1.89955E+10	9.20569E+09	4.23439E+09	1.70608E+09
8	3.44603E+10	1.95427E+10	9.45942E+09	4.34323E+09	1.74117E+09
9	3.54574E+10	2.00747E+10	9.70318E+09	4.44796E+09	1.77511E+09
10	3.63728E+10	2.05806E+10	9.93677E+09	4.54712E+09	1.80693E+09
11	3.74744E+10	2.11727E+10	1.02041E+10	4.66105E+09	1.84307E+09
12	3.85964E+10	2.17730E+10	1.04749E+10	4.77398E+09	1.87884E+09
13	3.96873E+10	2.23491E+10	1.07308E+10	4.88026E+09	1.91205E+09
14	4.06848E+10	2.28693E+10	1.09581E+10	4.97269E+09	1.94069E+09
15	4.15408E+10	2.33066E+10	1.11436E+10	5.04551E+09	1.96229E+09
16	4.22000E+10	2.36185E+10	1.12659E+10	5.08990E+09	1.97516E+09
17	4.25965E+10	2.37640E+10	1.13064E+10	5.09949E+09	1.97593E+09
18	4.25805E+10	2.36866E+10	1.12453E+10	5.06928E+09	1.96689E+09
19	4.21634E+10	2.34630E+10	1.11362E+10	5.02312E+09	1.95418E+09
20	4.20061E+10	2.32440E+10	1.10309E+10	4.98065E+09	1.93913E+09
21	4.11917E+10	2.27246E+10	1.08174E+10	4.90050E+09	1.91354E+09
22	4.02185E+10	2.21933E+10	1.06133E+10	4.82491E+09	1.89446E+09
23	3.88038E+10	2.17186E+10	1.04642E+10	4.77406E+09	1.88876E+09
24	3.82263E+10	2.15915E+10	1.04226E+10	4.75869E+09	1.88401E+09
25	3.73962E+10	2.12952E+10	1.03327E+10	4.72600E+09	1.86809E+09
26	3.67420E+10	2.09653E+10	1.02152E+10	4.68137E+09	1.85169E+09
27	3.63552E+10	2.07451E+10	1.01082E+10	4.63723E+09	1.83997E+09
28	3.64033E+10	2.06613E+10	1.00430E+10	4.60702E+09	1.83573E+09
29	3.65714E+10	2.06166E+10	1.00083E+10	4.59093E+09	1.83048E+09
30	3.70498E+10	2.05760E+10	9.92482E+09	4.54562E+09	1.80541E+09
31	3.68078E+10	2.04258E+10	9.81075E+09	4.48443E+09	1.78065E+09
32	3.62594E+10	2.02359E+10	9.70994E+09	4.43473E+09	1.76407E+09

MCNP Calculations of Benchmark Problems

- Based on NUREG/CR-6115 Problem Definitions
  - Standard core
  - PLSA core
  - BWR core
- Exact Three-Dimensional Geometry
  - Explicit core/baffle/shroud/barrel geometry
  - Pin-wise source description for peripheral assemblies
- Multi-Group ENDFB/VI Library Based on BUGLE-93

MCNP Calculations of Benchmark Problems

- Region-Wise Importance Weighting
- > 1-MeV Fluence Edit at Selected Vessel Inner-Wall Locations
- DORT/MCNP Differences ≲5% Consistent with Methods Uncertainties
  - MCNP statistics
  - DORT geometry
  - DORT numerics
  - DORT synthesis

Fig. 5.3.7

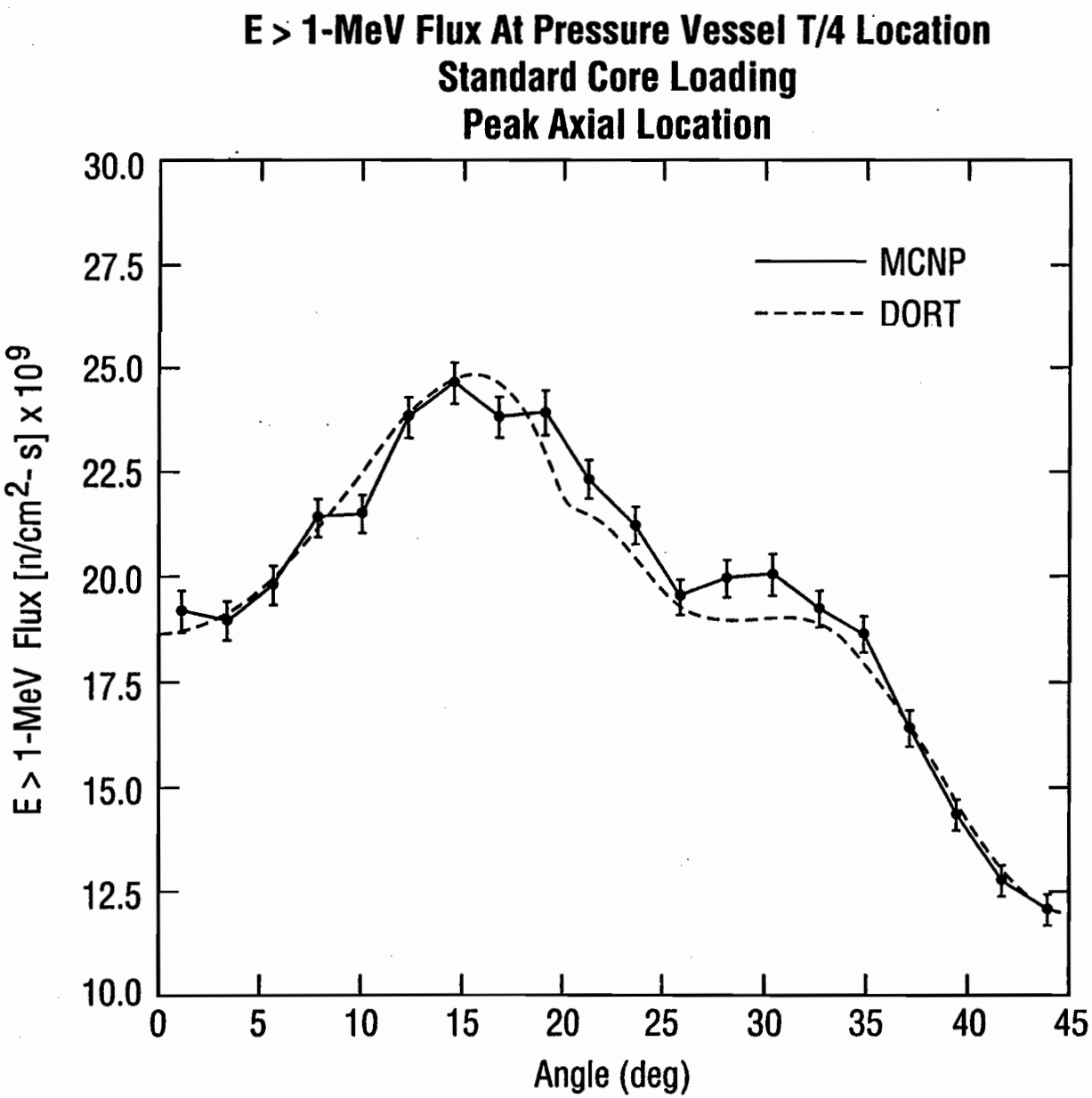

E > 1-MeV Flux At Pressure Vessel T/4 Location
Standard Core Loading
Peak Axial Location
MCNP
DORT
E > 1-MeV Flux [n/cm2- s] x 10^9
30.0
27.5
25.0
22.5
20.0
17.5
15.0
12.5
10.0
0
5
10
15
20
25
30
35
40
45
Angle (deg)

Figure 5.3.9

E > 1-MeV Flux At Pressure Vessel Lower Weld
Partial Length Shield Assembly Core Loading

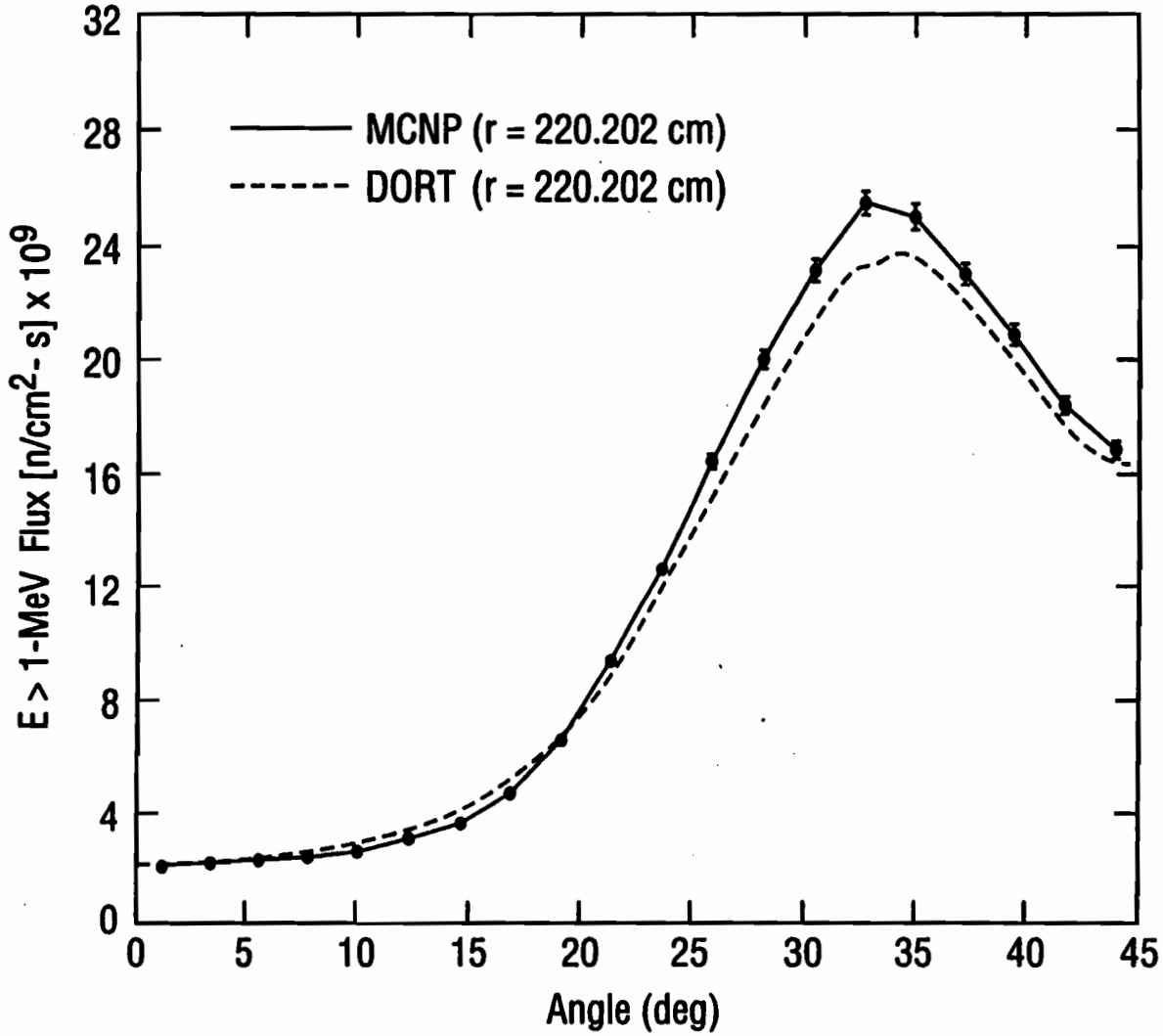


Figure 5.4.5

Comparison of MCNP and DORT E > 1-MeV Flux In Downcomer (r= 278.10 cm) at The Core Axial Midplane

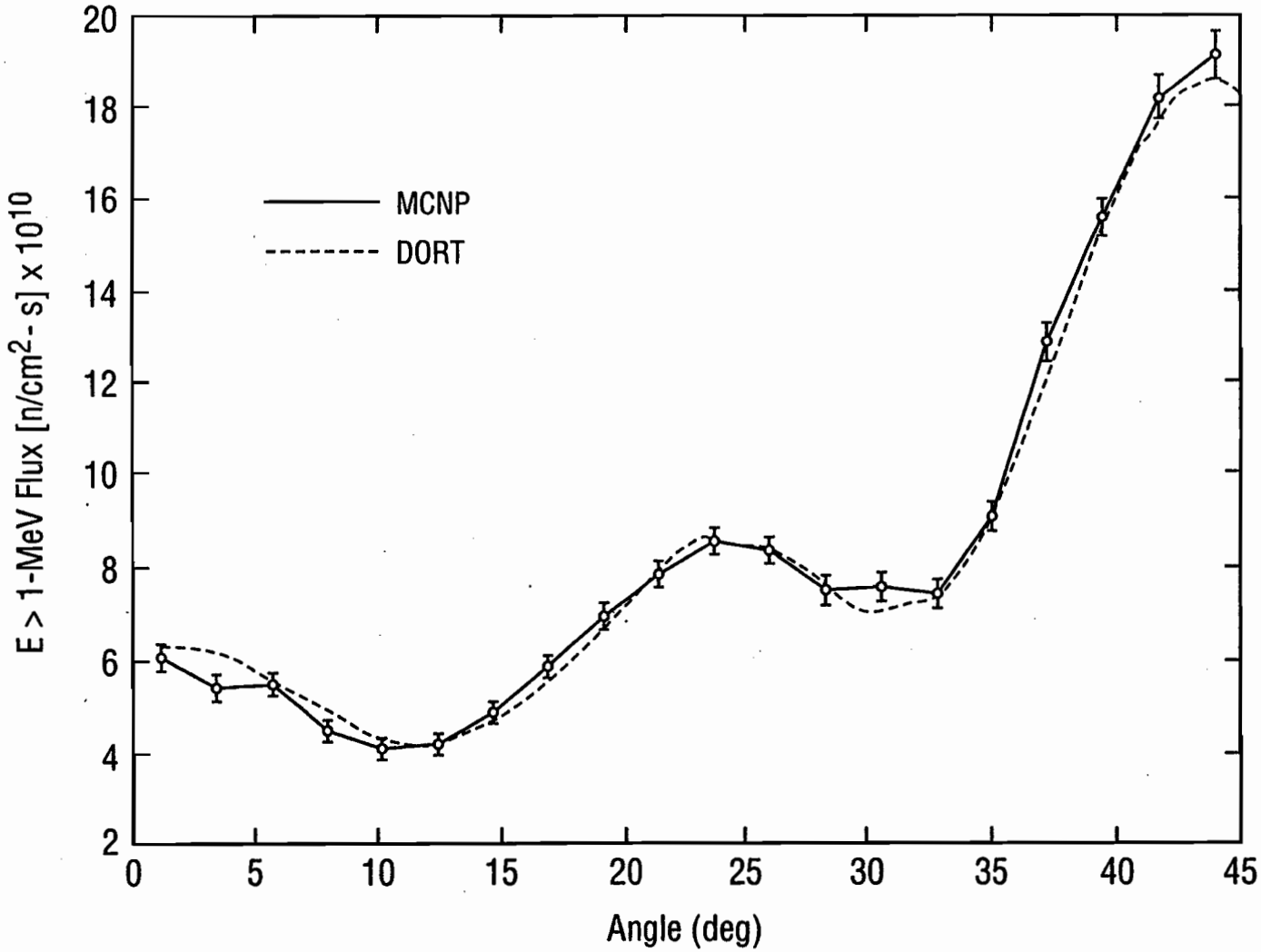
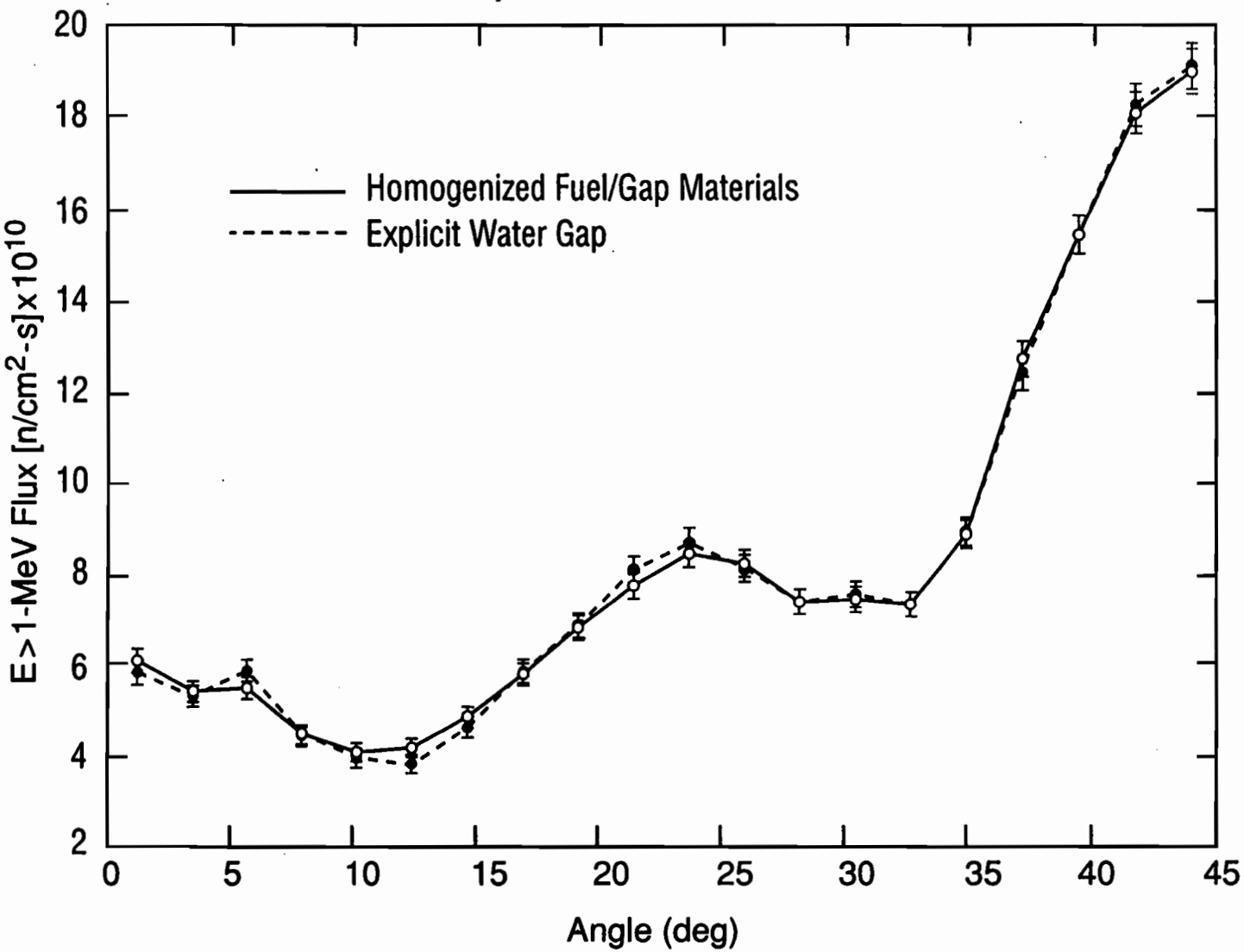


Figure 5.4.7

E > 1-MeV Flux In Downcomer (r = 278.10 cm) at The Core Axial Midplane With Probable Error



RESPONSES TO THE GENERAL COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE	
1) Comment: The final version of NUREG-6115 should be released.	1) Response: The draft version of NUREG/CR-6115 was released in September 1999. The final version of NUREG/CR-6115 will be published in early calendar 2001.
2) Comment: NEI recommends a standard problem round robin exercise among the NRC contractor and industry participants using the draft regulatory guide.	2) Response: The proposed round robin is outside the scope of the DRG-1053. The Guide provides highly detailed guidance on the methods for measuring and calculating fast neutron fluence which satisfy the requirements of both Appendix-G to Part 50 and 10 CFR 50.61. These methods have been employed in several measurement benchmark analyses which have been referenced and are publicly available. The NUREG/CR-6115 Benchmark Problem calculations will be available in early calendar 2001.

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE	
1) Comment: The Resolution of Comments Document includes responses with insights that would be beneficial to the user of the Guide. Specifically Responses IV.8.3, IV.10.3, V.42.3, VI.16.3, VIII.5.3, VIII.12.3 and IX.3.3.	1) Response: Responses IV.8.3, IV.10.3 have been added to the Guide. Responses V.42.3 and VIII.5.3 were included as part of a previous update. Responses VI.16.3, VIII.12.3 and IX.3.3 are considered too detailed and/or are outside the scope of the Guide and are not being included
6) Comment: The Guide should include a reference to ASTM Standard E693 for characterizing neutron exposures of iron and low alloy steels in terms of displacements per atom (dpa), ASTM Standard E-2005 for benchmark testing and ASTM Standard E-1018 for the selection of dosimetry cross sections.	6) Response: The references for these ASTM standards have been added.

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE	
Comments	**Responses**
18) Comment: The Monte Carlo energy bin structure for determining the dosimeter response should satisfy the same requirements as the broad multi-group energy structure given in Regulatory Position 1.1.2.	18) Response: The text has been revised to state that "to insure an accurate integration over the dosimeter cross section, the energy bin structure used to determine dosimeter response scoring should satisfy the requirements of Regulatory Position 1.1.2 concerning the selection of the multi-group library group structure."
21) Comment: The statistical acceptance criteria presented are generally specific to the MCNP Monte Carlo code and should be presented as an example rather than as a minimum set of criteria.	21) Response: The wording in the Guide has been changed to indicate that the statistical criteria are a representative set rather than a minimum set.

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE

27) Comment: (a) Additional guidance should be provided indicating how the uncertainty estimates based on the operating reactor measurements and the simulator benchmarks are to be combined to determine the fluence uncertainty estimate.	27) Response: (a) "Because the weighting of the analytic and benchmark uncertainty estimates depends on the details of the specific application which can vary widely, it is not possible to specify a practical and generically valid prescription for determining the weights. However, the following example illustrates factors that should be considered. In the case where as-built measurements of the vessel diameter are available and reasonable estimates of the core neutron source and other input uncertainties can be determined, the analytic uncertainty estimate should be reliable and have an uncertainty of ~ 15% (1-sigma). Assume that there are a statistically significant number of accurate (σ < 5%) operating reactor measurements and the uncertainty estimate based on this data has an uncertainty of ~ 20%. The uncertainty estimate based on the vessel simulator measurements is assumed to be less certain and has an uncertainty of ~ 25%. Using a weighted mean in which the weight is inversely proportional to the square of the standard deviation of the estimate (i.e., σ^{-2}), the weights are $w_A = 0.5$, $w_O = 0.3$ and $w_s = 0.2$.

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE	
27) Comment: (b) Additional guidance on how the calculation of the benchmark problem will be used to determine the fluence uncertainty should be provided.	27) Response: (b) As suggested, the benchmark problem calculation will be used as a "go/no-go" test rather than as an estimate of the fluence uncertainty. The text of Section 1.4.2.3 has been modified to reflect this change.
32) Comment: The analytic uncertainty analysis requires that model parameter sensitivity calculations be performed. The Guide states that, in some cases, the Monte Carlo sensitivity calculations will require that the change in the model parameter be increased to provide a reliable estimate of the sensitivity. Also, since the sensitivities are not generally linear, additional calculations may be required to determine the nonlinear dependence of the sensitivity.	32) Response: Additional text has been added noting that the dependence of the sensitivity on the perturbed parameter may not be linear and, if the parameter is significantly outside the parameter one-sigma standard deviation, several calculations may be required to determine the nonlinear dependence of the sensitivity.

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE	
Comments	**Responses**
37) Comment: (a) While guidance on the agreement between measured and calculated integral quantities is provided, no guidance on the expected agreement for group fluxes is given. (b) DG-1053 does not define the bias.	37) Response: (a) The requirement to compare measured and calculated group fluxes has been removed. (b) A reference for the bias has been added to Footnote-1.

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE	
37) Comment: (c) More information is required on the level of detail required for benchmark analysis.	37) Response: (c) The benchmark analysis of Section 1.4 requires the calculation of (1) a pressure vessel simulator (e.g., the PCA or PSF measurements) (2) operating reactor dosimetry measurements (either in-vessel, cavity or both) and (3) the NUREG-6115 benchmark. References 65, 72 and 73 provide several examples of benchmark analyses. References 74, 75 and 79 have been added to provide additional examples of benchmark analyses. Calculations of operating reactor dosimetry are well known throughout the industry and are performed for both in-vessel capsules and cavity measurements. Analyses of the benchmark problems are documented in NUREG-6115. The information provided in the references cited in the Guide is considered sufficient detail for performing benchmark analyses.

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE

40) Comment: (a) The referenced simulator benchmarks (References 5, 44, 58-63) and the calculation benchmark (NUREG-6115) do not provide sufficient detail to perform the benchmark calculations.	40) Response: (a) The NUREG-6115 report provides a complete definition of the benchmark problem except for the neutron source which is provided on a separate computer disc. References 74, 75 and 79 have been added to complete the documentation of the benchmarks.
(c) There are no results available that use the DG-1053 method.	(c) The NUREG-6115 Report includes benchmark results that have been determined using the DG-1053 method.
(d) BUGLE-96 is the latest version of the BUGLE cross section data and DOORS is the latest version of the DORT program. These versions of the nuclear data and transport code should be used to analyze the NUREG-6115 benchmark problem.	(d) The codes and nuclear data are being updated continuously. Changes included in the BUGLE-96 nuclear library and the DOORS code are presently being evaluated. If these changes have a significant effect on the application of the NUREG-6115 benchmark problems and there is a need to issue a revision to NUREG-6115, this will be considered.

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE	
40) Comment: (b) Additional guidance on the use of NUREG-6115 in qualifying the calculational methods and what constitutes an acceptable solution should be provided.	40) Response: (b) The following guidance has been added: "The calculation of the benchmark problems allows a detailed assessment and verification of the numerical procedures, code implementation, and the various modeling approximations relative to state-of-the-art solutions for representative operating configurations. If the differences between the benchmark problem calculation and the reference solution are substantially larger than what would be expected based on the differences in the methods approximations and nuclear data used in the two calculations, the agreement is considered unacceptable. In this case, the calculation should be reviewed and the differences between the two solutions explained. When the cause of the deviation is determined to be an error in the calculation, the calculational method must be revised."

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE

40) Comment: (e) In addition to the solutions calculated with DORT and MCNP, the NUREG-6115 Report should provide reference solutions calculated with different codes such as TWODANT and MCBEND.	40) Response: (e) While the use of the TWODANT and MCBEND codes will provide an indication of the code-to-code variation, in view of the extensive benchmarking and testing of these codes, the uncertainty introduced by the code selection is very small when compared to the uncertainty in the plant fluence calculations. Consequently, the use of the DORT and MCNP codes in performing the calculations of NUREG-6115 is considered adequate.
(f) If the purpose of performing the NUREG-6115 benchmark calculations is to validate the implementation of DORT, the DORT electronic input and output files should be provided to insure that the modeling parameters are the same as used in NUREG-6115.	(f) The purpose of the NUREG-6115 benchmark calculations is to validate both the implementation of DORT and the fluence calculation modeling techniques. Providing the DORT electronic files would prevent the validation of the modeling techniques which is considered to be most important.

RESPONSES TO THE COMMENTS PROVIDED BY THE NUCLEAR ENERGY INSTITUTE

Comments	Responses
51) Comment: The term "every few years" in the statement "to ensure long-term consistency and to confirm measurement uncertainty, dosimetry measurements must be performed every few years in well characterized neutron fields," should be quantified. 55) Comment: The guidance conflicts with the least-squares approach.	51) Response: Because of the differences between the various measurement systems and procedures, this period has been purposely left unquantified in order to allow the needed flexibility in its application. 55) Response: While the Guide allows the use of the Least-Squares Adjustment (LSA) method, many licensees do not employ this method and, consequently, the guide provides guidance on both approaches. Where appropriate, the guide has been revised to remove conflicts between the two methods. The suggested wording has been added.

RESPONSES TO THE COMMENTS PROVIDED BY THE PENNSYLVANIA STATE UNIVERSITY A. Haghighat	
Comments	**Responses**
1) Comment: Because of the axial variation of the BWR void distribution, the source and material distributions require three-dimensional representations. Consequently, it is not evident that the DG-1053 methodology is applicable to BWRs. 16) Comment: The verification of the variance reduction method used in the fluence calculation is impractical/impossible.	1) Response: The DG-1053 methodology is applied to a typical BWR in NUREG-6115 and shown to be applicable to BWRs. 16) Response: The variance reduction method is an approximation that can have a substantial effect on the calculated fluence. Consequently, this method should be verified to ensure the fluence prediction is reliable. The Guide states that the verification of the variance reduction method should be qualified by comparing the Monte Carlo variance reduction predictions with estimates made without the application of the variance reduction technique. This verification method has been used successfully in the Monte Carlo analyses of NUREG-6115.

RESPONSES TO THE COMMENTS PROVIDED BY BECHTEL POWER CORP. William C. Hopkins	
Comments	**Responses**
2) Comment: The attached figure should be included in the Guide to clarify the dependence of the fluence determination on the fluence uncertainty.	2) Response: The proposed figure provides a significant clarification of the procedure used to determine the fluence and will be included (as Figure-4) in the Guide.

Modifications to the Safety Goal Policy Statement

Joseph A. Murphy
Office of Nuclear Regulatory Research

14

Background

- In SECY-00-0077 (3/30/2000), staff proposed several modifications to the safety goal policy statement.
- Discussed with ACRS on Feb. 3, 2000.

Background

- ACRS letter of 4/17/2000 recommended:
  - Consideration of a "three-region approach" that defines CDF and large, early release frequency (LERF) boundaries that would be consistent with "adequate protection" and that would define "how safe is safe enough."
  - The concept of risk limits for individual plant applications. These risk limits would be quantitatively expressed limits on CDF and LERF and would possibly consider additional limits for societal risk, land contamination, and a cap on temporary changes in risk.
  - Guidance on defense in depth to address uncertainties in the risk assessments.

Commission SRM

- By SRM dated 6/27/ 2000, Commission approved modifications proposed in SECY-00-0077, with two exceptions:
  - Elevation of the qualitative statement of prevention of severe core damage accidents to a qualitative safety goal was disapproved.
  - Commission also disapproved recommendation to include the statement there be no adverse impact on the environment in the safety goal policy statement.

Commission SRM

- The Commission supported expressing the Commission's intent to protect the environment and to consider the need to minimize adverse environmental impacts in its regulatory decision-making.
- The Commission directed that the policy statement state that safety goals are "goals" and not limits.

Modifications to Policy Statement

- Staff has proceeded as directed to modify the policy statement in accordance with SRM:
  - Reflect Plant-Specific Usage of Safety Goals and definition of "How Safe is Safe Enough".
  - Maintain CDF as a subsidiary objective.
  - Expand treatment of uncertainty, using R.G. 1.174.
  - Incorporate Commission's White Paper definition of defense-in-depth.
  - Delete reference to a general performance guideline. Incorporate a Large Early Release Frequency subsidiary goal of 10^{-5} per reactor year.
  - Incorporated a statement expressing the Commission's intent to protect the environment, and indicating that the NRC considers the need to minimize adverse environmental impacts in its regulatory decision-making.

Proposed Modifications

- Package to ACRS includes a line-in line-out version so exact changes can be evaluated.
- This version also includes references so that the source of the new material added can be ascertained.

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

December 8, 2000

MEMORANDUM FOR: Dana Powers, ACRS Chairman

FROM: John T. Larkins, Executive Director, ACRS

SUBJECT: TRIP REPORT TO MUNICH, GERMANY, ACRS/RSK TECHNICAL EXCHANGE, NOVEMBER 6-10, 2000

Drs. Kress, Bonaca, Uhrig, Larkins, and Wallis met with the German Reactor Safety Committee (RSK) in Munich, Germany on November 6-10, 2000, as part of a continuing technical exchange between the RSK and the ACRS. This meeting was a follow-up to an exchange held at MIT in Cambridge, Massachusetts, in November 1999. A principal focus of the meeting was to discuss safety issues associated with the utilization of digital instrumentation and control (I&C) systems in light water reactors. However, several other matters of interest to the ACRS and NRC were also discussed, as noted below. A listing of the membership of the RSK and the principal subcommittee is attached.

This meeting was reasonably informative, although most of us had some familiarization with the issues presented by RSK. There was enough useful dialogue that a continuation of these bilateral meetings is felt to be worthwhile by all the participants. The RSK also expressed the position that these exchanges were worth continuing and would, in particular, like to explore specific issues on reactor safety and nuclear safety research. I intend to meet with the management of the Office of Nuclear Regulatory Research (RES) and discuss what we heard about ongoing reactor safety research in Germany and potential areas for cooperative research or technical exchanges. Some of the ongoing work [e.g., fire safety, PRA model development, advances in applying computational fluid dynamics (CFD) to thermal-hydraulic analysis] are areas of interest to the Committee. A listing of reactor safety research being supported by the federal government in Germany is attached and more specific information on particular programs is included in the handouts and viewgraphs.

This trip report is a compilation of the insights and comments from me and the four ACRS members who made the trip.

Monday, November 6, 2000

Individual members of the ACRS met with representatives of Gesellschaft für Anlagen und Reaktorsicherheit (GRS) in Garching and at the University of Munich to discuss several matters, including split-up of GRS into a new GRS (that still serves as a consultant to the Federal and State nuclear regulatory authorities) and the Institute für Sicherheitstechnologie (ISTec), a wholly-owned subsidiary that carries out work primarily for utilities and industry. Dr. Uhrig met with Dr. Dieter Wach, who previously visited the University of Tennessee to discuss ongoing work on nuclear power plant diagnostics, surveillance, and control. The topics discussed included the following areas where the work was carried out primarily by ISTec:

1. Vibration analysis performed for utilities on a routine basis as part of a German Government recommended program to monitor the condition of nuclear plants.

2. Research related to machine diagnostics, loose parts monitoring, valve diagnosis and process diagnosis.

3. Information Technologies (IT) Systems, including laboratory and process monitoring, as well as integrated information systems, archiving systems, data bases, and multimedia applications.

4. A brief discussion of waste management that had been added as an additional field of activity since Dr. Uhrig's last visit to GRS.

There were several handouts (a list of references is attached) at these meetings which are available upon request.

Tuesday, November 7, 2000

The ACRS participants were driven to the Siemens I&C Research Laboratory in Erlangen (some two hours from Munich, even at German Autobahn speeds). The meeting consisted primarily of presentations by Siemens personnel, with the emphasis on their new TelePerm digital platform for I&C and safety systems, the philosophy of its design, its special features, and its advantages over traditional analog systems. A significant amount of the discussion focused on software problems and the associated verification and validation, as well as the preparation of the specifications. The specifications were prepared in a graphical language consisting of blocks and links, and automatic code generators wrote the code to implement the specifications implicit in the graphical language. Siemens personnel defended this process as being significantly superior to traditional methods used in the United States, where the emphasis is on assuring that the details of the process and the specifications are correct.

The Siemens group did an excellent job describing the development of systems to generate software to control plant protection systems. This included a discussion of the platform for generic applications in nuclear power plants, which has been reviewed and approved by the NRC staff. One open issue is related to the acceptance criteria and any required testing related to a plant-specific application. Siemens has taken what appears to be an unique approach to issues of diversity in their software system, through the use of a software generator (which generates all of the functional programs) for different functional requirements. Siemens has also developed a code verification program to test the code generator to ensure that there are no mistakes in the software. We had discussions on the development of standards by the Kerntechnischer Ausschuss (KTA) and International Commission on Standards (IEC) for software development and other requirements for commercial, off-the-shelf (COTS) software.

As a separate issue, Siemens appears to be making a concerted effort to rejuvenate its technical staff and has replaced a number of senior managers with younger staff. They, like some of the RSK members, are very much concerned with maintaining and improving technical competency in a number of key areas.

Wednesday, November 8, 2000

We were driven to the GRS/ISTec facilities in Garching, where we spent the morning discussing several issues with their personnel. After a presentation of an overview of the activities of GRS/ISTec, Dr. Dieter Wach made a presentation on digital I&C systems. He covered much of the material discussed on Monday in an abbreviated fashion. However, the comments presented above on our discussions on Monday are applicable to this part of the discussion.

From private discussions with several GRS/ISTec personnel, it is clear that they expect the current plan to phaseout nuclear power when the plants reach an average age of 32 years to be changed, and they eventually expect to extend plant licenses through their normal review of the plants at ten-year intervals.

In the afternoon, we met with the full RSK and initiated discussions on: Boron Dilution Events, Thermal-Hydraulic Aspects, Risk-Informing Regulation, License Renewal, Pebble Bed Modular Reactor, and Reactor Safety Research. Because of time constraints, we were unable to discuss Spent Fuel Issues. Boron dilution events appear to be a serious concern for the RSK, and we spent a significant amount of time discussing the issue.

We were told of seven potential deboration events at the Konvoi plant. Dr. Sonnenburg described several plausible thermal-hydraulic phenomena that can contribute to a deboration event. Even when emergency core cooling is injected into the hot leg in the direction towards the core, it can be carried back to the steam generator by the countercurrent natural convection flow following a small cold leg break. This has been confirmed by experiment. They also have some quite detailed calculations of deborated slug buildup in mid-loop operation with failure of residual heat removal. The reactivity changes associated with the worst scenarios are large, with potential reactor transient powers of ten times full power. When asked about the risk implications, they mentioned rough estimates of 10^{-6} in core damage frequency (CDF). This seems less of a criterion for them, as they tend to look more deterministically at regulation and will probably try to make the problem go away entirely by technical means rather than through risk considerations.

They are doing CFD assessments similar to ongoing work in RES. What we learned from this is that we should perhaps not be as enthusiastic as we were to do away with the test facility at the University of Maryland and should perhaps hear more from the RES staff about its resolution, or lack thereof, of the deboration generic safety issue. This is an area where individuals working on similar problems in the two countries could profitably collaborate.

Thursday, November 9, 2000

On Thursday, we continued discussions on the issues listed in the agenda for Wednesday afternoon. Thermal-hydraulics is one of the top priority research areas in Germany, and we were presented with some of the ongoing work, particularly as it related to boron dilution. The Upper Plenum Test Facility downcomer test of boron mixing and cold water plumes should be of interest to the NRC for analysis related to Pressurized Thermal Shock (PTS). Also, they appear to be fairly well along on developing a CFD code for thermal-hydraulic calculations, which should be of interest to RES. Related to this discussion was the use of the "Wilks"

formula that purportedly describes how many tests need to be run in order to get 95% confidence in the prediction level. This might have an impact on the logical assessment of thermal-hydraulic, and other, codes and their relationships to uncertainty estimation.

Risk-informed Regulation

Dr. Mario Bonaca presented a summary of the ongoing NRC initiatives to risk-inform the NRC regulation. Regulatory Guide 1.174 was introduced as the key document guiding the review and approval of plant changes on the basis of risk insights. In response to questions raised by RSK members, it was stated that the surrogate objectives specified in Regulatory Guide 1.174 are guidelines rather than firm criteria, and the risk insights are utilized as part of a broader integrated evaluation that includes other considerations, such as defense-in-depth, and so on. RSK members raised questions regarding the initiatives being pursued under options 1, 2, and 3. From these questions, it is apparent that the RSK members are quite skeptical regarding risk-informing the regulation. Even for relatively simple applications, such as ranking of components based on safety significance, the RSK members, and especially the French member of RSK, seem to hold more faith in the current deterministic process.

This general skepticism regarding a move to risk -informed regulation was openly expressed during the brief presentation of Mr. Hahn (Vice Chairman, RSK) on this subject. Mr. Hahn first listed the current perceived impediments to progress in risk-informing the regulations. Among these, lack of clearly defined safety goals and surrogate criteria in Germany, incomplete PRAs, lack of PRA quality standards, large uncertainties often not well quantified, and vulnerability to manipulation of results. The ACRS delegation recognized these difficulties but stated that the ACRS did not see them as a major obstacle to further progress. A debate ensued on the merits of probabilistic versus deterministic, with the RSK members clearly in favor of the deterministic approach. A member of the ACRS delegation expressed his continuing disappointment with so much of the technical community viewing the two approaches as adversarial, when probabilistic considerations often quantified at the system level in failure mode and effect analyses (FMEA) and casualty analyses are central to engineering decisions in the design of current plants. For example, the whole protection system was designed to ensure that the consequences of accidents of higher expected frequency would be minimal, while some fuel damage was allowed for less frequent accidents. There was a general consensus on this point, and an agreement that the subject of risk-informing the regulations should be covered more extensively during future bilateral meetings. From separate conversations with individual RSK members, it is also apparent that the current short horizon of nuclear power in Germany contributes to lack of progress in this area.

License Renewal (LR)

The ACRS delegation provided a summary description of the LR process in the US. It was made clear that the focus of LR is on managing the aging of passive, long-lived components during the LR period. There was a consensus that the aging of cables is a major concern not yet resolved.

Germany does not impose a time limit on the operating life of nuclear facilities, but requires a plant review every ten years of operation. This review is intended to demonstrate that the plant is safe by current standards and that nothing has transpired over the past ten years that

would raise questions regarding the ability of the plant to operate safely for the next ten years. This review does not appear to be focused on the managing of the aging of components, although aging management is an expectation of the German program and it would be raised as an issue in the 10-year review wherever aging would raise specific concerns. The 10-year review is to provide an integrated assessment of plant status based on the following reviews:

- A general analysis of the safety status of the plant
- An evaluation of whether the plant PRA conclusions have changed
- An analysis of physical protection
- An assessment of the conditions of the safety components
- A review of plant specific operating experience
- A review of the plant accident analysis
- A review of severe accident capabilities

From discussions with individual RSK members, the low focus on component aging seems to have been influenced by the short-term horizon of the German nuclear program.

Pebble Bed Modular Reactor

We had a very good presentation on the design of the Pebble Bed Modular Reactor (PBMR), which is currently being considered for construction in South Africa. Dr. Kugeler, RSK, gave an overview of design and made some comments to the effect that the main difficulty in the South African design might be the civil engineering structures and the general layout of the system. The ACRS members were interested in issues associated with the reliability of the fabrication of the small fuel spheres and that there would need to be an excellent quality control program to ensure reliable fuel production. Dr. Uhrig also made a brief presentation on the PBMR covering subjects not covered by Dr. Krugeler, including Eskom's approach to safety and licensing, and the economic aspects of the PBMR approach.

Reactor Safety Research in Germany

There were two presentations on reactor safety research in Germany, one which covered the distribution of responsibilities and where the work was being carried out and a second on GRS codes for analysis of design bases and severe accidents. The six top priority areas for reactor research are:

1. Determination of service limits for materials and components, modeling of material behavior.
2. Experiments and analyses on thermal hydraulics during transients and loss of coolant accidents, on reactor physics and fuel rod behavior, on integrity of the reactor pressure vessel, and on core degradation.
3. Experiments and analyses regarding molten core stabilization in the containment, steam explosion, hydrogen distribution and combustion, as well as countermeasures and fission product behavior.
4. Further development of methods for probabilistic safety analysis, instrumentation and control, the assessment of human factors, and further development of modern methods for plant diagnostics.

5. Pursuance of innovative concepts (e.g., high temperature reactors), minimization and avoidance of plutonium, transmutation in subcritical reactors, observing safety aspects of foreign developments.
6. Know-how transfer regarding safety assessments of Eastern reactors.

In addition to the priority areas for reactor safety research, members of the RSK also discussed a need in Germany to maintain a "Competency Pool" of necessary technically qualified individuals and the fact that some of their sponsored research is designed to maintain that pool.

Conclusion: This was a good technical exchange; however, we attempted to cover too many items in the time available, and in the future we need to better plan the agenda. The Chairman of the RSK and other members strongly support continuing the technical exchanges, particularly ongoing research and generic issues. The ACRS representatives see value in these technical exchanges, but certainly the discussions need to be focused on areas of mutual interest, and the frequency of these meetings needs to be discussed.

Attachments: As stated

cc: ACRS members
J. Lyons
R. Savio
ACRS staff

References

Meeting Viewgraphs and Handouts

1. Viewgraphs, "Presentation of Developments in the Area of Safety I&C," "New Safety Concepts, Specifications," "Experience from Backfitting Projects," "Control Room Design for Tianwan," and "Situation in the USA," various presenters, Siemens, Erlangen, Germany, received November 7, 2000.

2. Nuclear Safety Standards Commission (KTA) 3501, "Reactor Protection System and Monitoring Equipment of the Safety System," dated June 1985.

3. Agenda for ACRS Visit to ISTec/GRS, "Safety Assessment of Digital I&C," received November 8, 2000.

4. Brochure on the Institut für Sicherheitstechnologie GmbH (ISTec) on Equipment Monitoring (e.g., Vibration Diagnosis, etc.) Waste Management, Instrumentation and Control, and other Programmatic Information, received November 8, 2000.

5. Handout, "Discussion ISTec/ACRS about safety assessment of digital I&C," D. Wach, ISTec, received November 8, 2000.

6. Handout, "Independent Generic and Plant-Specific System Qualification - A Methodology For Cost-Effective Assessment of Software-Based I&C Systems Important to Safety," D. Wach, ISTec, November 8, 2000, Reprint from IAEA Specialists' Meeting, Buenos Aires, Argentina, 1-3, June 1999.

7. Handout, "Working Program of the Subcommittee Electric Installations," received November 8, 2000.

8. Viewgraphs from Presentation on "Qualification of Software-Based I&C for Nuclear Power Plants" by Arndt Linder, ISTec, received November 8, 2000.

9. Handout, "Methodology and Tools for Independent Verification and Validation of Computerized I&C System Important to Safety," A. Lindner and H. Miedl, ISTec, November 8, 2000, Reprinted from IAEA, Group Specialists' Meeting, October 27-29, 1997, Budapest Hungary.

10. Viewgraphs from Presentation on "Future developments on digital I&C" by Dr. Graf, dated October 2000, received on November 9, 2000.

11. Confidential, (Foreign Proprietary) Draft Version of Section 7, "Electrical Systems of the Safety System and of the other Safety-related Systems," [replacement guidelines to reflect state of the art I&C Systems] from 305th RSK-meeting on November 20, 1996.

12. Viewgraphs, "Thermal-Hydraulic Phenomena Associated with Deboration in Transients," H.G. Sonnenburg, GRS, November 8, 2000.

13. Viewgraphs, "Note on the reactivity balance of boron dilution accidents," S. Langenbuch, GRS, received November 8, 2000.

References (continued)

14. Viewgraphs, "Potential of Boron Dilution after Failure of Residual Heat Removal System in Mid-Loop-Operation," W. Pointner, GRS, received November 8, 2000.

15. Viewgraphs, "Thermal-Hydraulic Aspects, Reactor Safety Research, and Objectives of the ThAI Facility," November 8, 2000.

16. Viewgraphs, "Spent Fuel Issues," GRS, received November 9, 2000.

17. Viewgraphs, "Analysis of Deboration Events," GRS, November 8, 2000.

18. Viewgraphs, "Parts of the Periodic Safety Review," RSK/ACRS Meeting, received November 8, 2000.

19. "Safety Codes and Guides-Translations," Bundesamt für Strahlenschutz Salzgitter, including Guides for the Periodic Safety Review of Nuclear Power Plants, printed December 1996.

20. Viewgraphs, "Reactor Safety Research in Germany," sponsored by the Ministry for Economics (BMWi), received November 9, 2000.

21. Report of the Working Group (Evaluation Commission) convened by the Federal Ministry of Economics and Technology (BMWi) on "Nuclear Reactor Safety and Repository Researching in Germany," January 21, 2000.

22. Brochure on the Thermal-hydraulics, Aerosol, Iodine (ThAI) Experimental Program at Battelle Ingenieurtechnik in Dusseldorfer, Germany, received November 9, 2000.

23. Brochure on research programs at the Karlsruhe Research Center Technik und Umwelt, received November 9, 2000.

24. "Euratom Framework Program research in reactor safety, main achievements of FP-4 ('94-'98), some preliminary results of FP-5 ('98-02)," G. Van Goethem, J. Martin Bermeijo, A. Zurita and P. Lemaitre, received November 7, 2000.

25. Viewgraphs, "Selected Safety Research Topics-Possibilities for Co-operation," V. Teschendorff, GRS, received November 9, 2000.

26. Viewgraphs, "Aspects of the South African PBMR-Project," K. Kugeler, Forschungszentrum Jülich, received November 9, 2000.

Reactor Safety Commission (RSK)

K.-D. Bandholz, R.L. Donderer, V. Engel, L. Hahn (Chairman), **W. Hartel, W. Hawikhorst, E. Kersting** (1. Vice-Chairman), **K. Kugeler, M. Reimann, M. Sailer, J. Scherrer, U. Schneider, M. Speidel, W. Thomas, R. Wieland** (2. Vice-Chairman)

Installations- and Systems-Engineering Committee (AST)

Hartel, Bandholz, Gerding, Kersting, Langetepe, Reimann, Scherrer, Schneider, Watzinger, Zinn

Pressure-retaining Components and Materials Committee (DKW)

Speidel, Engel, Andrzejczak, Bartonicek, Berger, Erhard, Erve, Jungclaus, Linder, Magdowski, Otremba, Schneider

Electric Installations Committee (EE)

Bandholz, Steckenborn, Glück, Graf, Hampel, Hartmann, Lindner, Zawilak

Reactor Operation Committee (RB)

Wieland, Hahn, Donderer, Grauf, Hartel, Hoffmann, Kotthoff, Schempp, Wilpert

Fundamental Issues Committee (GF)

Reimann, Kersting, Donderer, Hahn, Kugeler, Mertins, Schneider, Sgarz, Straub

Fuel Cycle and Disposal Committee (VE)

Sailer, Thomas, Appel, Bröskamp, Drotleff, Hawickhorst, Kugeler, Neumann, Odoj, Rittscher, Storck, Thein, Zech

Attachment 1

Reactor Safety Research in Germany

Contents

1. Distribution of responsibilities
2. Research institutions
3. Research fields of top priority identified by an Evaluation Commission
4. Competence pool “Kompetenzverbund”
5. Fields of current research
6. International cooperation

Distribution of responsibilities for reactor safety and repository research

Federal Ministry for Education and Research (BMBF):
institutional funding for the German Research Centres

Federal Ministry for Economics and Technology (BMWi): basic research work in relation to and with applications in reactor safety and repositories

Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU): investigations and studies in support of its responsibilities for regulation

Main contributors to reactor safety research

GRS

Research Centres Karlsruhe, Jülich, Rossendorf

Universities in Aachen, Bochum, Dresden, Karlsruhe, Stuttgart, Munich, Berlin, etc.

Research Institutions like

MPA (material science), IzfP (nondestructive testing)

Battelle Ingenieur Gesellschaft

(Industry)

Evaluation of Nuclear Research in Germany

To review the

Nuclear Reactor Safety and Repository Research in Germany an

Evaluation Commission was convened by BMWi:

Participants:

BMWi (Chair), BMU, BMBF,

Leading managers of research institutions: BGR, FZJ, FZK, FZR and GRS

Heads of the project sponsors of BMWi

6 Top Priority Areas

1. Determination of service limits for materials and components, modelling of material behaviour
2. Experiments and analyses on thermal hydraulics during transients and loss of coolant accidents, on reactor physics and fuel rod behaviour, on integrity of the reactor pressure vessel, on core degradation
3. Experiments and analyses regarding molten core stabilisation in the containment, steam explosion, hydrogen distribution and combustion, as well as counter measures and fission product behaviour.

6 Top Priority Areas

4. Further development of methods for PSA, for instrumentation and control, for the assessment of human factors, further development of modern methods for plant diagnostics,
5. Pursuance of innovative concepts (e.g., HTR), minimisation and avoidance of plutonium, transmutation in subcritical reactors, observing safety aspects of foreign developments
6. Know-how transfer regarding safety assessments of Eastern reactors

Competence Pool

Intensified programme co-ordination to maintain the necessary technically qualified training possibilities and long term safety competence.

Activities at the universities should be funded

Under way:

Programme co-ordination

Current Reactor Safety Research

Project funded reactor safety research of the BMWi:

1. Component Integrity
2. Transient and Accident Analysis
3. Man/Machine Interaction
4. Probabilistic Safety Assessment

Component Integrity

Material behaviour

Fracture mechanics and failure modes

- Corrosion assisted crack formation
- Damage processes on micro and nano meter level

Non-destructive fault detection

Online diagnosis

Ultrasonic methods for austenitic steels

Transient and accident analysis

Thermal hydraulics (ATHLET)

Coupling between neutron kinetics and thermal hydraulics

Fuel rod behaviour

Core degradation (ATHLET-CD)

Debris-wall interaction

In-vessel retention

Thermal hydraulics in the containment, COCOSYS

- Fission product behaviour (ISP KAEVER)
- Iodine (ThAI)

Man/Machine Interaction

Human factors

Event analyses, Implicit norms

Information to the operator

Reliability of computer based information systems

Probabilistic Safety Assessment

Method development

- German risk study A, B (DWR)
- BWR, incl. low power and shutdown status
- Level 2 treatment of severe accidents
  fire, earthquake, personnel actions

COOPRA

Research Centre Karlsruhe (FZK)

Final Disposal, including Repository Safety, Transmutation, Immobilisation of Waste

Innovative concepts, basic phenomena

Reactor safety (severe accidents)

a) Hydrogen combustion

b) Steam explosions in RPV and mechanical loading of RPV

c) Melt behaviour in containment

d) Radiological consequences

FZK, Reactor Safety

a) see separate report

b) In vessel steam explosion

QUEOS – steel spheres in water

PREMIX – thermite melt in water

ECO – tiggered, energy transfer

BERDA – RPV loading

FZK, Reactor Safety

c) Melt outside the RPV

KAJET/KAPOOL - melt behaviour in reactor cavity

KATS - melt spreading

COMET - long term passive melt cooling

d) Radiological consequences

COSYMA, RODOS decision system

Research Centre Jülich

Institute for Safety Research and Reactor Technology (ISR)

Accidental Risks of Large Technical Systems

Safety Research for Nuclear Plants

GRS

Thermalhydraulics	ATHLET
and Severe accidents	ATHLET – CD
Containment	COCOSYS
Reactor physics Fuel behaviour Debris/Wall RPV	
ATLAS 2000	Simulator
PSA Methods	DWR, SWR, Level 2, non full power uncertainties
NPP of Soviet Design	

Research Centre Rossendorf (FZR)

Reactor dynamics

- DYN3D

Thermal hydraulics

- ROCOM, NOKO

Material behaviour

- ageing by radiation
- High thermal and high mechanical loading

Reactors of Soviet Design

International Co-operation

Technical Exchange and Co-operative Arrangement between BMFT (now BMWi) and US NRC in the Field of Reactor Safety Research and Development (1995)

Information exchange and participation

CSARP	Severe Accidents
COOPRA	PSA
ACEX/MACE	EPRI

International Co-operation

OECD – projects:

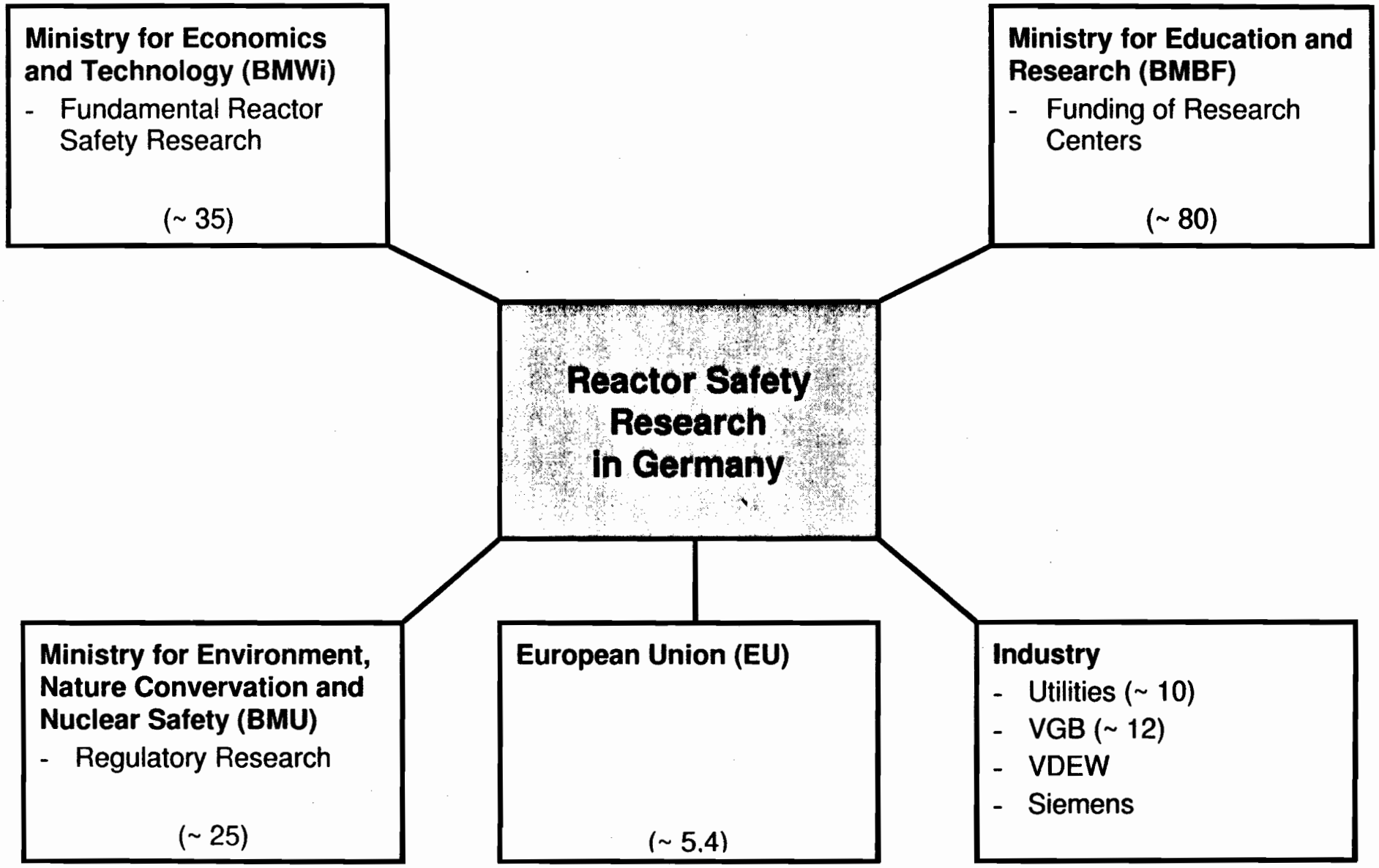
LOFT, TMI – VIP	
RASPLAV, MASCA	
LHF	in vessel retention
HALDEN	Man/machine, fuel

In preparation:

CABRI	high burn up, RIA
SETH (PKL, PANDA)	Thermalhydraulics
OECD – MACE	melt cooling

NPP of Soviet design, OECD Code validation matrix

Overview of Reactor Safety Research in Germany



Contribution in Million DM/Year

Areas in Which the BMU is Performing Regulatory Nuclear Safety Research

1. Reactor Vessel and Piping Integrity
2. Aging of Reactor Components
3. In-Service Inspections
4. Thermal Hydraulic Code Applications
5. Generic Safety Issues
6. Plant Performance and Application of Plant Analyses
7. Nuclear Fuel Analysis / High Burnup
8. Safety of I & C
9. Core Melt and Reactor Coolant System Failure
10. Severe Accident Analysis
11. Reactor Containment Safety
12. Containment Structural Integrity
13. Seismic Safety
14. Fire Safety
15. Probabilistic Risk Assessment / Periodic Safety Review
16. Radiation Safety and Consequence Analysis
17. Emergency Preparedness
18. Transport Safety of Radioactive Material
19. Security of Spent Fuel and High Level Waste Shipment
20. Waste Management
21. Safeguards (Physical Protection)
22. Decommissioning
23. Knowledge Management
24. Advancement of Safety Standards and Safety Criteria

Fire Safety

Human factor influence in fire events

- Analysis of NPP fires reported to the authorities in the past
  - Human influence in the fire occurrence
  - Human factor concerning fire detection (alarm, reaction of the personnel, actuation periods, etc.)
  - Human factor concerning fire extinguishing (actuation, false actuation, potential delay in extinguishing activities, etc.)
- Development of criteria for assessing the human influence in NPP fire events
- Statistical treatment of evaluated data
- Development of an international approach for guidelines to deal with the human factor in NPP fire events

Fire Safety

- Fire modeling incl. Applicability of fire simulation codes within deterministic fire hazard analysis (FHA) and fire PSA (FPRA)

  - Validation / verification of existing tools for NPP specific problems
    => extension of the „International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications“

  - Development of a harmonized data base for use in fire simulation models and codes

  - Development of a common guideline how and to what extent to apply calculational methods within FHA and FPRA

Fire Safety

Type of co-operation

- Partners principally working on their own
- Regular common meetings for exchange and discussion of results (e.g. every 6 months)
- Common work on guidance documents (e.g. for periods of 1 - 2 weeks)
- Financial funding:
  - For each organization on a national basis (e.g. GRS by BMU

Fire Safety

Co-operation partners

- US partners:
  - USNRC
  - Sandia National Laboratories (SNL)
  - National Institute of Standards and Technology (NIST)
  - Further partners?
- German partners:
  - GRS
  - Braunschweig and Vienna Universities of Technology
  - Further partners?

(15)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
478th	17	17-2

Title

FUTURE ACRS ACTIVITIES

Authors

John T. Larkins

List of Documents Attached

Future ACRS Activities - 478th ACRS Meeting, December 6-9, 2000

17

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

From Staff Person

John T. Larkins

ANTICIPATED WORKLOAD
December 6-9, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	Kress	Markley	Modifications to the Reactor Safety Goal Policy Statement	Report	–	PO 12/6 (A.M) P&P 12/5
Bonaca	Wallis	Boehnert	Issues Related to Core Power Uprate Reviews	Report	–	P&P 12/5 (P.M)
Kress	–	Boehnert/Singh	Control Room Habitability	Report	SAM 11/15	THP 11/13-14 M&M 11/16
Powers	–	Duraiswamy/Shoop	DPO on Steam Generator Tube Integrity	Report	P&P 12/5 (P.M)	PO 12/6 (A.M)
	–	El-Zeftawy	Research Report to the Commission [possible finalization of letter @ retreat?] (draft)	Report		
		Singh	**Meeting with Commissioner Diaz**	--		
Shack	–	Markley	Proposed Final Regulatory Guide DG-1053, "Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"	Report	M&M 11/16	PO 12/6 (A.M)

ANTICIPATED WORKLOAD
December 6-9, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Sieber	Apostolakis	Weston	Draft Safety Evaluation for South Texas Project Exemptions from special treatment requirements	Report	PO 12/6 (A.M)	SAM 11/15
Uhrig		Singh	ABB/CE and Siemens Digital I&C Applications (Subcommittee Report)	--	--	PO 12/6 (A.M)
Wallis		Boehnert	Report on Nov. 13-14 T/H Phenomena Subcommittee Meeting - Review of GE TRACG Code	--	THP 11/13-14	SAM 11/15 PO 12/6 (A.M)
		Boehnert	Response to Commission request for: a detailed discussion on how the perceived weaknesses with industry-developed thermal-hydraulic codes may adversely affect the NRC's regulatory role, and more specific recommendations on how those weaknesses should be addressed	Report (SRM due date 12/29)		

ANTICIPATED WORKLOAD
FEBRUARY 1-3, 2001

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Markley Larkins	ANS Standard on External-Events PRA Meeting with the NRC Chairman Meserve	Report	P&P 1/31 (P.M)	Joint M&M/THP 1/18 (A.M) ACRS/ACNW Joint Sub.1/19 Ret. 1/22-24/01
Kress		Weston	Regulatory Effectiveness of the ATWS Rule	Report	--	THP 1/16-17 Joint M&M/THP 1/18 (A.M.) ACRS/ACNW Joint Sub. 1/19 Ret. 1/22-24/01
Leitch		Singh	Reprioritization of GSI-152,"Design Basis for Valves that might be subjected to significant Blowdown Loads"	Report	--	Ret. 1/22-24/01
Powers		El-Zeftawy	Research Report to the Commission	FINAL Report	--	Joint M&M/THP 1/18 (A.M.) Ret. 1/22-24/01
Seale		Singh	Management Directive 6.4 and related Handbook Associated with the Revised Generic Safety Issue Process	Report	--	THP 1/16-17 (A.M) Joint M&M/THP 1/18 (A.M) Ret. 1/22-24/01
Shack	--	Boehnert	Treatment of Uncertainties in the Elements of the PTS Technical Basis Reevaluation Project	Report	Joint M&M/THP 1/18 (A.M)	Ret. 1/22-24/01

ANTICIPATED WORKLOAD
FEBRUARY 1-3, 2001

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Sieber		Weston	MOX Fuel Fabrication Facility- Overview	--		Ret. 1/22-24/01
Uhrig		Singh	Siemens and ABB/CE Digital I&C Applications	Report	--	Ret. 1/22-24/01
Wallis		Boehnert Boehnert	Siemens S-RELAP5 Appendix K Small-Break LOCA Code[1] RETRAN-3D Transient Analysis Code[1]	Report Report	THP 1/16-17	Joint M&M/THP 1/18 (A.M) Ret. 1/22-24/01

[1] The Planning and Procedures Subcommittee recommends that review of either S-RELAP5 or RETRAN-3D be deferred to the March 2001 ACRS meeting and that Dr. Wallis propose which one should be deferred.

ANTICIPATED WORKLOAD
March 1-3, 2001

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Bonaca	--		ANO, Unit 1 License Renewal Application	Interim Report	PLR 2/22	--
Wallis	--	Boehnert	Waterhammer Issues	Report	--	--

II. ITEMS REQUIRING COMMITTEE ACTION

1. Proposed Final License Renewal Guidance Documents (Open) (MVB/NFD) ESTIMATED TIME: 1½ hours

Purpose: Determine a Course of Action

Review requested by the NRC staff [S. Hoffman, NRR]. The Committee issued a report to the Commission on the draft license renewal guidance documents on November 15, 2000. Subsequent to the public comment period, the staff plans to provide the ACRS with copies of the proposed final drafts of the Standard Review Plan for license renewal, the Generic Aging Lessons Learned (GALL) report, Regulatory Guide-1104, and NEI 95-10 by March 1, 2001.

Dr. Bonaca has agreed to propose a course of action after receiving the documents. The staff plans to provide the ACRS with the proposed final versions of these documents in early March 2001.

2. Degraded Reactor Coolant System Pressure Boundary at V.C. Summer Nuclear Station (Open) (JDS/WJS/MTM/MWW) ESTIMATED TIME: 1½ hours

Purpose: Determine a Course of Action

Review requested by the ACRS. [K. Cotton, NRR] On October 7, 2000, the licensee identified an accumulation of boric acid near the "A" loop of the reactor vessel. Upon closer examination, the licensee identified an axial flaw less than three inches long on a weld between the reactor vessel nozzle and the "A" hot leg pipe. The weld location is about three feet from the vessel in a section of "spool-piece" piping. The flaw is about 17 inches from the top of the pipe. The "A" hot leg pipe has a nominal inside diameter of 29 inches and is approximately 2.5 inches thick.

The licensee issued its preliminary licensee event report (LER) on October 12, 2000. The licensee plans to perform a detailed equipment failure analysis, including a metallurgical failure analysis utilizing hot cell laboratory examinations, and root cause evaluation. In particular, the licensee plans to perform detailed fracture mechanics evaluation to assess critical flaw size and leak rate, i.e. leak-before-break calculations. The licensee plans to submit an LER supplement after the failure analysis is completed.

The staff issued requests for additional information (RAIs) on October 23 and November 3, 2000. The licensee submitted a ASME Section XI Code relief request for ultrasonic examination of the proposed piping repair on November 6, 2000. The licensee and staff held a public meeting to discuss this matter on November 21, 2000. The preliminary LER, RAIs, relief request, and meeting handouts related to this event were forwarded to the Committee on November 22, 2000.

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

3. Emergency Diesel Generator Failure at Seabrook (Open) (JDS/MWW) ESTIMATED TIME: 1½ hours

Purpose: Determine a Course of Action

Review requested by the ACRS. [R. Pulsifer, NRR] During surveillance testing of the "B" emergency diesel generator (EDG) at Seabrook on 10/29/00, a high differential pressure developed across the lube oil strainer. The strainer was inspected and debris removed. The strainer was re-installed and on 10/30/00, the test was rerun. After 3.5 hours, an emergency shutdown was performed because of high crankcase pressure and high vibration alarms. Operators in the EDG room observed a flash in the turbocharger area and heavy smoke in the room. The event evaluation and root cause teams determined that the #7 piston was severely scored and the piston liner was damaged and showed abnormal wear. The "A" EDG was subsequently inspected and scoring was identified on 2 different pistons. Region I dispatched a special inspection team to the site. Information regarding this event was provided to the Committee on November 17, 2000.

The Planning and Procedures Subcommittee recommends that Mr. Sieber propose a course of action.

4. Reprioritization of GSI-152, Design Basis for Valves that Might Be Subjected to Significant Blowdown Loads (Open) (GML/AS) ESTIMATED TIME: 1 hour

Purpose: Determine a Course of Action

NRC staff review request. [Owen Gormley, RES]. This issue was identified by the Office of Nuclear Regulatory Research (RES) following ACRS concerns raised during the 355th meeting regarding the resolution of GSI-87, "HPCI Steam Line Break Without Isolation." GSI-87 addressed the design bases for those MOVs that isolate the HPCI, RCIC, and RWCU systems in BWRs. These design bases required that the MOVs close against loads imposed by a double-ended pipe break at design basis flow conditions.

In resolving Issue 87, the staff issued Generic Letter No. 89-10 which required licensees to identify safety-related valves that might not perform adequately under design basis conditions. However, the ACRS believed that the design basis for the HPCI steam line valves and other valves in some plants might not specify this type of heavy duty. Thus, it was possible that heavy duty loads might not be considered for these valves by licensees in response to Generic Letter No. 89-10. The ACRS recommended in its letter of November 20, 1989, that the staff amend the generic letter to require licensees to examine their design bases to determine if safety-related valves, including but not limited to MOVs, were capable of operating against blowdown loads that might not have been considered (by licensees) in their original designs. However, the staff

chose to identify a new generic issue instead, GSI-152, because, unlike GSI 87, the question was the adequacy of the design bases rather than the ability of the valves to meet the requirements set forth in the design bases. This issue was previously assigned with a LOW priority ranking. The staff has recently reprioritized this issue and assigned a HIGH priority ranking.

The staff plans to provide the package related to reprioritization of GSI-152 to ACRS in early January 2001 and brief the Committee in February 2001.

The Planning and Procedures Subcommittee recommends that this matter be reviewed by the ACRS either at the February or the March 2001 ACRS meeting and that the ACRS staff find out whether deferring this item to the March meeting will have any impact on the staff's schedule.

Based on conversation with the staff, we understand that if ACRS review is deferred to March, the staff will not be able to meet the schedule commitment made to Congress.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 1, 2000

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards
Advisory Committee on Nuclear Waste

FROM: John W. Craig
Assistant for Operations
Office of the Executive Director for Operations

SUBJECT: PROPOSED AGENDA ITEMS FOR THE ACRS AND THE ACNW MEETINGS

Attached is a list of proposed agenda items for the ACRS (February 2001 - April 2001) and the ACNW (January 2001 - March 2001). This list was compiled based upon information received from (1) NRR, NMSS, RES, and IRO in response to the EDO request for the monthly update of proposed agenda items, and (2) the ACRS/ACNW staffs at a meeting held on November 28, 2000 with the OEDO, NRR, NMSS, and RES ACRS/ACNW coordinators [OEDO, I. Schoenfeld; NRR, M.G. Crutchley; NMSS, R.H. Turtil; and RES, J.A. Mitchell].

A copy of the Work Items Tracking System (WITS) list for January 2001 - April 2001 is also attached. This list includes a projection of office originated Commission papers that may be of interest to the ACRS/ACNW. Please provide timely feedback on your interest for briefings on particular items identified from the projected Commission papers that were not planned for formal review or information briefings but that are of interest to the Committees.

Attachments: As stated

ML003773326

PROPOSED AGENDA FOR
ACRS MEETINGS
(February 2001 - April 2001)

ACRS MEETING —— FEBRUARY 2001				
Item #	Title/Issue	Purpose	Priority	Documents
1	Status of MD 6.4, "Generic Issues Program" Contact: H. Vandermolen, DSARE/RES	Review and Comment	Medium	Draft SECY paper on MD 6.4 to be provided by January 9, 2001
2	Effectiveness of the ATWS Rule Contact: W. Raughley, DSARE/RES	Review and Comment	Medium	Draft ATWS report provided to ACRS in late September 2000.
3	Reprioritization of GSI-152, Valves Subject to Blowdown Loads Contact: O. Gormley, DET/RES	Review and Comment	High	Documents to be provided by 1/7/01.
4	Siemens S-RELAP5 Appendix K Small-Break LOCA Code Contact: R. Caruso/R.Landry, DSSA/NRR	Review and Comment	High	SER on Code to be provided late December
5	EPRI RETRAN-3D Code Contact: R. Caruso/R. Landry, DSSA/NRR	Review and Comment	Medium	SER on Code to be provided mid December
6	Overview of Licensing of Mixed Oxide Fuel Fabrication Facility Contact: A. Persinko, FCSS/NMSS	Information Briefing	Low	None.

ACRS MEETING — FEBRUARY 2001				
Item #	Title/Issue	Purpose	Priority	Documents
7	Treatment of Uncertainty in Elements of the PTS Reevaluation Project Contact: S. Malik, DET/RES	Review and Comment	High	Review documents to be provided 1/4/01.

ACRS MEETING — MARCH 2001				
Item #	Title/Issue	Purpose	Priority	Documents
1	Waterhammer Issues Contact: J. Tatum, DSSA/NRR	Review and Comment	High	EPRI interim report to be provided by 2/1/01.
2	ANO-1 License Renewal Contact: S. Hoffman, DRIP/NRR	Review and Comment	Medium	SER with open items to be provided by 1/10/01.

ACRS MEETING — APRIL 2001				
Item #	Title/Issue	Purpose	Priority	Documents
1	Hatch License Renewal and BWRVIP Documents Contact: S. Hoffman, DRIP/NRR	Review and Comment	Medium	SER with open items to be provided by 2/8/01.
2	SERs on BWR Vessel and Internal Project (BWRVIP) Contacts:T. Sullivan, G. Carpenter, B. Bateman DE/NRR; S. Hoffman, DRIP/NRR	Review and Comment	High	SE's have been continuously provided.

ACRS MEETING — APRIL 2001				
Item #	**Title/Issue**	**Purpose**	**Priority**	**Documents**
3	Risk-Based Performance Indicators Contact: S. Mays, DRAA/RES	Review and Comment	High	ACRS received the draft report on the results of Phase 1 development of risk-based indicators on October 16, 2000.
4	License Renewal Implementation Documents Contact: S. Lee, DRIP/NRR	Review and Comment	High	Final SRP, GALL Report, RG, and NEI 95-10 to be provided by 3/16/01.
5	Proposed Update to 10CFR Part 52 Contact: J. Wilson, DRIP/NRR	Review and Comment	Medium	Draft rule will be provided 30 days prior to meeting.

DRAFT 1:12/8/00
Larkins/Savio:car
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PROPOSED TOPICS

ACRS RETREAT AGENDA

DAY 1 (8 hours of discussion, 8:30am to 6:30pm)

1. Introduction, Scope, Objectives and Status of Actions on CY 1999 Self-Assessment Commitments (ACRS Chairman/JTL/RPS) (1/2 hour)

2. Summary of Stakeholder Comments and Discussion of CY 2000 Commitments (JTL/RPS) (1 hours)

3. Lead ACRS Member Analysis of Selected Work Products (3 hours)
(20 minutes for each lead ACRS member to give his analysis and 40 minutes for discussion)

   Issues will be selected during the December ACRS meeting and lead ACRS member/lead ACRS staff assignments made. The proposal is that the following topics be discussed:

   -Proposed modifications to the Safety Goal Policy (Kress)
   -Spent fuel pool fires (Powers)
   -Transient and accident analysis code review (Wallis)

   Lead ACRS member will use "metrics" in the attachment to guide his analysis.

4. Discussion of Selected Key ACRS Processes (P&P Subcommittee members) (2 hours)

   Analysis of issues selected during the December ACRS meeting to be provided by new P&P Subcommittee members. (See attached sheet-- **include process and criteria for selection of items for ACRS review**)

4. Discussion of Selected Key Technical Issues (7 hrs total, 1 ½ hrs on Day 1 and 5 ½ hrs on Day 2)

Identification and Quantification of Design Margins (Bonaca/Sieber)
Adequacy of PRA Models and Codes (Apostolakis)
Risk-informed Regulation (include discussion of GDCs and Appendices) (Powers/Shack)
AP 1000 Review Issues (discussion of what the ACRS expects to see in its review) (Kress/ Seale)
Potential new ACRS Initatives for CY 2001 (ACRS Chairman et al)

DAY 2 (7 ½ hours of discussion, 8:30am to 6:00pm)

5. Summary of Highlights of Discussion During Day 1 (½ hour)
(ACRS Chairman with comments from the other meeting attendees)

6. Discussion of Lessons Learned from Discussion of ACRS Work Products and ACRS processes (ACRS Chairman, lead ACRS members,and other meeting participants) (1 hour)

Summary of lessons-learned and any recommendations/ changes as a result of lessons learned

7. Discussion of Selected Key Technical Issues (7 hrs total, 1 ½ hour on Day 1 and 5 ½ hrs on Day 2)

Identification and Quantification of Design Margins (Bonaca/Sieber)
Adequacy of PRA Models and Codes (Apostolakis)
Risk-informed Regulation (include discussion of GDCs and Appendices) (Powers/Shack)
AP 1000 Review Issues (discussion of what the ACRS expects to see in its review) (Kress/ Seale)
Potential new ACRS Initatives for CY 2001 (ACRS Chairman et al)

DAY 3 (4 hours of discussion, 8:30am- 1:00pm)

8. Discussion of Retreat Highlights, Findings, and Commitments (ACRS Chairman with participation from other meeting attendees) (1/2 hour)

9. Discussion of the ACRS Research Report (3 hours)

10. Assessment of Technical Expertise Needed for CY 2001-2003 - ACRS members and consultants (ACRS Chairman/JTL - Input to be provided by COB on Day 2 by all ACRS members) (½ hour)

G:CY2000self
12/7/00

CY 2000 SELF ASSESSMENT

Select a small group of ACRS work products for critical analysis by the ACRS during its CY 2000 self assessment. The focus will be on lessons learned from analysis of the particular activity and not on success or failure. (Chose three of the following work products and construct specific questions to be used as tools for exploring the issues. Identify metrics (questions) that would be included in this evaluation. Separately from this, approve key ACRS Committee work processes that could be evaluated.

WORK PRODUCTS

An ACRS member will be assigned the lead for providing a critique at the retreat of each work product selected. The ACRS members indicated after each item will have the lead if this item is selected.

Low power shutdown operations risk (Powers)
*Proposed modifications to the Safety Goal Policy (Kress)
*Transient and accident analysis code review (Wallis)
License renewal (Bonaca)
Review of reactor operating experience (Leitch)
*Spent fuel pool fires (Kress)
Risk-Informing Part 50 (Apostolakis)
120-month ISI ASME code Updates (Shack)

METRICS

Did the work result in important ACRS advice (letter report, oral communication with the Commission, etc.) Or some other important work product?

What was the guiding ACRS regulatory philosophy and is it being consistently applied in ACRS advice? (To expand on the meaning, does the ACRS as a Committee have a consensus-based regulatory philosophy or do individual reports reflect the philosophy of the members having the most expertise on the subject and, whatever the answer, what is preferred? As examples, is the approach to the application of defense-in-depth consistent, are recommendations for regulatory changes risk-informed, and do recommendations for regulatory changes meet the requirements of the Backfit Rule).

Did the ACRS make a persuasive case, and why or why not? (Did the ACRS state its position clearly, explain its rationale clearly, and where needed make its arguments in a manner such that they could not be ignored?)

Was the ACRS review informed in that it considered the relevant information, stakeholder views, and focused on the relevant regulatory safety issues?

Did the ACRS influence the regulatory decision in a significant way, and why or why not?

Was the review efficient in its use of ACRS and stakeholder resources?

KEY ACRS PROCESSES

*ACRS interactions and communications with its stakeholders
*Selection of ACRS tasks and process for ACRS selection of self-initiated work
Meeting preparation (agenda planning, information dissemination, etc.)
Writing Committee reports
Joint ACRS/ACNW Subcommittee and joint ACRS/ACNW work
Communication with individual Commissioners
*Preparation for Commission briefings
Strategy and process for producing the annual research report
Annual visits to a region office and operating plant
Lessons learned from license renewal review process

*These three key ACRS processes are recommended as areas that the Committee should evaluate for lessons-learned. The first item should include a discussion of issues raised by NEI during our recent meeting and any actions the Committee might want to take in response to NEI comments.

ACRS MEETING HANDOUT

(16)

Meeting No.	Agenda Item	Handout No:
478th	17	17.2

Title MINUTES OF PLANNING & PROCEDURES SUBCOMMITTEE MEETING - DECEMBER 5, 2000

Authors JOHN T. LARKINS

List of Documents Attached

17

Instructions to Preparer
1. Punch holes
2. Paginate attachments
3. Place copy in file box

From Staff Person

JOHN T. LARKINS

G:PlanPro:ppmins.478
Dec. 8, 2000

SUMMARY/MINUTES OF THE PLANNING AND PROCEDURES SUBCOMMITTEE MEETING TUESDAY, DECEMBER 5, 2000

The ACRS Subcommittee on Planning and Procedures held a meeting on December 5, 2000, in Room 2B1, 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 1:00 p.m. and adjourned at 4:00 p.m.

ATTENDEES

D. A. Powers, Chairman
G. Apostolakis
M. Bonaca

ACRS STAFF

J. T. Larkins
J. Lyons
R. P. Savio
H. Larson
S. Duraiswamy
C. Harris
S. Meador
Mag. Weston

NRC STAFF

I. Schoenfeld

DISCUSSION

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the December ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the December ACRS meeting are included in a separate handout. 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 December 2000 ACRS meeting be as shown in the handout.

2) Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through March 2001 is included in a separate handout. 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.

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate. The Committee needs to consider the Subcommittee's recommendations on items listed in Section II of the Future Activities.

In addition, the Subcommittee recommends the following:

- Defer review of the Reprioritization of GSI-152 to the March ACRS meeting, if it does not impact the staff's schedule. [Based on conversation with the staff, we understand that deferring GSI-152 to the March meeting will impact the staff's schedule and the staff will not be able to meet the schedule commitment made to Congress.]

- In view of the heavy workload for the February 2001 meeting, review of either S-RELAP or RETRAN-3D should be deferred to the March ACRS meeting. Dr. Wallis should decide which item should be deferred.

- The Plant Operations Subcommittee Chairman, Jack Sieber, should prepare a review plan (similar to what was done for License Renewal) for the ACRS review of the MOX Fuel Fabrication Facility.

3) Election of ACRS Officers for CY 2001

The election of Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee will be held during the December 6-9, 2000 ACRS meeting. In accordance with Section 8.4 of the ACRS Bylaws, those members who do not wish to be considered for any of the above offices were requested to notify the ACRS Executive Director in writing at least two weeks prior to the December meeting.

Five members have informed the ACRS Executive Director that they do not wish to be considered for any of the offices.

4) ACRS Retreat for 2001

During the November meeting, the Committee decided to have a retreat on January 22-24, 2001 at Two White Flint North, Room T-2-B3. A proposed list of topics for discussion during the retreat is attached (pp. 1-2).

RECOMMENDATION

The Subcommittee recommends that the Committee review and comment on the proposed agenda for the retreat and assign a lead member for each topic. The Subcommittee suggests adding the topics noted below for discussion at the retreat:

- Risk-informing regulations, including GDC and appendices
- Adequacy of the PRA codes and models
- Quantifying margins
- AP 1000 Review Issues

With regard to the item on "Quantifying Margins," someone should develop information for use by the Committee in preparing a report during a future ACRS meeting.

5) CY 2000 Self-Assessment

During the November ACRS meeting, a list of ACRS work products that will be subject to a critical analysis by the ACRS members at the January 2001 ACRS retreat and proposed metrics for use in identifying the lessons learned were provided to the members for comment. A revised list of work products and metrics, which reflect incorporation of the comments received from some members, as well as proposed assignments for each work product are attached (pp. 3-4).

RECOMMENDATION

The Subcommittee recommends the following:

- The Committee should approve the list of work products, assignments, and metrics.

- The member responsible for a particular work product should lead the discussion at the retreat with regard to value added, lessons learned, effectiveness, timeliness, and quality associated with producing that work product.

- Prior to scheduling the proposed final license renewal guidance documents (SRP, GALL, and Regulatory Guide) for Committee review, it is important to evaluate whether the Committee could add value by reviewing this item.

  When evaluating the license renewal work product, the Committee should take into account the Committee's effectiveness, efficiency, and timeliness in reviewing the license renewal matters, including the proposed license renewal guidance documents.

- During the retreat, the Committee should discuss issues raised by NEI and the follow-up items and develop a course of action for dealing with such issues.

- Review of ACRS letter reports to assess value-added (i.e., whether or not some should have been Larkingsgrams)

6) ACRS Action Plan for CY 2001-2002

During the May 2000 ACRS meeting, the Committee approved the development of an ACRS Action Plan for CY 2001-2002. A draft Action Plan prepared by the ACRS staff was provided to the members for review and comment during the November ACRS meeting. A revised draft that incorporates, as appropriate, comments received from some members and the ACRS staff management was reviewed by the Planning and Procedures Subcommittee during its December 5, 2000 meting. Another revision, which incorporates the comments from the Subcommittee members is provided as a separate handout for review and approval by the full Committee at the December 2000 ACRS meeting. Subsequently, the ACRS Action Plan and Operating Plan will be forwarded to the Commission. We anticipate comments on the Action Plan from the Commissioners.

RECOMMENDATION

The Subcommittee recommends that the Committee approve the revised Action Plan.

7) Concern Expressed by a Private Citizen

Mr. Andy Bartlik, a private citizen who worked at the Fitzpatric Nuclear Power Plant, contacted Dr. Dana Powers during November 2000 to express his concern regarding physical separation of circuits for low pressure emergency core cooling systems. Mr. Bartlik contends that the NRC staff has not adequately reviewed or acted on potential deficiencies in electrical circuit separation at Fitzpatric. He provided copies of a General Electric Nuclear Energy Services Information Letter and a Deficiency Evaluation Report (DER) (pp. 5-10). The DER concludes that due to the large number of circuits involved and the complex redundancy arrangements established by accident analyses, it is unlikely that functionally redundant sets of equipment actually have the extent of physical independence as contemplated in NEDO-10139 and the June 24, 1975 submittal to the NRC.

The NRC staff is aware of Mr. Bartlik's concern and has requested additional information from the Fitzpatric licensee for use in determining the adequacy of electrical circuit separation. The staff expects to complete its review of Mr. Bartlik's concern by the end of February 2001.

RECOMMENDATION

The Subcommittee recommends that the Committee forward the information provided by Mr. Bartlik to the EDO for resolution and request an opportunity to review the staff's

evaluation when it becomes available. Also, the ACRS should inform Mr. Bartlik of the action taken by the Committee (pp. 11-12).

8) Burnup Credit

A representative of the RES staff has contacted Dr. Powers recently requesting a meeting with the ACRS/ACNW Joint Subcommittee to discuss a program, which includes analysis of issues related to burnup credit for transportation, being developed by RES in response to a request from NMSS. The issue of burnup credit has previously been reviewed by the ACNW as it relates to transportation shipping casks.

RECOMMENDATION

The Subcommittee recommends that Drs. Apostolakis and Kress, in coordination with the ACNW members of the Subcommittee, review this matter during a meeting of the Joint Subcommittee in March 2001 and have it referred to the appropriate Committee should a letter be warranted.

9) Bilateral Exchange

While in Germany for the Bilateral Exchange between the RSK and several ACRS members, the issue of the next quadripartite meeting came up. In a recent E-mail from the GRS (p. 13) there was an inquiry as to whether or not the ACRS would support the next quadripartite meeting in Germany and if we had some dates and possible issues for the agenda.

RECOMMENDATION

The Subcommittee recommends the following:

- The ACRS Executive Director, in coordination with the other countries involved, should plan for a Quadripartite meeting during October 2001. If that is not feasible, it should be planned for June 2002. During the next Quadripartite meeting, a discussion should be held with regard to expanding the membership by adding new countries (e.g., Switzerland, Spain, etc.).

- The members should read the trip report of the meeting with RSK on November 6-10 and discuss with the members who attended that meeting with respect to value added. During the January 22-24, 2001 retreat, the members should decide whether bilateral meetings should be continued in the future.

DRAFT 1:12/1/00
Larkins/Savio:car
G:retreat

PROPOSED TOPICS

ACRS RETREAT AGENDA

DAY 1 (8 hours of discussion)

1. Introduction, Scope, Objectives and Status of Actions on CY 1999 Self-Assessment Commitments (ACRS Chairman/JTL/RPS) (1 hour)

2. Summary of Stakeholder Comments and Discussion of CY 2000 Commitments (This is in response to George's request for CY 2000 set of stakeholder interviews)(JTL/RPS) (2 hours)

3. Discussion of the ACRS CY 2001 Action Plan and Priority Review Items (ACRS Chairman and lead ACRS members for priority review items. Lead members will provide a brief discussion of future Committee activities on each priority review item and what the Committee might hope to accomplish) (2 hours)

4. Lead ACRS Member Analysis of Selected Work Products (½ hour for each lead ACRS member to give his analysis and ½ hour for discussion) (2 hours)

   - Issues will be selected during the December ACRS meeting and lead ACRS member/lead ACRS staff assignments made.

   - Lead ACRS member will use "metrics" decided on during the December P&P Subcommittee mtg.

   - Time will be left for discussion of 2 work products on Day 1 (see attached sheet)

5. Discussions with 1 to 2 Commissioner(s) (1 hour)

Theme: "Commission Need for an Independent, but Responsive ACRS"

DAY 2 (8 hours of discussion)

6. Summary of Highlights of Discussion During Day 1 (½ hour)

   (ACRS Chairman with comments from the other meeting attendees)

7. Lead Member Analysis of Selected work Products (Continued from Day 1) (3 hours)

   - Discuss 3 more work products under the guidelines of Item 4. This will be a total of 5 work products which will probably be enough.

8. Discussion of Lessons Learned from Discussion of ACRS Work Products (ACRS Chairman) (1 hour)

   Summary of lessons-learned and any recommendations/ changes as a result of lessons learned

9. Discussion of Selected Key ACRS Processes (P&P Subcommittee members) (1 hour)

   - Analysis to be provided by new P&P Subcommittee members. (See attached sheet)

10. Potential New ACRS Initiatives for CY 2001 (ACRS Chairman) (1 hour)

11. Discussion with 1 - 2 additional Commissioners (1 hour)

    Theme - "Commission Need for An Independent, But Responsive ACRS"

DAY 3 (4 hours of discussion)

12. Summary of Retreat, Highlights, and Findings, and Recommendations (Lead will be ACRS Chairman with work being shared with ACRS Vice-Chairman and Member-at-Large) (3 hours)

13. Assessment of Technical Expertise Needed for CY 2001-2001 - ACRS members and consultants (ACRS Chairman/JTL) - Input to be provided by COB on Day 2 by all ACRS members (1 hour)

CY 2000 SELF ASSESSMENT

Select a small group of ACRS work products for critical analysis by the ACRS during its CY 2000 self assessment. The focus will be on lessons learned from analysis of the particular activity and not on success or failure. (Chose five of the following work products and construct specific questions to be used as tools for exploring the issues. Identify metrics (questions) that would be included in this evaluation. Separately from this, approve key ACRS Committee processes that could be self evaluated.

WORK PRODUCTS

An ACRS member will be assigned the lead for providing a critique at the retreat of each work product selected. The ACRS members indicated after each item will have the lead if this item is selected.

Low power shutdown operations risk (Powers)
Proposed modifications to the Safety Goal Policy (Kress)
*Transient and accident analysis code review (Wallis)
*License renewal (Bonaca)
Review of reactor operating experience (Leitch)
*Spent fuel pool fires (Kress)
*Risk-Informing Part 50 (Apostolakis)
120-month ISI ASME code Updates (Shack)

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*The ACRS staff recommends that the Planning and Procedures Subcommittee select these four work products for critical analysis at the retreat and time (and interest) permitting the lead member should be prepared to discuss the other areas.

METRICS

Did the work result in important ACRS advice (letter report, oral communication with the Commission, etc.) Or some other important work product?

What was the guiding ACRS regulatory philosophy and is it being consistently applied in ACRS advice? (To expand on the meaning, does the ACRS as a Committee have a consensus-based regulatory philosophy or do individual reports reflect the philosophy of the members having the most expertise on the subject and, whatever the answer, what is preferred? As examples, is the approach to the application of defense-in-depth consistent, are recommendations for regulatory changes risk-informed, and do recommendations for regulatory changes meet the requirements of the Backfit Rule).

Did the ACRS make a persuasive case, and why or why not? (Did the ACRS state its position clearly, explain its rationale clearly, and where needed make its arguments in a manner such that they could not be ignored?)

Was the ACRS review informed in that it considered the relevant information, stakeholder views, and focused on the relevant regulatory safety issues?

Did the ACRS influence the regulatory decision in a significant way, and why or why not?

Was the review efficient in its use of ACRS and stakeholder resources?

KEY ACRS PROCESSES

*ACRS interactions and communications with its stakeholders
*Selection of ACRS tasks and process for ACRS selection of self-initiated work
Meeting preparation (agenda planning, information dissemination, etc.)
Writing Committee reports
Joint ACRS/ACNW Subcommittee and joint ACRS/ACNW work
Communication with individual Commissioners
*Preparation for Commission briefings
Strategy and process for producing the annual research report
Annual visits to a region office and operating plant
Lessons learned from license renewal review process

*These three key ACRS processes are recommended as areas that the Committee should evaluate for lessons-learned. The first item should include a discussion of issues raised by NEI during our recent meeting and any actions the Committee might want to take in response to NEI comments.

Physical separation of circuits for low pressure Emergency Core Cooling Systems

SIL No. 630

July 17, 2000

The owner of a GE BWR/4 plant located in the United States recently identified a concern related to separation of electrical circuits for low pressure Emergency Core Cooling Systems (ECCS) within the same electrical division. GE Nuclear Energy performed an evaluation addressing the potential effects of an electrical failure on the single failure assumptions supporting plant loss-of-coolant accident (LOCA) analyses. No plant design deficiencies have been identified, and GE has determined that this concern could not create a substantial safety hazard or contribute to the exceeding of a technical specification safety limit. However, the results of the evaluation indicate that the plant electrical separation specifications for certain GE BWR/3 and /4 plants may not provide sufficient information to assure adequate electrical separation between certain safety-related components required to perform the ECCS function.

The purpose of this SIL is to discuss the GE evaluation and to recommend that owners of affected GE BWRs review and, if necessary, clarify documented plant separation requirements. This SIL applies to GE BWR/3 and /4 plants having Low Pressure Coolant Injection (LPCI) subsystems that were originally provided with loop selection logic, including those plants that subsequently eliminated loop selection logic (e.g., the "LPCI modification").

Discussion

The majority of GE BWR/3 and /4 plants were designed, prior to the issuance of 10CFR50.46 and 10CFR50 Appendix K, with low pressure ECCS consisting of a LPCI subsystem and a Core Spray (CS) system to provide diverse core cooling methods (flooding and spray cooling). The two 100% capacity CS subsystems and one LPCI subsystem (with four 33.3% capacity pumps) were separated into two electrical divisions, and redundant electrical circuits from the two divisions were provided for certain motor-operated valves (MOVs) in the LPCI subsystem. The LPCI subsystem was originally designed with loop selection logic to direct LPCI flow to the intact recirculation loop following a postulated Recirculation System pipe break. Any of the three subsystems was capable of providing the required core cooling function, and LOCA analyses demonstrated that the original design requirements were met even with complete loss of all equipment in one of the two electrical divisions.

Implementation of the 10CFR50.46 acceptance criteria with the Appendix K requirements for evaluation models and single failure assumptions resulted in very restrictive core operating limits. However, based on single failure analysis, credit could be taken for availability of more than one of the three low pressure ECCS subsystems (either two CS subsystems or one CS subsystem and at least partial functionality of the LPCI subsystem). In addition, several plants implemented the LPCI modification to increase post-LOCA core cooling capability to minimize the impact of these new requirements on core operating limits.

When performing plant LOCA analyses, GE requests that BWR owners identify the remaining ECCS subsystems available following postulated single failures consistent with the current plant licensing bases. The original plant separation requirements were adequate to assure that postulated single failures could only affect equipment in one electrical division. However, for some specific failure scenarios, these requirements may not be adequate to assure the

validity of assumptions supporting analyses to demonstrate compliance with the more conservative requirements of 10CFR50.46 and 10CFR50 Appendix K.

Two particular scenarios may be of concern if fault-limiting devices in the affected circuits are not adequate to prevent failure propagation. These scenarios are:

1. single failures in other systems that could credibly propagate within one division to both the LPCI (including recirculation pump discharge valve) and CS circuits within that division; or
2. single failures in either the LPCI (including recirculation pump discharge valve) or CS circuits in one division that could credibly propagate to the other subsystem within that division (i.e., LPCI to CS, or CS to LPCI).

The consequences of such postulated failures could invalidate the limiting single failure assumptions supporting the plant LOCA analysis. The original plant separation requirements would not have prevented these failure scenarios, but actual plant design implementation may be adequate to assure that such failure scenarios are not considered credible.

For known cases where additional intra-divisional separation requirements were established, reviews have identified that the plant electrical separation specifications were not revised to include these additional requirements. If these requirements are not properly documented and fully understood, the electrical cables for certain CS MOVs and LPCI MOVs (including the recirculation pump discharge valve) in the same division might be routed in the same wireways, and an electrical failure could potentially result in a reduction in ECCS capacity below the level assumed in the plant LOCA analysis. In addition, plant modifications, including the LPCI modification, might inadvertently violate these requirements.

Recommended action

GE Nuclear Energy recommends that owners of GE BWR/3 and /4 plants having LPCI subsystems that were originally provided with loop selection logic take the following actions:

1. Review plant electrical separation specifications, ECCS documentation, and associated licensing bases and commitments to determine if currently documented separation requirements are adequate to assure continued validity of the limiting single failure assumptions supporting the plant LOCA analysis.
2. If necessary, update plant electrical separation specifications.

To receive additional information on this subject or for assistance in implementing a recommendation, please contact your local GE Nuclear Energy Service Representative.

This SIL pertains only to GE BWRs. The conditions under which GE Nuclear Energy issues SILs are stated in SIL No. 001 Revision 6, the provisions of which are incorporated into this SIL by reference.

Product reference

E11— Residual Heat Removal (LPCI subsystem)
E21— Core Spray
R00—Plant Electrical

Issued by

Bernadette Onda Bohn, Program Manager
Service Information Communications
GE Nuclear Energy
175 Curtner Avenue
M/C 772
San Jose, CA 95125

DER-00-00064
Apparent Deficiencies with NEDO-10139

Objective:

The objective of this document is to describe an error in the methodology employed by NEDO-10139, which when considered in conjunction with the plant electrical separation, may potentially have resulted in functionally redundant cables, critical to ECCS operation, to being installed in a common wireway. Where applicable, this document will also discuss the post VY configuration of the facility.

Background:

The electrical separation criteria applicable to James A FitzPatrick Power Plant, requires that cables associated with Division A and B systems and equipment, be installed in a separate set of wireways, to minimize the potential for the loss of "redundant equipment", should a wireway be destroyed by a physical event. When this separation criteria was established, it was believed that a single Core Spray pump, augmented with either the Automatic Depressurization System (ADS) or the High Pressure Coolant Injection (HPCI), would be adequate for the full spectrum of Loss of Coolant Accidents (LOCAs).

During the licensing process, the above redundancy paradigm, was determined to be inadequate. This was in part due to changes in the ECCS performance requirements directed by the NRC in the Interim Acceptance Criteria (IAC). In addition, it had been determined that a single core pump may not be capable of achieving the design spray density on all fuel bundles.

In response to these issues, General Electric defined alternate redundancy paradigms, which credit either simultaneous operation of two LPCI pumps and one Core Spray pump, or the simultaneous operation of two (previously thought to redundant) Core Spray pumps.

The survival of at least one of the above described redundancy arrangements to an assumed loss of a single wireway is not necessarily assured by the established electrical separation criteria. This is because each of the above redundancy paradigms can rely on equipment of both established divisions simultaneously. This is readily apparent for the "Two Core Spray Pump" accident response strategy, as a large fraction of both Division A and B components would be required to function in concert, in order to ensure two core spray pumps would be available. Although less obvious, this is also true for accident response utilizing "Two LPCI pumps and One Core Spray pump".

Although the LPCI system has many independent Division A and B components, successful LPCI operation can simultaneously require proper function of both Division A and B components. For example, for plants which have not implemented the VY modification, for a recirculation suction line break on the A side of the plant, with the Division A LPCI response, the Division B LPCI injection valves must function. Recognizing this dual division dependence, General Electric assessed the capability of the ECCS to respond assuming a large range of failures, including the physical destruction of a wireway. This analysis is documented in NEDO-10139.

Although, the redundancy arrangements are more complex, and too difficult to describe in a brief background statement, a redundancy paradigm relying on two of the established divisions simultaneously is also relied upon for plant which have implemented the VY modification.

Problem Statements:

In NEDO-10139, General Electric represented that a failure, resulting in the loss of a wireway segment which, can totally defeat the LPCI function, would not in anyway affect either division of Core Spray. The review performed by General Electric appears to have not included all cables critical to ECCS operation and consequently has not actually demonstrated that failures in wireway segments which can totally defeat LPCI, is in fact independent of both divisions of Core Spray as claimed by NEDO-10139. The following lists specific reviews, which should have been performed, but appear to have not been:

<u>Wireway Failures Associated with the LPCI Injection Valve</u>

General Electric's assessment of the independence of Core Spray and LPCI did not include the wireways which contain the control cables from the LPCI control panel to the LPCI bus itself. The existing electrical separation criteria does not specify any special routing criteria for these cables, which would ensure their independence from cables necessary to support both divisions of Core Spray. Consequently, it is indeterminate if physical separation and independence is actual present as claimed.

In addition, the 600 volt power feeder to the injection valves themselves, do not have any special routing criteria. On a similar basis as described above, it is indeterminate if physical separation and independence is actually present as claimed.

The LPCI injection valve have auxiliary relay contacts in the control circuit of the valves. The failure of these control inputs could block proper function of the injection valves.

The physical routing of cables which transmit the signals from the instruments in the field to the logic panels has not been considered and been demonstrated to be independent of both divisions of Core Spray. For example, 10MOV-25A (one of the Division A LPCI injection valves), has a contacts associated with relays 10A-K63A and 10A-K66, in the closing circuit. Although these relays themselves are located in logic cabinet 9-32, failure of cables associated local devices can complete the closing coil circuit, causing the valve to cycle closed. This can occur even with an opening signal from Division B sources present. With both an opening and closing signal present, the valve would continually cycle between the open and closed position, potentially depleting the battery. Similar contact sets can be found in 10MOV-25B. Other RHR and CS valve control circuits should also be examined to determine if similar conditions are present.

Effect of Failure of Non-Automatic Valves on ECCS Response

The General Electric review in NEDO-10139 does not appear to include valves which are not required to operate to support proper ECCS function. The malfunction of some non-automatic valves can adversely affect ECCS function. For example, the spurious opening of the suppression pool cooling return flow path could divert LPCI injection flow to the torus. Although this flow path would be expected to be blocked by two valves in series, both these valves are of the same division, with no special routing criteria, and my be installed in the same wireway, and subject to common mode failure. As failure of theses valves can directly disable both divisions of LPCI, both divisions of Core Spray should have been shown to be independent of the control circuitry of these valves. A similar condition may apply to the drywell spray valves isolation valves, and to a lesser extent the wetwell spray isolation valves.

For plants which have implemented the VY modification, the spurious closure of the by-pass valve around the RHR heat exchanger could block LPCI flow from two pumps of opposite divisions. Based on the redundancy paradigm applicable to plants which have implemented the VY modification, wireway failures along the routing of cables which can cause the spurious closure of this valve should be demonstrated to be independent of both divisions of Core Spray and the RHR pumps injecting in the alternate loop. This independence is not assured based on the established electrical separation criteria for the facility. In addition, based on a review of a July 24, 1975 submittal to the NRC relative to the VY modification at JAF, it is not apparent that the necessary review, required to show the required level of physical separation, was preformed during implementation of the modification. A similar condition appears to be applicable to the RHR mini flow valves of plants which have implemented the VY modification.

DER-00-00064
Apparent Deficiencies with NEDO-10139

Failure to Establish Adequate Administrative Controls to Maintain Plant Design Basis

NEDO-10139 established a new redundancy paradigm, which essentially required different electrical separation criteria than in place at JAF. NEDO-10139 attempted to show that adequate electrical separation was present to support this new redundancy arrangement. However, NEDO-10139 fell short of actually calling for revision to the electrical separation specifications, to ensure that the new redundancy arrangement maintained the required level of electrical independence. Consequently, it appears that design control measure, in accordance with the requirements of 10CFR50 Appendix B, Section III, have not been established. Absent appropriate design control measures, it is possible that design changes could have placed the facility in non-compliance with it's design basis.

Based on discussions with knowledgeable engineer, in responsible charge at the facility, the above described redundancy paradigms are neither thoroughly understood nor known. An example of a modification which may have decreased the level of independence below the required degree is modification F1-91-305, titled "LPCI Alternate Power Supply Circuit Modification". This modification routed circuits, whose failure could potentially disable one of the LPCI independent power supply circuits, in the general raceway system, applying train assignment as the primary criterion. The modification makes no mention of any special separation requirements applicable to circuits associated with LPCI, which also need to be independent of both divisions of Core Spray as well as the LPCI pumps of the alternate loop.

Conclusion:

Due to the large number of circuits involved, and the complex redundancy arrangements established by the current accident analyses, it is unlikely that functionally redundant sets of equipment actually have the extent of physical independence as contemplated in the NEDO-10139 and the July 24, 1975 submittal to the NRC.

PREDECISIONAL DRAFT

Initial Draft: 11/29/2000
TSK/nfd
G: \DUDLEY\Allegation

, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PHYSICAL SEPARATION OF CIRCUITS FOR LOW PRESSURE EMERGENCY CORE COOLING SYSTEMS

The purpose of this memorandum is to forward information received from Mr. Andy Bartlick, a member of the public, concerning the adequacy of electrical circuit separation at the James A. Fitzpatric Power Plant. Mr. Bartlick contacted Dr. Dana Powers, ACRS Chairman, and e-mailed him the attached documents. Mr. Bartlick contents that the NRC staff has not adequately reviewed or acted on this issue. I understand that the staff is reviewing the issue of electrical circuit separation and expects to reach a resolution by the end of February 2001. The ACRS would like the opportunity to review the staff's resolution when it becomes available.

Attachments:
1. E-mail from Andy Batlik to Dana A. Powers, Chairman, ACRS, Subject: SIL-630 and Related Deficiency Report, dated November 20, 2000.
2. General Electric Nuclear Energy Services Information Letter SIL No. 630, "Physical Separation of Circuits for Low Pressure Emergency Core Cooling Systems," dated July 17, 2000.
3. Deficiency Evaluation Report DER-00-00064, "Apparent Deficiencies with NEDO-10139," undated.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
S. Collins, NRR
J. Calvo, NRR
O. Chopra, NRR

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G:\DUDLEY\ Bartlik
11/29/2000

November , 2000

Dear Mr. Bartlik:

In response to your concern related to the physical separation of circuits for low pressure Emergency Core Cooling Systems at James A. Fitzpatrick Power Plant, the Advisory Committee on Reactor Safeguards (ACRS) contacted the NRC staff. The ACRS also forwarded to the staff the General Electric Nuclear Energy Services Information Letter SIL No. 630, "Physical Separation of Circuits for Low Pressure Emergency Core Cooling Systems," and Deficiency Evaluation Report DER-00-00064, "Apparent Deficiencies with NEDO-10139," which you sent to me.

The NRC staff is aware of your concern and has requested the licensee to provide information demonstrating the adequacy of electrical circuit separation. The ACRS has request an opportunity to review the staff's evaluation of the licensee's information when it becomes available.

We will inform you of the results of our review.

Sincerely,

Dana A. Powers, Chairman
Advisory Committee on Reactor Safeguards

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From: "Renzo Candeli" <bmcan@grs.de>
To: <JTL@nrc.gov>
Date: Fri, Dec 1, 2000 8:12 AM
Subject: Quadripartite Meeting

Dear Dr. Larkins,

during our meeting in Munich on November 8 and 9, the Chairman of the RSK proposed to organize next year the next Quadripartite Meeting in Germany.

I would like to ask, if the ACRS still agree to this proposal in order to begin with the organization of the meeting. Furthermore I would like to know, if you already have some proposals concerning the dates to held the meeting and wished issues for the agenda.

In this concern, I shall be able to hear the response of the French commission (GPR) on December 14, 2000, during a GPR/RSK meeting in Cologne.

I thank you in advance.

Sincerely

Candeli