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

October 31, 2000

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

SUBJECT: SUMMARY REPORT - 476TH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, OCTOBER 5-7, 2000, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

Dear Chairman Meserve:

During its 476th meeting, October 5-7, 2000, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report and letter. In addition, the Committee authorized Dr. Larkins, Executive Director, ACRS, to transmit the memorandum noted below:

REPORT

- Union of Concerned Scientists Report, "Nuclear Plant Risk Studies: Failing the Grade" (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated October 11, 2000)

LETTER

- Pressurized Thermal Shock Technical Basis Reevaluation Project (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, ACRS, dated October 12, 2000)

MEMORANDUM

- Proposed Revision to 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage" (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated October 11, 2000)

HIGHLIGHTS OF KEY ISSUES CONSIDERED BY THE COMMITTEE

1. Discussion of Union of Concerned Scientists Report, "Nuclear Plant Risk Studies: Failing the Grade"

The Committee heard a presentation by and held a discussion with a representative of the Union of Concerned Scientists (UCS) concerning the August 2000 UCS report entitled, "Nuclear Plant Risk Studies: Failing the Grade," concerning the use of risk information in NRC decision making. The Committee discussed the issues raised in the report as well as feedback that UCS has received from interested parties. The UCS representative informed the Committee that its report has been criticized for using obsolete results from outdated Individual Plant Examinations (IPEs). UCS stated that they prefer to use the information provided in the updated licensee probabilistic risk assessments (PRAs) but noted that these PRAs are not available to the public. UCS also noted that the current IPE information was used in the development of the significance determination process of the revised reactor oversight process.

The Committee and UCS discussed several equipment studies (e.g., concerning high pressure core injection and reactor core isolation cooling systems, etc.) completed by the Office of Nuclear Regulatory Research (formerly the Office for Analysis and Evaluation of Operational Data). In particular, the Committee considered the UCS contention that these studies were flawed with system-level bias. The Committee also discussed the issues of assumptions and consequences in risk analysis and the UCS concern over differences in PRA results for "sister" plants.

Conclusion

The Committee provided a report to the Chairman on this matter dated October 11, 2000.

2. NEI 00-02, "Industry PRA Peer Review Process Guidelines"

The Committee heard presentations by and held discussions with representatives of the Nuclear Energy Institute (NEI) and ERIN Engineering concerning the proposed industry certification process described in the document NEI 00-02, "Industry PRA Peer Review Process Guidelines." NEI stated that NEI 00-02 was developed from the peer review process developed by the BWR Owners Group. NEI further stated that the certification process does not provide an overall grade for PRAs, but can be used as a complement to or in lieu of industrial standards for PRA quality.

The Committee considered the industry's proposed approach to use NEI 00-02 as a means of addressing the issue of PRA quality for risk-informed decision making (i.e., as a template for NRC review). The Committee also considered the recent meetings between the NRC and industry representatives on this matter. The Committee noted that the staff had expressed concern that subjective standards may lead to inconsistent grading of subtier elements and that the results vary depending on the makeup of the peer review team. NEI acknowledged that most PRAs need to be improved. At the conclusion of the meeting, NEI invited the ACRS and NRC staff to observe and/or participate in the peer review process in order to gauge the rigor applied to licensee risk-informed decisions using the peer review panel.

Conclusion

This briefing was for information only. No Committee report was required.

3. Staff Views on ASME Standard for PRA for Nuclear Power Plant Applications

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning their views on Revision 12 of the ASME Standard for PRA for Nuclear Power Plant Applications. The staff used SECY 00-0162 as guidance in formulating their comments on the draft. The issue of PRA quality is central to risk-informed regulations and one that the Commission has continually raised with the staff. As a result of reviewing the draft standard, the staff concluded that the current version of the standard (1) does not address PRA quality, (2) is difficult to use in determining where there are strengths and weaknesses in the PRA results, (3) will provide limited assistance to staff in performing focused review of PRA submittals, and (4) will provide minimal assistance in making more efficient use of NRC resources.

The staff discussed major concerns that led to the conclusions chapter by chapter. Among the concerns discussed were the definition of the categories within which the PRA applications would fit, the requirements for the risk assessment application process, the completeness and accuracy of the technical content of the document, and the changes necessary for the standard to be acceptable.

Conclusion

This briefing was for information only. No committee action was required.

4. Pressurized Thermal Shock Technical Basis Reevaluation Project

The Committee heard presentations by and held discussions with the NRC staff regarding the status of activities associated with the Pressurized Thermal Shock (PTS) Technical Basis Project. The staff described the FAVOR probabilistic fracture mechanics computer code and provided a more detailed description of the development of a fracture toughness distribution. The ACRS Members and the staff discussed reevaluating the reactor vessel failure acceptance value based on a source term that assumes air oxidation of fuel instead of steam-oxidation. They discussed the appropriateness of using a Weibull distribution for the fracture toughness data and a one dimensional finite analysis for crack distributions. They also discussed the effects of neutron fluence, the characterization of flaws, damage accumulation, and the calculation of thermal-hydraulic uncertainties.

Conclusion

The Committee provided a letter to the Executive Director for Operations (EDO) on this matter dated October 12, 2000.

5. Discussion of Industry Issues

The Committee met with Ralph Beedle, NEI Senior Vice President and Chief Nuclear Officer, Nuclear Generation Division and members of his staff to discuss NEI's current regulatory initiatives, emerging industry issues, NEI's recent reorganization, and other issues of mutual interest. These issues included: risk-informing 10 CFR Part 50; license renewal; decommissioning; and the revised reactor oversight process.

NEI representatives frequently meet with the ACRS to discuss particular regulatory issues. In contrast, these discussions focused on strategic planning, regulatory philosophy, and ACRS and NEI views on emerging issues.

Conclusion

The discussions were informative and productive. The ACRS Chairman plans to schedule this type of stakeholder exchange on a regular basis.

6. GSI-168, Equipment Qualification

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning the proposed resolution and the status of Generic Safety Issue (GSI) 168, "Environmental Qualification (EQ) of Low-Voltage Instrumentation and Control (I&C Cables)."

The staff presented a brief technical and regulatory background, and the research results for resolution of GSI 168. The Office of Nuclear Regulatory Research (RES) sponsored the research program to resolve issues related to the qualification of certain electric components used in commercial nuclear power plants. Brookhaven National Laboratory (BNL) was selected as the lead laboratory to provide the technical assistance to RES. The objective of this research program was to provide information to assist the staff to resolve specific issues related to the EQ process for low-voltage I&C electric cables. Initially, a comprehensive list of 43 issues were identified. Based on a through review and analysis of the literature, 24 issues were resolved by considering past research results, and 19 issues were unresolved. Of the latter, six issues were identified that required additional testing of the cables to resolve. These issues are summarized below:

- How do the properties of cables subjected to accelerated aging techniques used in the original qualification compare with the properties of naturally aged cables of equivalent age?

- What are the limitations in using an estimated value of the activation energy to predict the chemical degradation during thermal aging?

- Do multiconductor cables have different failure mechanisms than single conductor cables, and, if so, are these unique failure mechanisms properly accounted for in the qualification process?

- Do bonded jacket cables have different failure mechanisms than unbonded jacket cables, and, if so, are these unique failure mechanism properly accounted for in the qualification process?

- Are there any effective condition monitoring techniques for determining cable condition in situ?

- Can condition monitoring techniques be used to predict LOCA survivability?

The staff concluded that the research test results suggested that some of the cable types would not function during a LOCA after 40 years of service. Most of them would not survive after 60 years of service, if they are operated at rated temperatures. It should be noted that most cables operate at temperatures significantly less than the rated temperature. Further, the staff concluded that risk studies give relatively high core damage frequency values, conditioned on all of the cables failing during LOCA. Additionally, research data would not support a sufficiently low failure rate to reduce the core damage frequency values to an acceptably low level.

Conclusion

This briefing was for information only and no Committee action was required on this matter.

7. ACRS Review of Generic Guidance Documents Associated with License Renewal

Dr. Bonaca, Chairman of the Plant License Renewal Subcommittee, noted that the staff and industry were developing a set of guidance documents for preparing and reviewing license renewal applications. He explained that his Subcommittee plans to review these documents during a meeting on October 19-20, 2000. The Members identified and discussed questions resulting from their initial review of the documents.

Conclusion

This discussion was for information only.

8. Annual Report to the Commission on the 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. The Committee indicated that the focus of its report will be on the long-term research needed to facilitate the execution of the NRC's mission in the future. The Committee also discussed certain acceptance criteria that could be useful in evaluating the research programs.

Conclusion

The Committee will continue its discussion and preparation of the ACRS 2001 report to the Commission on the NRC safety research program during future ACRS meetings. A Subcommittee meeting with the NRC staff has been scheduled for November 1, 2000, to discuss this matter.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee discussed the response from the EDO, dated August 31, 2000, to ACRS comments and recommendations included in the ACRS letter dated July 20, 2000, concerning the proposed final ASME Standard for PRA for Nuclear Power Plant Applications.

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

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from August 29 through October 4, 2000, the following Subcommittee meetings were held:

- Materials and Metallurgy Subcommittee - September 21, 2000

  The Subcommittee on Materials and Metallurgy discussed the status of activities associated with the Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project. These activities included determining a flaw distribution, embrittlement correlation, and fracture toughness.

- Planning and Procedures - October 4, 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.

PROPOSED SCHEDULE FOR THE 477TH ACRS MEETING

The Committee agreed to consider the following topics during the 477th ACRS Meeting, November 2-4, 2000:

Revised Report of the Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants
Briefing by and discussions with representatives of the NRC staff regarding the revised version of the report and the staff's response to previous ACRS concerns.

Risk-Informed Regulation Implementation Plan (RIRIP)
Briefing by and discussions with representatives of the NRC staff regarding the update to the RIRIP.

Proposed Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50

Briefing by and discussions with representatives of the NRC staff regarding the proposed NRC framework for risk-informed changes to the technical requirements of 10 CFR Part 50 described in Attachment 1 to SECY-00-0198.

Differing Professional Opinion (DPO) on Steam Generator Tube Integrity

Report by the Chairman of the Ad Hoc Subcommittee on DPO issues regarding the outcome of the October 10-14 subcommittee meeting, proposed subcommittee recommendations, schedule for completing the review, and related matters.

Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues

Briefing by and discussions with representatives of the NRC staff, the Nuclear Energy Institute, and the National Fire Protection Association (NFPA) on the revised NFPA 805 standard, post-fire safe shutdown circuit analysis, and other related fire protection issues.

ABB/CE and Siemens Digital I&C Applications

Report by the Subcommittee Chairman on a subcommittee meeting on this matter and his recommendation regarding further review by the full Committee.

License Renewal Guidance Documents

Briefing by and discussions with representatives of the NRC staff regarding the proposed Standard Review Plan for License Renewal, the Generic Aging Lessons Learned Report, a Regulatory Guide, and NEI 95-10, "Industry Guidelines for Implementing the Requirements of the License Renewal Rule."

Research Report to the Commission

Discussion of the status of the draft ACRS report on the NRC Safety Research Program.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

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

November 21, 2000

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

Sherry Meador

SUBJECT: PROPOSED MINUTES OF THE 476th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - OCTOBER 5-7, 2000

Enclosed are the proposed minutes of the 476th 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

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

December 1, 2000

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 476th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), OCTOBER 5-7, 2000

I certify that based on my review of the minutes from the 476th 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

December 1, 2000
Date

CERTIFIED

Date Issued: 11/21/2000
Date Certified: 12/1/2000

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OCTOBER 5-7, 2000

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APPENDICES

I. Federal Register Notice
II. Meeting Schedule and Outline
III. Attendees
IV. Future Agenda and Subcommittee Activities
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CERTIFIED

MINUTES OF THE 476TH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
October 5-7, 2000
ROCKVILLE, MARYLAND

The 476th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on October 5-7, 2000. Notice of this meeting was published in the *Federal Register* on September 20, 2000 (65 FR 56945) (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, MS 1F-15, 11555 Rockville Pike, 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. <u>Chairman's Report</u> (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. Discussion of Union of Concerned Scientists Report, "Nuclear Plant Risk Studies: Failing the Grade" (Open)

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

Dr. George Apostolakis, Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment introduced the topic. He stated that the purpose of this meeting was to review the Union of Concerned Scientists (UCS) August 2000 report entitled, "Nuclear Plant Risk Studies: Failing the Grade," concerning the use of risk information in NRC decision making. He stated that the UCS report was critical of the NRC's use of risk information in regulatory matters and noted that the ACRS has been supportive of initiatives related to risk-informed regulation. He introduced Mr. David Lochbaum, nuclear safety engineer at UCS, and noted that the Committee had met previously with Mr. Lochbaum during the October 5-7, 2000 ACRS meeting, to discuss this matter.

UCS Presentation

Mr. Lochbaum discussed the issues raised in the report as well as feedback that UCS has received from interested parties. He informed the Committee that the UCS report has been criticized for using obsolete results from outdated Individual Plant Examinations (IPEs). Mr. Lochbaum stated that UCS would prefer to have used updated probabilistic risk assessments (PRAs) and noted that licensee PRAs are not available for public review. UCS also noted that the same IPEs were used in the development of the significance determination process (SDP) of the revised reactor oversight process RROP. The Committee also discussed the UCS concern over differing PRA results for plants of like design or "sister" plants.

Mr. Lochbaum presented several slides taken from the NRC system reliability studies completed by the Office of Nuclear Regulatory Research (RES) (studies completed by the former Office for Analysis and Evaluation of Operational Data). He highlighted aspects of these studies (e.g., isolation condenser, high pressure core injection, reactor core isolation cooling systems, etc.) to illustrate the UCS view that the IPEs are seriously flawed in that they fail to estimate accurately and conservatively system unreliability. UCS contends that there is system-level bias in all these studies, in part, due to industry partitioning and rounding of operational data. UCS concluded that these inaccuracies and non-conservative estimates form an unsuitable foundation for regulatory decision making.

NRC Staff Comments

At the conclusion of the briefing, Messrs. Richard Barrett and Gareth Parry, NRR, provided comments on the UCS presentation and associated report. The staff acknowledged that there is a lot of information in the UCS report that is technically correct and took exception with the UCS assertion that NRC decisions have not been judicious. It was stated that lessons from past operating experience as well as issues related to PRA quality are being and will continue to be addressed. Identified discrepancies with "sister" plants are being evaluated (e.g., Wolf Creek and Callaway, etc.) and that the staff plans to examine the results of licensee/industry peer review processes as they become available. The staff also stated that IPEs were used as a first step in getting the SDP process started as a screening tool for inspection. The staff informed the Committee that they expect the SDP to become more plant-specific as experience is gained in using the SDP and in more closely examining PRAs related to plant operating experience.

Dr. Apostolakis questioned whether UCS viewed PRA as a weak, immature technology. Dr. Bonaca questioned whether UCS dislikes the use or treatment of bottom-line numbers (i.e., quantitative analysis). Mr. Lochbaum stated that the major concern is that the control over the inputs, assumptions, and methodology would not prevent abuses, whether institutional or accidental. Dr. Apostolakis questioned whether the NRC's use of risk information had led to incorrect or improper decisions. Mr. Lochbaum stated that the he is not suggesting that the "risk-informed road" is wrong, just that the road should be fixed before going down it. Dr. Apostolakis noted that there is a dichotomy between Mr. Lochbaum's statements and the messages codified in the UCS report and agreed that key is the validity of regulatory decision making.

Dr. Bonaca noted that the trend curve presented by Mr. Lochbaum was encouraging in that it provided evidence of a decline in design-basis related accident sequence precursor (ASP) events, in part, due to the increased attention to design basis issues from 1995-1997. Mr. Lochbaum stated that the trend chart does not indicate that there were no risk-significant design-basis issues. Concerning the study of high pressure core injection reliability, Mr. Lochbaum noted that some data was outside the error band and questioned the use of generic data. Dr. Apostolakis agreed that plant-specific data should be used. Mr. Lochbaum also noted that many other parts of the NRC are unaware of the NRC reliability studies. Members of the Committee agreed and noted that the ACRS has previously expressed similar concerns regarding the awareness and limited use of these studies.

Dr. Powers questioned how the public can gain access to plant-specific risk analysis in order to have confidence. Mr. Lochbaum stated that it would have to be entered into the licensing docket for the individual plant and acknowledged that the information can be voluminous and difficult to review. Dr. Apostolakis suggested that the critical information to consider is the risk analysis which supports a particular regulatory decision and/or application.

Conclusion

The Committee provided a report to Chairman Meserve dated October 11, 2000, on this matter.

III. NEI 00-02, "Industry PRA Peer Review Process Guidelines" (Open)

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

Dr. George Apostolakis, Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment, introduced the topic. He stated that the purpose of this meeting was to review the proposed industry certification process described in the document NEI 00-02, "Industry PRA Peer Review Process Guidelines."

Industry Presentation

Messrs. Biff Bradley of the Nuclear Energy Institute (NEI) and Karl Fleming of ERIN Engineering, Inc., led the discussion concerning NEI 00-02. Mr. Bradley provided an overview of NEI 00-02, including the industry's planned use of the peer review results to facilitate focused NRC review (i.e., as a template for NRC review) of risk-informed revisions to the special treatment requirements of 10 CFR Part 50 (Option 2). Mr. Fleming discussed a case study involving the Byron/Braidwood nuclear power plants. Significant points made during the presentation include:

- NEI 00-02 was developed from the peer review process initially developed by the BWR Owners Group.
- The certification process does not provide an overall grade for PRAs, and can be used as a complement to or in lieu of industrial standards for PRA quality. The industry supports development of a PRA Standard(s), however, an overly prescriptive Standard cannot ensure quality. Similarly,

a Standard cannot obviate the need for NRC review of specific applications.

- PRA is inherently judgmental. Therefore, a high-quality peer review will always be required.

- The peer review process will normally involve 6-7 reviewers with detailed knowledge of nuclear power plants and associated PRAs. It involves an evaluation of PRA elements and sub-elements via a structured process including the use of checklists to ensure a consistent, comprehensive review.

- The process will normally involve 2-3 person-months of effort, including document reviews, onsite review, and preparation of the final report. The first case study was for the Byron and Braidwood nuclear stations. All NSSS owners groups have committed to implementing NEI 00-02.

- The industry recognizes that some existing PRAs need to be improved in order to support regulatory decision making. NEI agreed with the UCS criticism that there is a need to improve industry PRAs.

At the conclusion of the meeting, NEI invited the ACRS and NRC staff to observe and/or participate in the peer review process in order to gauge the rigor applied to licensee risk-informed decisions using the peer review panel.

Dr. Apostolakis questioned whether licensees would request, using lower-level certified PRAs, regulatory decisions where the PRA is not sufficiently broad. He suggested that a lot of debate could be avoided if licensees would pursue certifying a Grade or Category 3 PRA rather than settling for limited approaches at the Grades 1 and 2 level. Mr. Fleming stated that resolving the A- and B-Type issues will require getting to Grade 3, and that the main focus was to benchmark the current PRAs, identify what information is needed to upgrade those PRAs, and decide on what needs to be done for additional applications.

Dr. Apostolakis requested to review a copy of the Byron and Braidwood case study documents. The NRC staff agreed to provide the docketed materials associated with this matter.

<u>Conclusion</u>

This briefing was for information only. No Committee report was required.

IV. Staff Views on ASME Standard for PRA for Nuclear Power Plant Applications (Open)

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

Dr. George Apostolakis, Chairman of the ACRS Subcommittee on Probabilistic Risk Assessment, introduced this topic. He indicated that the Committee had reviewed the ASME Standard, however, did not have the opportunity to review the staff's comments on the standard.

NRC Staff Presentation

The presentation on the Staff Views of ASME Standard for PRA for Nuclear Power Plant Applications was made by Ms. Mary Drouin, RES and Mr. Gareth Parry, NRR. The staff comments were on Revision 12 of the draft Standard. SECY 00-0162 was used as guidance in formulating staff comments on the draft. The issue of PRA quality is central to risk-informed regulations and one that the Commission has continually raised with the staff. As a result of review of the draft standard, the staff concluded that the current version of the standard (1) does not address PRA quality, (2) is difficult to use in determining where there are strengths and weaknesses in the PRA results, (3) will provide limited assistance to staff in performing focused review of PRA submittals, and (4) will provide minimal assistance in making more efficient use of NRC resources.

Following this discussion was a chapter by chapter discussion of the major concerns that led to the conclusions. Among the concerns discussed were the definition of the categories within which the PRA applications would fit, the requirements for the risk assessment application process, the completeness and accuracy of the technical content of the document, and the changes needed to be made to the current document.

Conclusion

This briefing was for information only. No committee action was required.

V. Pressurized Thermal Shock Technical Bases Reevaluation Project (Open)

[Note: Mr. Noel F. Dudley was the Designated Federal Official for this portion of the meeting.]

Mr. William Shack, Chairman of the Materials and Metallurgy Subcommittee, stated that the Subcommittee had a very good and full discussion of the status of the Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project at its September 21, 2000 meeting. He explained that the staff would walk through the overall probabilistic fracture mechanics calculation to show the Committee how all the pieces of the Reevaluation Project fit together and would provide a more detailed description of the development of a fracture toughness distribution. Dr. Shack mentioned that the source term used in developing the PTS screening criterion may be inappropriate in that assumes steam oxidation of the fuel while PTS will result in air oxidation of the fuel.

Mr. Michael Mayfield, RES, explained that three different divisions in RES were working on the PTS Reevaluation Project and that the Reevaluation Project would continue over the next year. Mr. Terry Dickson, Oak Ridge National Laboratory, explained how the results of the Fracture Analysis of Vessels: Oak Ridge (FAVOR) code can be applied to the PTS Reevaluation Project.. He described the integration into the FAVOR code of the results of different evolving technologies, such as, embrittlement correlations and fracture toughness models. He explained the structure of the code and the overall methodology for incorporating probabilistic risk assessment information.

The ACRS Members and the staff discussed the effects of lowering the present acceptance value, the effects of surface breaking cracks, existence of a possible thermal plume in the hot leg downcomer, and neutron fluence maps.

Mr. Mark Kirk, RES, provided a status of the development of fracture toughness distributions and the associated uncertainty analysis. He explained that the goal of the activity is to characterize toughness using all available data in a way that is consistent with PRAs. He described the approach used and the addition of new data to the database. Mr. Kirk outlined the uncertainty framework and he summarized some of the initial results and the on-going activities.

The ACRS Members and the staff discussed the appropriateness of using a Weibull to characterize the fracture toughness data and a one dimensional finite analysis to characterize crack distributions. They also discussed the effects of neutron fluence, characterization of flaws, damage accumulation, and calculating thermal-hydraulic uncertainties.

<u>Conclusion</u>

The Committee provided a letter dated October 12, 2000, to the EDO on this matter.

VI. Discussion of Industry Issues (Open)

[Note: Dr. Richard P. Savio was the Designated Federal Official for this portion of the meeting]

The Committee met with Ralph Beedle, NEI Senior Vice President and Chief Nuclear Officer, Nuclear Generation Division and members of his staff to discuss NEI's current regulatory initiatives, emergency industry issues, NEI recent reorganization, and other issues of mutual interest. The mutual interest issues included: risk-informing 10 CFR Part 50; license renewal; decommissioning; and the revised reactor oversight process.

NEI representatives frequently meet with the ACRS to discuss particular regulatory issues. These discussions focus on strategic planning, regulatory philosophy, and ACRS and NEI views on emergency issues.

Conclusion

The discussions were informative and productive and the ACRS Chairman plans to schedule this type of stakeholder exchange on a regular basis.

VII. GSI-168, Equipment Qualification (Open)

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

Dr. Robert E. Uhrig, Chairman of the ACRS Subcommittee on Plant Systems, introduced this topic. He stated that the safety-related electrical cables, primarily low voltage as well as medium voltage, associated with transmission of signals from plant to I&C devices and instruments conduct power to safety-related devices. The environmental qualification issue started in 1971. He stated that the purpose of this briefing was to discuss the proposed resolution and the status of Generic Safety Issue (GSI) 168, "Environmental Qualification (EQ) of Low-Voltage Instrumentation and Control (I&C Cables)."

NRC Staff Presentation

Mr. Mike Mayfield led the staff's discussion of GSI-168 and stated that the staff is not prepared to present the closure package to the Committee. Mr. Edward Hackett presented a brief technical and regulatory background of the research results for resolution of GSI 168. RES sponsored the research program to resolve issues related to the process used for qualification of certain electric components used in commercial nuclear power plants. Brookhaven National Laboratory (BNL) was selected as the lead laboratory to provide the technical assistance to RES. The objective of this research program was to provide information to assist the staff to resolve specific issues related to the EQ process for low-voltage I&C electric cables. Initially, a comprehensive list of 43 issues was identified. Based on a through review and analysis of the literature, 24 issues were resolved by considering past research results, and 19 issues were unresolved. Of the latter, 6 issues were identified and required additional testing of the cables to resolve. These issues are summarized below:

- How do the properties of cables subjected to accelerated aging techniques used in the original qualification compare with the properties of naturally aged cables of equivalent age?
- What are the limitations in using an estimated value of the activation energy to predict the chemical degradation during thermal aging?
- Do multiconductor cables have different failure mechanisms than single conductor cables, and, if so, are these unique failure mechanisms properly accounted for in the qualification process?
- Do bonded jacket cables have different failure mechanisms than unbonded jacket cables, and, if so, are these unique failure mechanism properly accounted for in the qualification process?
- Are there any effective condition monitoring techniques for determining cable condition in situ?
- Can condition monitoring techniques be used to predict loss-of-coolant accident (LOCA) survivability?

Mr. Hackett concluded that the research test results suggested that some of the cable types would not function during a LOCA after 40 years of service. Most of them would not survive after 60 years of service, if they are operated at rated temperatures. Further, the staff concluded that risk studies give relatively high CDF values, conditioned on all of cables failing during LOCA. Additionally, research data would not support a sufficiently low failure rate to reduce the CDF values to an acceptably low level.

Conclusion

This briefing was for information only and no Committee action was required on this matter.

VIII. ACRS Review of Generic Guidance Documents Associated with License Renewal (Open)

[Note: Mr. Noel F. Dudley was the Designated Federal Official for this portion of the meeting.]

Dr. Bonaca, Chairman of the Plant License Renewal Subcommittee, noted that the staff and industry were developing a set of guidance documents for preparing and reviewing license renewal applications. He explained that the Subcommittee planned to review these documents during the October 19-20, 2000 ACRS subcommittee meeting. The members identified and discussed questions resulting from their initial reviews of the documents.

Conclusion

This session was for information only.

IX. Annual Report to the Commission on the NRC Safety Research Program

[Note: 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. The Committee indicated that the focus of its report will be on the long-term research needed to facilitate the execution of the NRC's mission in the future. The Committee also discussed certain acceptance criteria that could be useful in evaluating the research programs. The acceptance criteria may include metrics such as regulatory needs, objectives, level of effort, probability of success, why NRC should perform such research programs and not the industry, and public confidence.

Conclusion

The Committee will continue its discussion and preparation of the ACRS 2001 report to the Commission on the NRC research programs during future ACRS meetings. A subcommittee meeting on the report has been scheduled for November 1, 2000.

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 (Open)

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

- The Committee discussed the response from the EDO, dated August 31, 2000, to ACRS comments and recommendations included in the ACRS letter dated July 20, 2000, concerning the proposed final ASME Standard for PRA for Nuclear Power Plant Applications.

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

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

The Committee heard a report from Dr. Powers and the Executive Director, ACRS, on the Planning and Procedures Subcommittee meeting held on October 4, 2000. The following items were discussed:

Review of the Member Assignments and Priorities for ACRS Reports and Letters for the October ACRS Meeting
Member assignments and priorities for ACRS reports and letters for the October ACRS meeting are included in a separate handout. Reports and letters that would benefit from additional consideration at a future ACRS meeting was discussed.

Anticipated Workload for ACRS Members
The anticipated workload of the ACRS members through December 2000 was discussed. 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.

Issues Associated with Core Power Uprates
The Planning and Procedures Subcommittee discussed the central issues pertaining to core power uprates, especially those identified by an ACRS Senior Fellow, Dr. Cronenberg [Note: There has been outside interest regarding the report prepared by Dr. Cronenberg]. In addition, the Subcommittee discussed a proposed White paper prepared by Dr. Bonaca on "Potential Synergistic Effects of Industry Initiatives to Extend Plant Life, Increase Production, and Reduce Regulatory Burden." The full Committee agreed with the Subcommittee's recommendation that the issues raised by Dr. Bonaca in his White paper be included in the ensuing ACRS report to the Commission on the NRC Safety Research Program. Also, the Subcommittee decided to continue its discussion of the issues associated with the core power uprates and related matters during its October 4, 2000 meeting. The staff will brief the Committee during the December 2000 ACRS meeting. Proposed topics include:

- NRR position regarding the need for applying risk-informed decisionmaking to "significant" power uprate applications. Criteria for determining "special circumstances."
- NRR plans for revising guidance documents and developing a Standard Review Plan Section for power uprate reviews.
- Synergisms among changes in nuclear plants and their potential impact on plant safety.
- Higher burnup fuel
- Power uprates
- Use of "best-estimate" or "more-realistic" analyses
- Plant life extension

- RES activities associated with core power uprates.

Mixed Oxide Fuel Fabrication Facility
The Department of Energy (DOE) plans to construct a Mixed Oxide (MOX) Fuel Fabrication Facility (FFF) on its Savannah River site to manufacture fuel from weapons program plutonium. A consortium composed of Duke Power, Cogema Fuels, and Stone & Webster (DCS) will construct and operate the facility for DOE. The NRC staff expects the application for construction authorization to be submitted in December 2000 and the operating license application in June 2002. NRC staff review of the safety design basis for the MOX FFF will occur at the construction authorization stage. The NRC staff also expects DCS to submit the license amendment for loading MOX fuel test assemblies in McGuire Unit 2 in August 2001. A license amendment for batch loading of MOX fuel in the McGuire and Catawba reactors is expected in December 2003. The Commission is interested in the ACRS review of and comment on the MOX FFF. Review of the safety issues associated with the use of MOX fuel in operating reactors is within the purview of the ACRS.

Meeting with the Nuclear Energy Institute
As part of the CY 1998 and CY 1999 ACRS self assessment, feedback was solicited from various stakeholders, including some industry groups. Some stakeholders stated that the ACRS was not sufficiently informed of the industry's concerns and needs. As a result, a meeting with representatives of NEI was scheduled for the October 5-7, 2000 ACRS meeting to discuss items of mutual interest. Topics agreed to by NEI are provided below:

- Risk informing 10 CFR Part 50)
- License renewal
- Decommissioning
- Revised reactor oversight process

Estimation of Resources for FY 2001
Due to the anticipated high workload facing the ACRS in FY 2001, it is important to plan how to use member time most efficiently and effectively. Assuming the number of ACRS members remain constant throughout FY 2001, the maximum member time that will be available is 1,300 days.

During last month's Planning and Procedures Subcommittee meeting, we discussed the need to manage better the number of subcommittee meetings and the number of members participating in subcommittee meetings. Senior staff engineers with input from Subcommittee chairmen were asked to revise the estimate of the number of subcommittee meetings for FY 2001. The current estimate shows 36 subcommittee meetings, 10 full Committee meetings and 1 retreat, consuming a total of approximately 1,155 days. The Planning and Procedures subcommittee will need to scrutinize these proposed subcommittee meetings to assess where some cuts might be made or combining of subcommittee meetings might be done. This needs to be done to make sure we do not exceed the maximum days available for members to work and also not to overburden members.

Progress Made in Addressing the Commitments Resulting from the CY 1999 ACRS/ACNW Self Assessment for CY 2000

Results of the ACRS/ACNW self assessment for CY 1999 were documented in SECY-00-0102 and sent to the Commission on May 5, 2000. In that SECY paper, the ACRS had made several commitments. The Subcommittee discussed the actions taken to address these commitments and provided feedback on the adequacy of these actions. The Subcommittee also provided guidance for conducting the CY 2000 self assessment.

ACRS Retreat for 2001

During the September meeting, the Committee agreed to have a retreat in January 2001 together with its visit to the San Onofre Nuclear Plant. Since the San Onofre visit did not materialize, a decision on the dates and location for the retreat should be made.

Proposed ACRS Meeting Dates for CY 2001

The proposed dates for CY 2001 ACRS meetings were discussed.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 477th ACRS Meeting, November2-4, 2000.

The 476th ACRS meeting was adjourned at 12:30 p.m. on October 7, 2000.

Room link at the NRC Web site (http://www.nrc.gov).

Dated at Rockville, Maryland, this 14th day of September 2000.

For the Nuclear Regulatory Commission.

Girija S. Shukla,

Project Manager, Section 2, Project Directorate IV and Decommissioning Division of Licensing Project Management Office of Nuclear Reactor Regulation.

[FR Doc. 00–24162 Filed 9–19–00; 8:45 am]

BILLING CODE 7590-01-P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Meeting of the Ad Hoc Subcommittee; Notice of Meeting

The Ad Hoc Subcommittee will hold a meeting on October 10–13, 2000, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Tuesday, October 10, 2000—1:00 p.m. until the conclusion of business

The Ad Hoc Subcommittee will discuss its approach for reviewing the technical merits of the Differing Professional Opinion (DPO) issues associated with steam generator tube integrity, and developing comments and recommendations for consideration by the full Committee.

Wednesday, October 11, 2000—8:30 a.m. until the conclusion of business

The Ad Hoc Subcommittee will hear presentations by and hold discussions with the DPO author and other interested persons regarding the DPO issues, views on the adequacy of the staff's approach for resolving these issues, and remaining major issues of contention.

Thursday, October 12, 2000—8:30 a.m. until the conclusion of business

The Ad Hoc Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff and other interested persons regarding the status of resolution of the DPO issues and related matters.

Friday, October 13, 2000—8:30 a.m. until the conclusion of business

The Ad Hoc Subcommittee will continue its discussion of the DPO issues with the staff and the DPO author, as needed, and will develop proposed comments and recommendations for consideration by the full Committee.

The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

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

During the initial portion of the meeting, the Ad Hoc Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Ad Hoc Subcommittee will then hear presentations by and hold discussions with the DPO author, representatives of the NRC staff, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting either Mr. Sam Duraiswamy (telephone 301–415–7364) or Ms. Undine Shoop (telephone 301–415–8086) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individuals one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: September 12, 2000.

James E. Lyons,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 00–24159 Filed 9–19–00; 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 October 5–7, 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).

Thursday, October 5, 2000

8:30 A.M.–8:45 A.M.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:45 A.M.–10:00 A.M.: Discussion of Union of Concerned Scientists Report, "Nuclear Plant Risk Studies: Failing the Grade" (Open)—The Committee will hear presentations by and hold discussions with representatives of the Union of Concerned Scientists (UCS), the NRC staff, and other interested parties concerning the August 2000 UCS report on nuclear plant risk studies.

10:15 A.M.–11:30 A.M.: NEI 00–02, "Industry PRA Peer Review Process Guidelines" (Open)—The Committee will hear presentations by and hold discussions with representatives of the Nuclear Energy Institute (NEI) and the NRC staff regarding the proposed industry PRA certification guidelines described in the document NEI 00–02.

11:30 A.M.–12:30 P.M.: Staff Views on ASME Standard for PRA for Nuclear Power Plant Applications (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the staff's August 14, 2000 response to the American Society of Mechanical Engineers (ASME) draft Revision 12 ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications.

1:30 P.M.–3:00 P.M.: Pressurized Thermal Shock Technical Bases Reevaluation Project (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the pressurized thermal shock technical bases reevaluation project.

3:30 P.M.–4:30 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.

4:30 P.M.–6:00 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

6:00 P.M.–7:00 P.M.: Discussion of Topics for Meeting with the NRC Commissioners (Open)—The Committee will discuss issues associated with risk informing 10 CFR 50, quality of PRAs, spent fuel pool fire safety study, more realistic (best estimate) thermal-hydraulic codes and status of ACRS activities on license renewals.

Friday, October 6, 2000

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

8:35 A.M.–9:15 A.M.: Discussion of Topics for Meeting with the NRC Commissioners (Open)—The Committee will discuss matters scheduled for the meeting with the NRC Commissioners associated with risk informing 10 CFR 50 and related matters.

9:30 A.M.–12:00 Noon: Meeting with the NRC Commissioners (Open)—The Committee will meet with the NRC Commissioners, Commissioners' Conference Room, One White Flint North to discuss risk informing 10 CFR 50 and related matters.

1:30 P.M.–3:00 P.M.: Discussion of Industry Issues (Open)—The Committee will hear a presentation by R. Beedle, Senior Vice President, NEI on issues of mutual interest.

3:15 P.M.–4:45 P.M.: GSI–168, Equipment Qualification (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the GSI–168, Equipment Qualification.

4:45 P.M.–5:30 P.M.: ACRS Review of Generic Guidance Documents Associated with License Renewal (Open)—The Committee will discuss concerns identified during their initial review of the draft guidance documents.

5:30 P.M.–5:50 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.

5:50 P.M.–6:00 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.

6:00 P.M.–6:30 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.

6:30 P.M.–7:30 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, October 7, 2000

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

8:35 A.M.–12:30 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

12:30 P.M.–1 P.M.: Annual Report to the Commission on the NRC Safety Research Program (Open)—The Committee will discuss the format and content of the annual ACRS report to the Commission on the NRC Safety Research Program.

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

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 28, 1999 (64 FR 52353). 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., EDT.

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

Dated: September 14, 2000.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 00–24160 Filed 9–19–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

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

September 13, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
476TH ACRS MEETING
OCTOBER 5-7, 2000

THURSDAY, OCTOBER 5, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:45 A.M. Opening Remarks by the ACRS Chairman (Open)
- 1.1) Opening statement (DAP/JTL/HJL)
- 1.2) Items of current interest (DAP/NFD/HJL)
- 1.3) Priorities for preparation of ACRS reports (DAP/JTL/HJL)

2) 8:45 - ~~10:00~~ 9:45 A.M. Discussion of Union of Concerned Scientists Report, "Nuclear Plant Risk Studies: Failing the Grade" (Open) (GA/MTM)
Briefing by and discussions with representatives of the Union of Concerned Scientists (UCS), the NRC staff, and other interested parties concerning the August 2000 UCS report on nuclear plant risk studies.

~~10:00~~ 9:45 - 10:15 A.M. ***BREAK***

3) 10:15 - ~~11:30~~ 11:40 A.M. NEI 00-02, "Industry PRA Peer Review Process Guidelines" (Open) (GA/MTM)
- 3.1) Opening remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the Nuclear Energy Institute (NEI) and the NRC staff regarding the proposed industry PRA certification guidelines described in the document NEI 00-02.

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

4) ~~11:30~~ - ~~12:30~~ P.M. 11:40-12:40 Staff Views on ASME Standard for PRA for Nuclear Power Plant Applications (Open) (GA/MTM)
- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's August 14, 2000 response to the American Society of Mechanical Engineers (ASME) draft Revision 12 ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications.

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

~~12:30~~ - ~~1:30~~ P.M. 12:40-1:40 ***LUNCH***

5) 1:40 ~~1:30~~ - 3:30 P.M. Pressurized Thermal Shock Technical Bases Reevaluation Project (Open) (WJS/NFD)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the pressurized thermal shock technical bases reevaluation project.

3:40-3:50 Break

6) ~~3:30 - 4:30 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) 3:50 ~~4:30~~ - 6:00 P.M. Discussion of Proposed ACRS Reports (Open)
Preparation of proposed ACRS reports on:

3:55-7:15 7.1) Union of Concerned Scientists Report on Nuclear Plant Risk Studies (GA/MTM)

7.2) NEI 00-02, "Industry PRA Peer Review Process Guidelines" (GA/MTM)

7.3) Pressurized Thermal Shock Technical Bases Reevaluation Project (WJS/NFD)

7:30 End of Session

8) 6:00 - ~~7:00~~ P.M. ~~Discussion of Topics for Meeting with the NRC Commissioners (Open) (DAP, et al./JTL, et al.)~~
~~Discussion of topics and preparation for meeting with the NRC Commissioners scheduled for 9:30 a.m. - 12:00 Noon, Friday, October 6 concerning:~~

~~8.1) Risk Informing 10 CFR 50 (WJS/MTM)~~
- ~~NEI Letter of January 19, 2000~~
- ~~Proposed Revision to 10 CFR 50.44 Concerning Combustible Gas Control System and Advance Notice of Proposed Rulemaking (10 CFR 50.69 and Appendix T)~~

~~8.2) Quality of PRAs (GA/MWW)~~
- ~~Assessment of the Quality of PRAs~~
- ~~ASME Standard on PRAs~~

~~8.3) Spent Fuel Pool Fire Safety Study (TSK/MME)~~

~~8.4) More Realistic (Best Estimate) Thermal-Hydraulic Codes (GW/PAB)~~

~~8.5) Status of ACRS Activities on License Renewals (MVB/NFD)~~

FRIDAY, OCTOBER 6, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

9) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (DAP/JTL)

10) 8:35 - 9:15 A.M. Discussion of Topics for Meeting with the NRC Commissioners (Open) (DAP, et al./JTL, et al.)
Discussion of topics listed under Item 8.

9:15 - 9:30 A.M. ***BREAK***

11) 9:30 - 12:00 Noon — Meeting with the NRC Commissioners (Open) (DAP, et al./JTL. et al.)
Meeting with the NRC Commissioners, Commissioners' Conference Room, One White Flint North, to discuss topics listed under Item 9 and other items of mutual interest.

12:00 - 1:30 P.M. — *****LUNCH*****

12) 1:30 - 3:00 P.M. — Discussion of Industry Issues (Open) (DAP/RPS)
Presentation by R. Beedle, Senior Vice President, NEI, on issues of mutual interest.

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

3:20 - 4:50
13) ~~3:15~~ - 4:45 P.M. — GSI-168, Equipment Qualification (Open) (REU/AS)

14.1) Remarks by the Subcommittee Chairman
14.2) Briefing by and discussions with representatives of the NRC staff regarding the GSI-168, Equipment Qualification.

5:50
14) 4:45 - ~~5:30~~ P.M. — ACRS Review of Generic Guidance Documents Associated with License Renewal (Open) (MVB/NFD)
The Committee members will discuss concerns identified during their initial review of the draft guidance documents.

5:50 - 6:30
15) ~~5:30~~ - ~~5:50~~ P.M. — Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (DAP/JTL/HJL)

15.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
15.2) 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.

16) 5:50 - 6:00 P.M. — Reconciliation of ACRS Comments and Recommendations (Open) (DAP, et al./HJL, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

17) 6:00 - 6:30 P.M. — Break and Preparation of Draft ACRS Reports
Cognizant ACRS members will prepare draft reports for consideration by the full Committee.

7:45
18) 6:30 - ~~7:30~~ P.M. — Discussion of Proposed ACRS Reports (Open)
Preparation of proposed ACRS reports on:

19.1) GSI-168, Equipment Qualification (REU/AS)
6:30-7:40 19.2) Union of Concerned Scientists Report on Nuclear Plant Risk Studies (GA/MTM)
19.3) NEI 00-02, "Industry PRA Peer Review Process Guidelines" (GA/MTM)
10/7/2000 - 8:30-9:40 19.4) Pressurized Thermal Shock Technical Bases Reevaluation Project (WJS/NFD)

SATURDAY, OCTOBER 7, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

19)	8:30 - 8:35 A.M.	Opening Remarks by the ACRS Chairman (Open) (DAP/JTL)
20)	8:35 - 12:30 P.M.	Discussion of Proposed ACRS Reports (Open) - The Committee will continue its discussion of proposed ACRS reports as noted in Item 19.
21)	12:30 - 1:00 P.M.	Annual Report to the Commission on the NRC Safety Research Program (Open) (DAP/MME) Discussion of the current status of the review by the members of the topical areas previously assigned.
22)	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.**

APPENDIX III: MEETING ATTENDEES

476TH ACRS MEETING
OCTOBER 5-7, 2000

NRC STAFF (October 5, 2000)
A. Levin, OCM/RAM
J. Williams, NRR
G. Parry, NRR
E. Throm, NRR
J. Bongarre, NRR
C. Douitt, NRR
J. Dozio, NRR
M. Cheok, NRR
S. Dinsmore, NRR
S. Magruder, NRR
R. Barrett, NRR
S. West, NRR
M. Rubin, NRR
E. McKenna, NRR
L. Lois, NRR
S. Mays, RES
J. Mitchell, RES
M. Drouin, RES
A. Ramey-Smith, RES
L. Abramson, RES
S. Malik, RES
N. Siu, RES
B. Jones, RES
M. Mayfield, RES
R. Woods, RES
E. Hackett, RES
D. Jackson, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
N. Chapman, SERCH/Bechtel
S. Vance, Shaw Pittman
D. Lochbaum, UCS
J. Meyer, ISL
H. Fonticella, Dominion
B. Bradley, NEI
J. Mallay, Siemens
K. Fleming, Erin Engineering & Research
B. Hansen, ENS
M. Knapik, McGraw-Hill

T. Dickson, ORNL
B. Handles, CCNPPI
M. Natishan, PEAI

NRC STAFF (October 6, 2000)
S. West, NRR
E. McKenna, NRR
P. Shemanski, NRR
J. Calvo, NRR
J. Mitchell, RES
J. Vora, RES
H. VanderMolen, RES
E. Hackett, RES
J. Rosenthal, RES
A. Buslik, RES
C. Poslusny, NMSS

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
J. Petro, Winston & Strawn
D. Raleigh, SERCH/Bechtel
L. Hendricks, NEI
A. Marion, NEI
D. Walters, NEI
A. Nelson, NEI
B. Horin, Winston & Strawn
G. Toman, EPRI

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

October 17, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
477TH ACRS MEETING
NOVEMBER 2-4, 2000

THURSDAY, NOVEMBER 2, 2000, 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 (DAP/JTL/HJL)
   - 1.2) Items of current interest (DAP/NFD/HJL)
   - 1.3) Priorities for preparation of ACRS reports (DAP/JTL/HJL)

2) 8:35 - 10:45 A.M. Revised Report of the Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (TSK/MME/MWW)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding the revised version of the report and the staff's response to previous ACRS concerns.

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

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

3) 11:00 - 12:30 P.M. Risk-Informed Regulation Implementation Plan (RIRIP) (Open) (GA/MTM)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding the update to the RIRIP.

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

12:30 - 1:30 P.M. *****LUNCH*****

4) 1:30 - 2:30 P.M. Proposed Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Open) (GA/MTM)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed NRC framework for risk-informed changes to the technical requirements of 10 CFR Part 50 described in Attachment 1 to SECY-00-0198.

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

5) 2:30 - 4:30 P.M. Differing Professional Opinion (DPO) on Steam Generator Tube Integrity (Open) (DAP/SD/US)

- 5.1) Report by the Chairman of the Ad Hoc Subcommittee on DPO Issues regarding the outcome of the October 10-14 subcommittee meeting, proposed subcommittee recommendations, schedule for completing the review, and related matters.
- 5.2) Briefing by and discussions with the DPO author and representatives of the NRC staff, as needed, on additional information related to DPO issues.

6) 4:30 - 5:30 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) 5:30 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:

- 7.1) Framework for Risk-Informed Changes to 10 CFR Part 50 (GA/MTM)
- 7.2) Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (TSK/MME/MWW)
- 7.3) Risk-Informed Regulation Implementation Plan (GA/MTM)
- 7.4) DPO on Steam Generator Tube Integrity (DAP/SD/US)

FRIDAY, NOVEMBER 3, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 - 10:30 A.M. Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues (Open) (JDS/DAP/AS)

- 9.1) Remarks by the Subcommittee Chairman
- 9.2) Briefing by and discussions with representatives of the NRC staff, Nuclear Energy Institute, and National Fire Protection Association (NFPA) on the revised NFPA 805 standard, post-fire safe shutdown circuit analysis, and other related fire protection issues.

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

10) 10:45 - 12:00 Noon ABB/CE and Siemens Digital I&C Applications (Open) (REU/AS)
Report by the Subcommittee Chairman on a subcommittee meeting on this matter and his recommendation regarding further review by the full Committee.

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

11)	1:00 - 3:00 P.M.	License Renewal Guidance Documents (Open) (MVB/RLS/NFD) 11.1) Remarks by the Subcommittee Chairman 11.2) Briefing by and discussions with representatives of the NRC staff regarding proposed Standard Review Plan for License Renewal, Generic Aging Lessons Learned Report, Regulatory Guide, and NEI 95-10, Industry Guidelines for Implementing the Requirements of the License Renewal Rule.. Representatives of the nuclear industry will provide their views, as appropriate.
	3:00 - 3:15 P.M.	*****BREAK*****
12)	3:15 - 4:30 P.M.	Research Report to the Commission (Open) (DAP/MME) Discussion of the status of the draft ACRS report on the NRC Safety Research Program. Representatives of the NRC staff will provide their views, as appropriate.
13)	4:30 - 5:00 P.M.	Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (DAP/JTL/HJL) 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, and organizational and personnel matters relating to the ACRS.
14)	5:00 - 5:15 P.M.	Reconciliation of ACRS Comments and Recommendations (Open) (DAP, et al./HJL, 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)	5:15 - 6:00 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)	6:00 - 7:30 P.M.	Proposed ACRS Reports (Open) Discussion of proposed ACRS reports on: 16.1) Framework for Risk-Informed Changes to 10 CFR Part 50 (GA/MTM) 16.2) Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (TSK/MME/MWW) 16.3) Risk-Informed Regulation Implementation Plan (GA/MTM) 16.4) DPO on Steam Generator Tube Integrity (DAP/SD/US) 16.5) Performance-Based, Risk-Informed Fire Protection Standard (JDS/DAP/AS)

16.6) Research Report to the Commission (DAP/MME)
16.7) License Renewal Guidance Documents (MVB/RLS/NFD)

SATURDAY, NOVEMBER 4, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

17) 8:30 - 1:00 P.M. Proposed ACRS Reports (Open) - The Committee will continue its discussion and preparation of proposed ACRS reports listed under item 16.

18) 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.**

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
476th ACRS MEETING
OCTOBER 5-7, 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 October 5-7, 2000

2 Discussion of Union of Concerned Scientists Report, "Nuclear Plant Risk Studies: Failing the Grade"
2. Presentation to ACRS on Nuclear Plant Risk Studies: Failing the Grade by D. Lochbaum, Union of Concerned Scientists [Viewgraphs]

3 NEI 00-02, "Industry PRA Peer Review Process Guidelines"
3. Industry PSA Peer Review Process presentation by NEI [Viewgraphs]
4. Role of the Industry PRA Peer Review Process to Support Quality PRA Applications presentation by K. Fleming, Erin Engineering and Research, Inc. [Viewgraphs]

4 Staff Views on ASME Standard for PRA for Nuclear Power Plant Applications
5. Recent Activities on ASME Revision 12 PRA Standard presentation by M. Drouin, RES, and G. Parry, NRR [Viewgraphs]

5 Pressurized Thermal Shock Technical Bases Reevaluation Project
6. Status of the Favor Code Development presentation by Oak Ridge National Laboratory [Viewgraphs]
7. Fracture Toughness Distributions and Uncertainty Analysis presentation by M. Kirk, RES [Viewgraphs]

12 Discussion of Industry Issues
8. Nuclear Energy Institute Organizational Chart, presentation by R. Beedle, Senior Vice President & Chief Nuclear Officer [Handout]

13 GSI-168, Equipment Qualification
9. Environmental Qualification (EQ) of Low-Voltage I&C Cables presentation by RES [Viewgraphs]
10. GSI-168 (Pre-Decisional) author RES [Handout 13-1]

14 ACRS Review of Generic Guidance Documents Associated with License Renewal

11. Guidance for ACRS Review of License Renewal Generic Document presentation by Dr. M. Bonaca [Handout]

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

12. Final Draft Minutes of Planning and Procedures Subcommittee Meeting - **XX**, 1998 [Handout #15.2]

16 Reconciliation of ACRS Comments and Recommendations

13. Reconciliation of ACRS Comments and Recommendations [Handout #**XX**]

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Discussion of Union of Concerned Scientists Report, "Nuclear Plant Risk Studies: Failing the Grade"
	1. Table of Contents
	2. Proposed Schedule
	3. Project Status Report, dated October 5, 2000
	4. NRC Staff's Talking Points on UCS report, dated August 22, 2000
	5. UCS report August 2000
3	NEI 00-02, "Industry PRA Peer Review Process"
	6. Table of Contents
	7. Proposed Schedule
	8. Status Report dated October 5, 2000
	9. NRR Request for additional information (RAI) dated September 19, 2000
	10. RES RAI dated August 11, 2000
	11. RES memorandum to NRR regarding request for technical assistance
	12. Letter from NRR to R. Beedle, NEI, dated June 9, 2000
	13. User needs request dated June 19, 2000
4	Staff Views on ASME Standard for PRA for Nuclear Power Plants
	14. Table of Contents
	15. Proposed Schedule
	16. Status Report dated October 5, 2000
	17. Memorandum dated August 17, 2000, from Ashok Thadani, RES, to John W. Craig, Assistant for Operations, EDO, Subject: Staff Comments on June 2000 Draft ASME Standard on PRA Quality
	18. Talking Points for 9/19/00 Conference Call on ASME Standard, RES
	19. Principles/Objectives for the ASME Standard, RES
	20. Letter dated September 7, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Assessment of the Quality of Probabilistic Risk Assessment
	21. SECY-00-0162, Memorandum dated July 28, 2000, from William D. Travers, EDO, NRC for the Commissioners, Subject: Addressing PRA Quality in Risk-Informed Activities
	22. Letter dated July 20, 2000, from Dana A. Powers, Chairman, ACRS, to William D. Travers, EDO, Subject: Proposal Final ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications
	23. Letter dated August 31, 2000, from William D. Travers, EDO, to Dana A. Powers, Chairman, ACRS, Subject: ACRS Letter dated July 20, 2000,

"Proposed Final ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Operations"

5 Pressure Thermal Shock Technical Basis Reevaluation Project

24. Table of Contents
25. Proposed Schedule
26. Status Report dated October 5, 2000
27. Bowman, K. and Williams, P., Oak Ridge National Laboratory, "Technical Basis for Statistical Models of Extend K_{Ic} and K_{Ia} Fracture Toughness Databases for RPV Steels," dated February 2000
28. Li, F. et al. University of Maryland, "K_{Ic}/K_{Ia} Uncertainty Characterization," dated June 23, 2000

12 Discussion of Industry Issues

29. Table of Contents
30. Proposed Schedule
31. Status Report dated October 6, 2000
32. NEI Strategic Plan - Nuclear Energy: The Renaissance Revealed - A Strategic Director for the 21st Century.
33. NEI 1999 Annual Report

13 Proposed Resolution of Generic Safety Issue 168, "Environmental Qualification of Electrical Equipment"

34. Table of Contents
35. Proposed Schedule
36. Status Report dated October 6, 2000

14 License Renewal Guidance Documents

37. Table of Contents
38. Proposed Schedule
39. Status Report dated October 6, 2000
40. Memorandum from Noel Dudley, ACRS/ACNW, to ACRS Members, Subject: ACRS Review Plans for License Renewal Guidance Documents, dated August 29, 2000
41. Status of License Renewal Issue Inventory

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

476 FULL COMMITTEE MEETING

October 5, 2000

NRC STAFF SIGN IN FOR ACRS MEETING

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GARETH PARRY	B-8060	NRR/DSSA
EDWARD D THROM	B-7179	NRR/DSSA
JAMES DONGARRA	B-6093	NRR/DIPM
CLIFF DOUTT	B-8622	NRR/DSSA/SPSB
Jerry Dozier	B 8726	NRR/DRIP/RLSB
Mike Cheok	B7917	NRR/DSSA/SPSB
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STEWART MAGRUDER	B-6721	NRR/DRIP/RGEB
RICH BARRETT	A-7370	NRR/DSSA/SPSB
ALAN LEVIN	A-7651	OCM/RAM
S. E. MAYS	B 8405	RES/DRAA/OERAB
Steve West	B7258	NRR
Jocelyn Mitchell	B-6685	RES
Mary Drouin	B-7583	RES
Steve Crockett	A-7549	OGC
[illegible]	B6966	RES/DRAA
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Eileen McKenna	B8226	NRR/DRIP/RGEB
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SHAH MALIK	B-7396	RES/DET/MEB
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
476TH FULL COMMITTEE MEETING

OCTOBER 5, 2000
Today's Date

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NAME	AFFILIATION
Nancy Chapman	SERCH/Bechtel
Scott Vance	Shaw Pittman
Dave Lochbaum	Union of Concerned Scientists
Jim Meyer	ISL
Herb Fontecilla	Dominion
Biff Bradley	NEI
James Mallay	Siemens
Karl Fleming	Erin Engineering & Research
Brian Hansen	ENS
Mike Knapik	McGraw-Hill
Terry Dickson	Oak Ridge National Laboratory
Mark Kirk	NRC
[illegible]	
Bob Hardies	CCNPPI
Marjorie Natishan	PEAI
~~Debbie Jackson~~	

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

OCTOBER 5-7, 2000
Date(s)

OCTOBER 6, 2000
Today's Date

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PAUL Shemanski	B-7076	NRR/DE
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Harold VanderMolen	B7214	RES/DSARE/REAHFB
ED HACKETT	B7487	RES/DET
Jack Rosenthal	A6661	RES/REAHFB
ARTHUR BUSLIK	B 6433	RES/PRAB
Jose A. Calvo	A 6314	NRR/EEIB
P T Kuo	B7543	NRR/RLSB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
476TH FULL COMMITTEE MEETING

OCTOBER 6, 2000
Today's Date

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Deann Raleigh	SELCH Bechtel Power
Lynnette Hendricks	NEI
Alex Marion	NEI
Doug Walters	NEI
Alan Nelson	NEI
Bill Horin	Winston + Strawn
Gary Toman	EPRI

476th ML003762440

Status of the FAVOR Code Development

Terry Dickson, Richard Bass, and Paul Williams
Heavy-Section Steel Technology Program
Oak Ridge National Laboratory

Advisory Committee on Reactor Safeguards
Pressurized Thermal Shock Screening Criterion Re-evaluation

October 5, 2000
Nuclear Regulatory Commission
Rockville, Maryland

Oak Ridge National Laboratory
U.S. Department of Energy

UT-BATTELLE

This Presentation Describes the Evolution of an Advanced Computational Tool for RPV Integrity Evaluations : FAVOR

1/ How FAVOR is Applied in PTS Re-evaluation

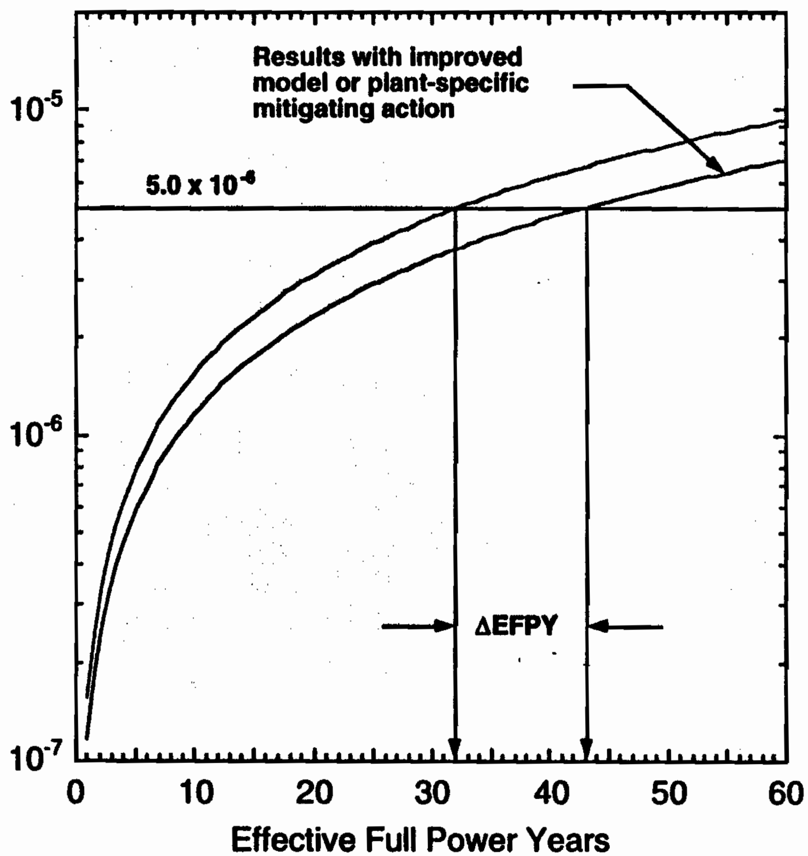
2/ Integration of Evolving Technology into FAVOR

3/ FAVOR Structure

4/ Overall PRA Methodology

5/ PFM Details

Application of FAVOR to PTS Re-evaluation Addresses the Following Two Questions



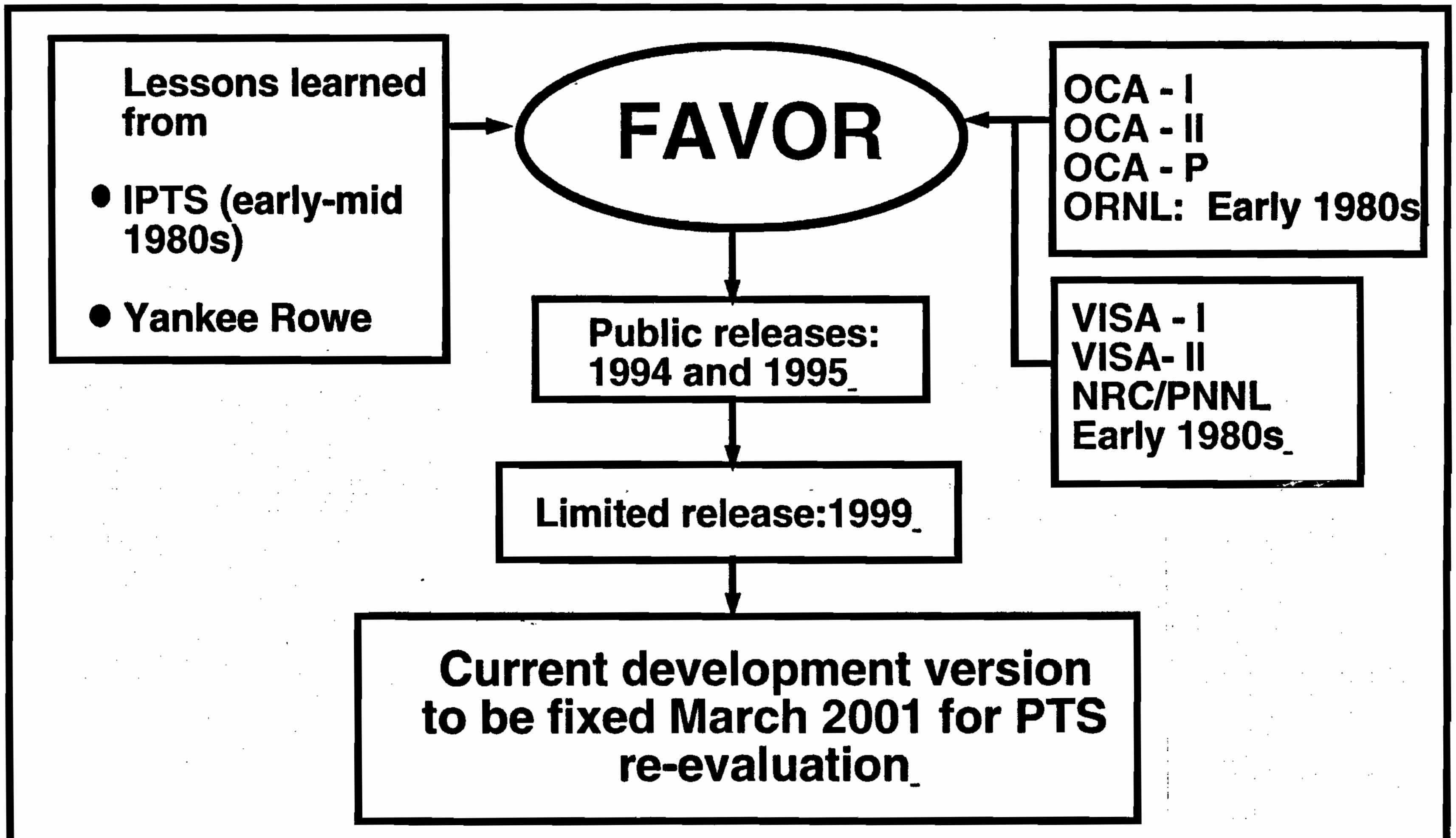
- At what time in operating life does frequency of RPV failure exceed acceptable value (currently 5 x 10e-06)?
- How does integration and application of advanced technology affect the calculated result?

Near-Term Schedule for Development of the FAVOR Code has been Defined

- Current schedule specifies FAVOR to be ready for PTS re-evaluation analyses on March 1, 2001

- In the interim period:
  - models are being finalized
  - finalized models are being implemented
  - scoping studies are being performed

Development of the FAVOR Code was Initiated in Early 1990s by Combining Best Attributes of OCA/VISA with Evolving Technology



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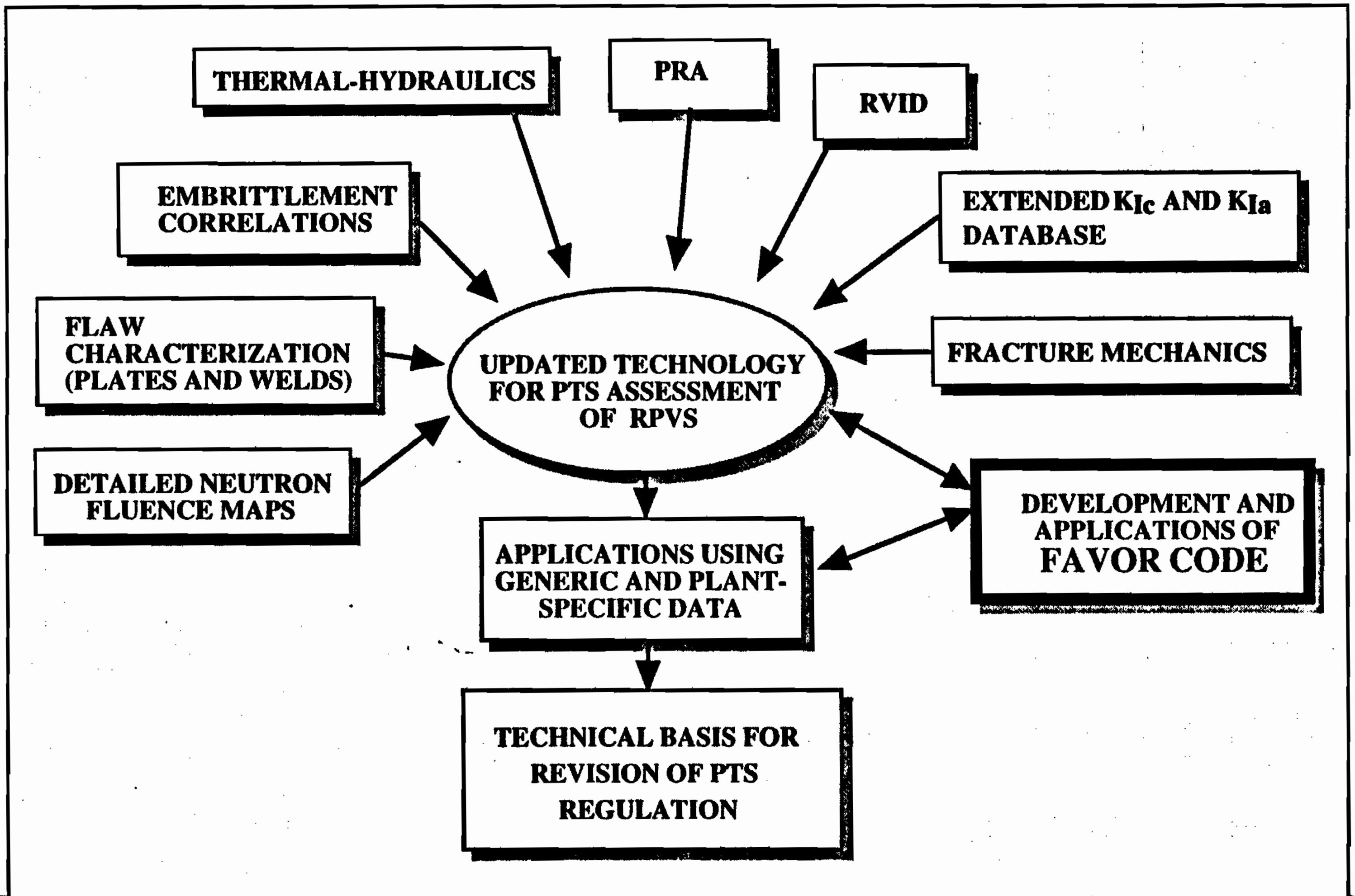
Elements of updated technology are currently being integrated into the FAVOR computer code to re-examine the current PTS regulations*
**(<u>F</u>racture <u>A</u>nalysis of <u>V</u>essels: <u>O</u>ak <u>R</u>idge)*



Advanced Technology is Integrated into FAVOR to Support Possible Revision of PTS Regulation

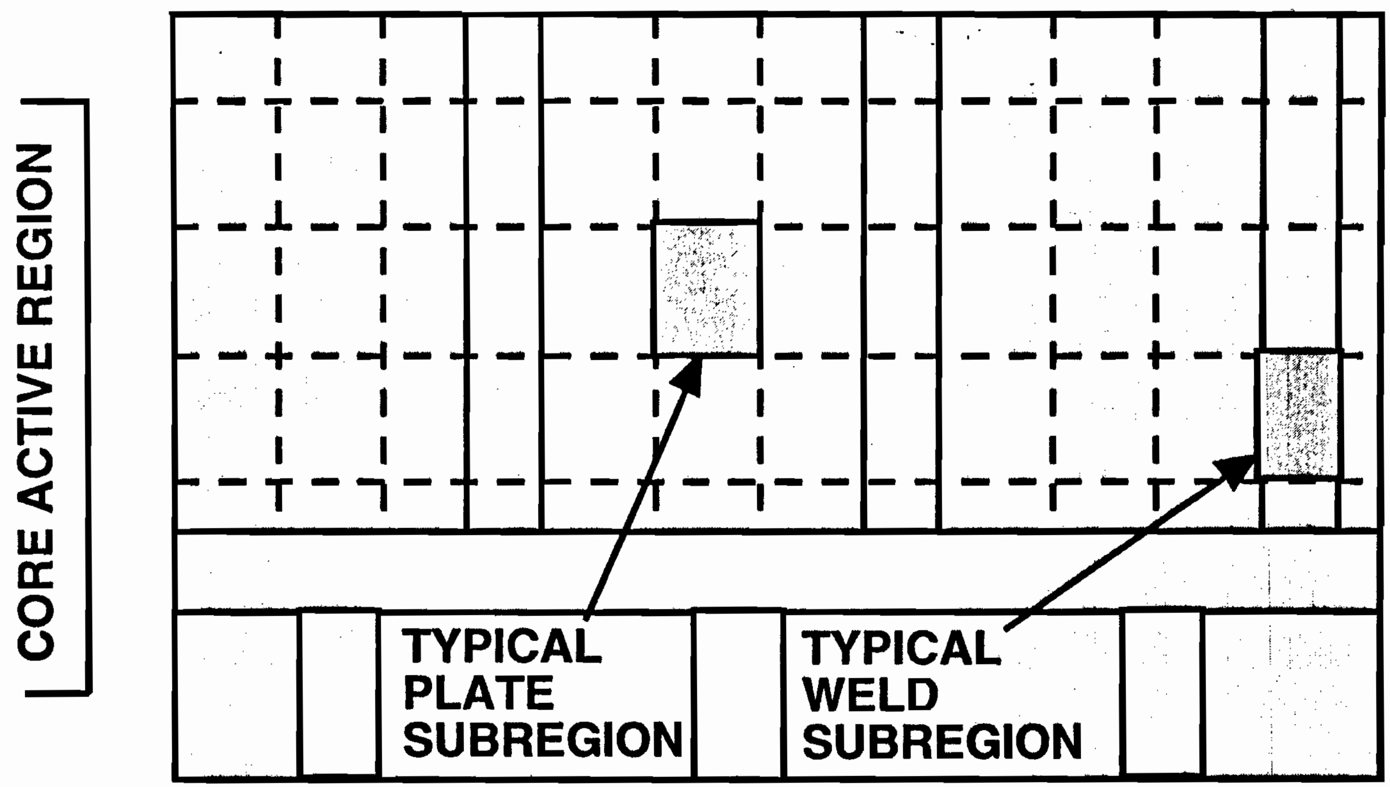
- Flaw characterizations from NRC research (plates and welds)
- Detailed fluence maps
- Embrittlement correlations
- RVID
- Fracture toughness models
- Surface-breaking and <u>embedded</u> flaws
- Inclusion of through-wall weld residual stresses
- New PFM methodology

A Significant Improvement Since the Derivation of Current PTS Regulations is Flaw Characterization*

** analyses assumed all flaws were inner-surface breaking flaws*

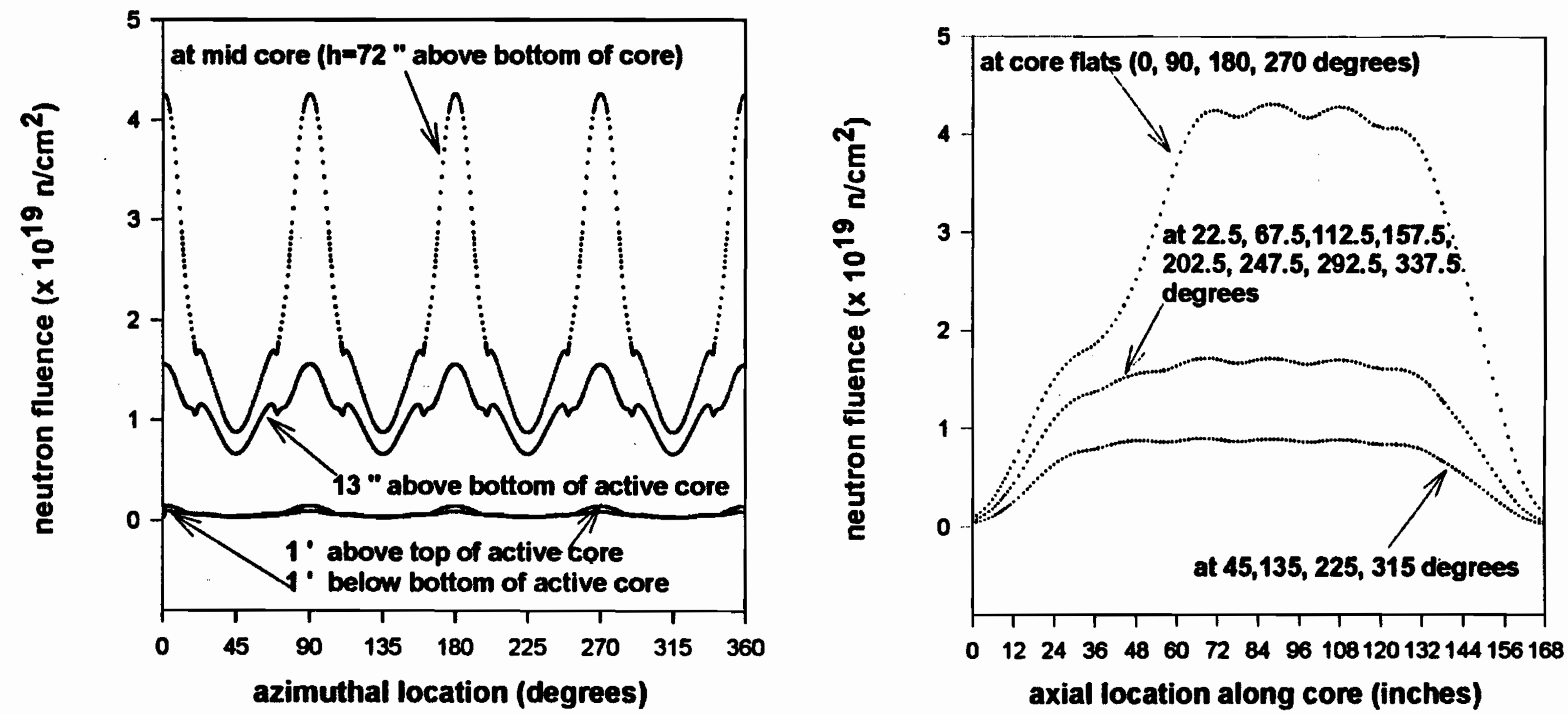
- **Recent NDE and DE of RPV material at PNNL has established an improved technical basis for flaw-related data used as input for PFM analyses**
- **A significantly higher number of flaws were found than was postulated in PFM analyses from which current PTS regulations were derived; however, all flaws detected thus far are embedded**
- **Application of PVRUF flaw densities to commercial PWR results in over 3500 flaws in 1st 3/8 thickness of RPV wall**

FAVOR utilizes a methodology that allows the RPV beltline to be discretized into sub-regions, each with its own distinguishing embrittlement-related parameters. This accommodates chemistries from the RVID and detailed neutron fluence maps

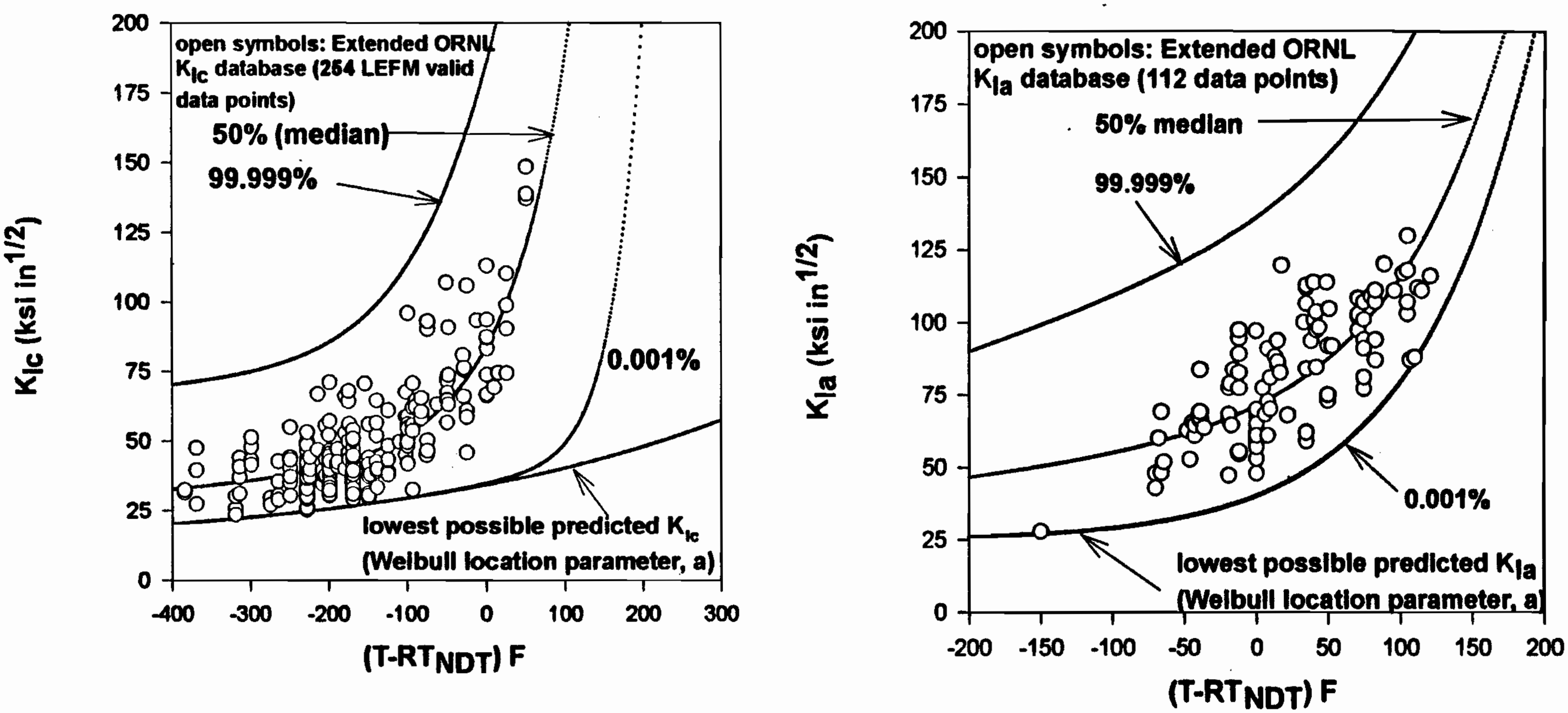


TYPICAL VESSEL BELTLINE LAYOUT

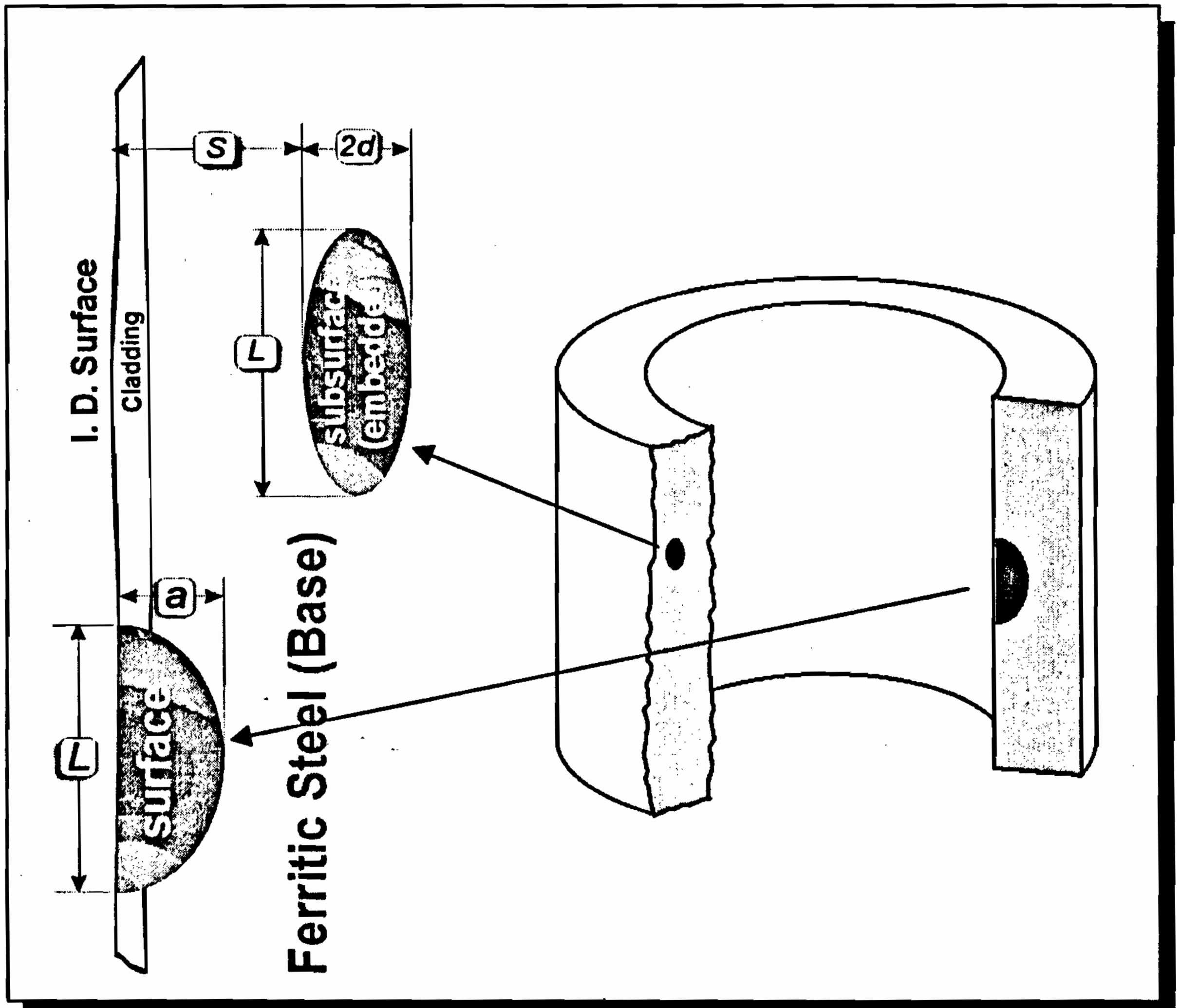
Brookhaven National Laboratory is generating very detailed neutron fluence maps for selected PWRs corresponding to 32 EFPY and 40 EFPY



New Statistical Models for Enhanced Plane-Strain Static Initiation (K_{Ic}) and Arrest (K_{Ia}) Fracture Toughness Data Bases were Implemented into FAVOR



The FAVOR PFM Model Now Includes Inner Surface-Breaking and/or Embedded Flaws



This Presentation Describes the Evolution of an Advanced Computational Tool for RPV Integrity Evaluations : FAVOR

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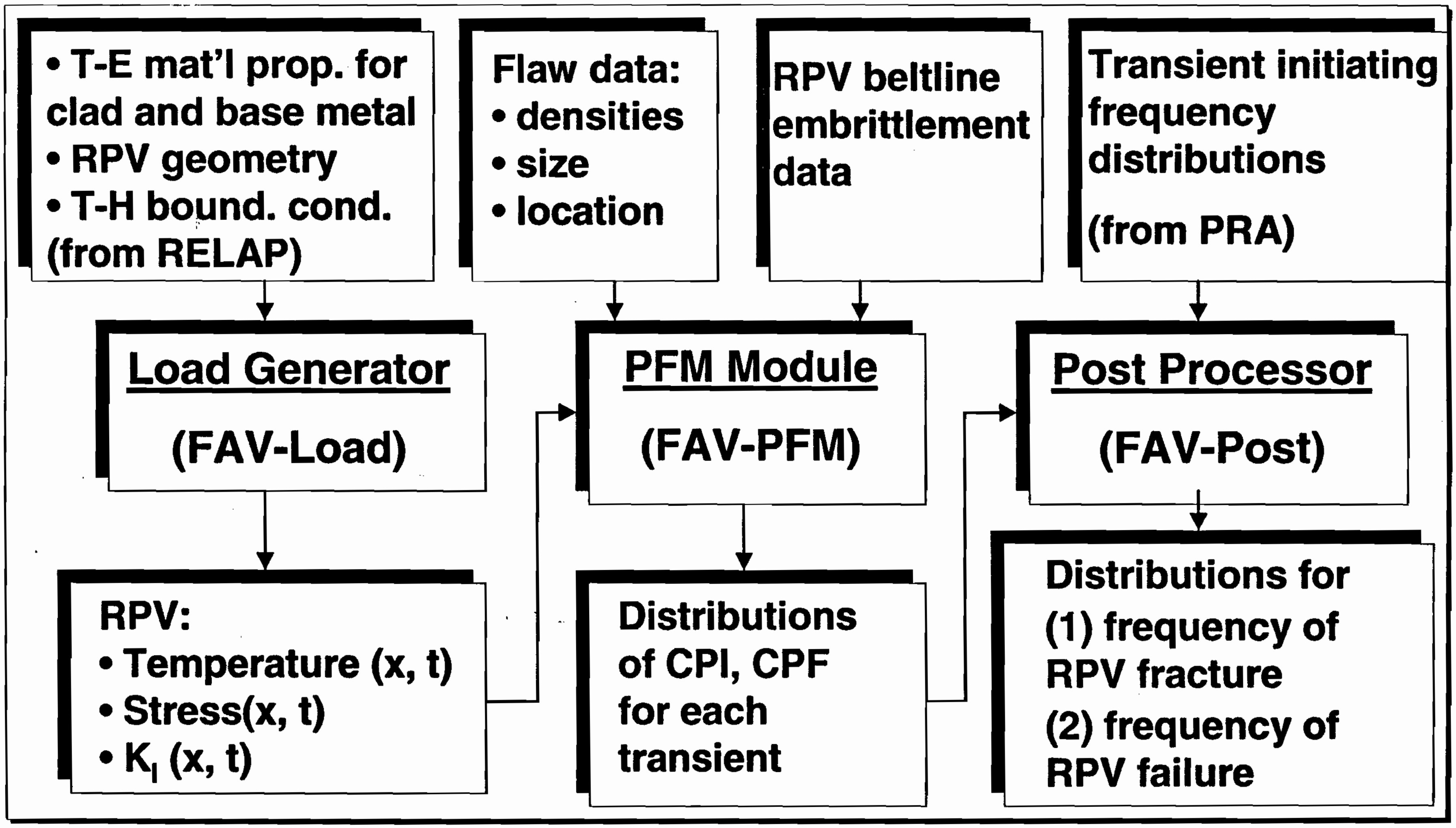
The Current FAVOR Code Consists of Three Separate Modules



This Presentation Describes the Evolution of an Advanced Computational Tool for RPV Integrity Evaluations : FAVOR

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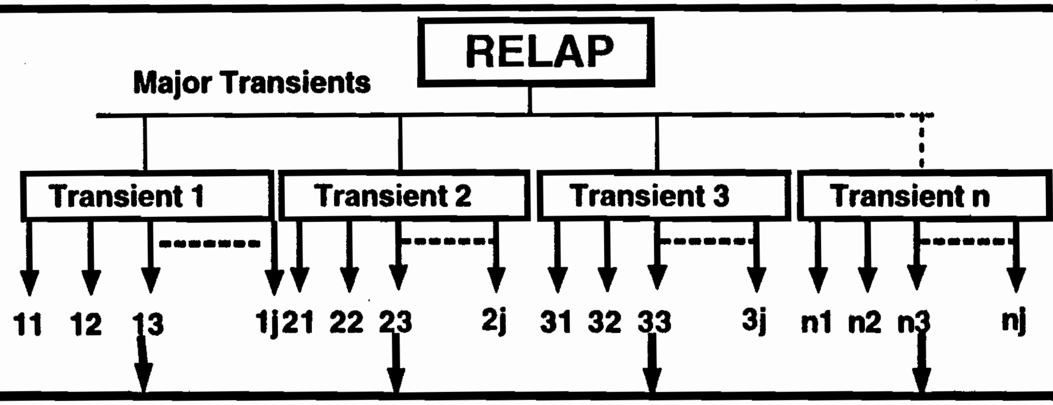
FAVOR Analyses Incorporate Uncertainty Associated with Thermal Hydraulics by Including Variants for Each of the Transients



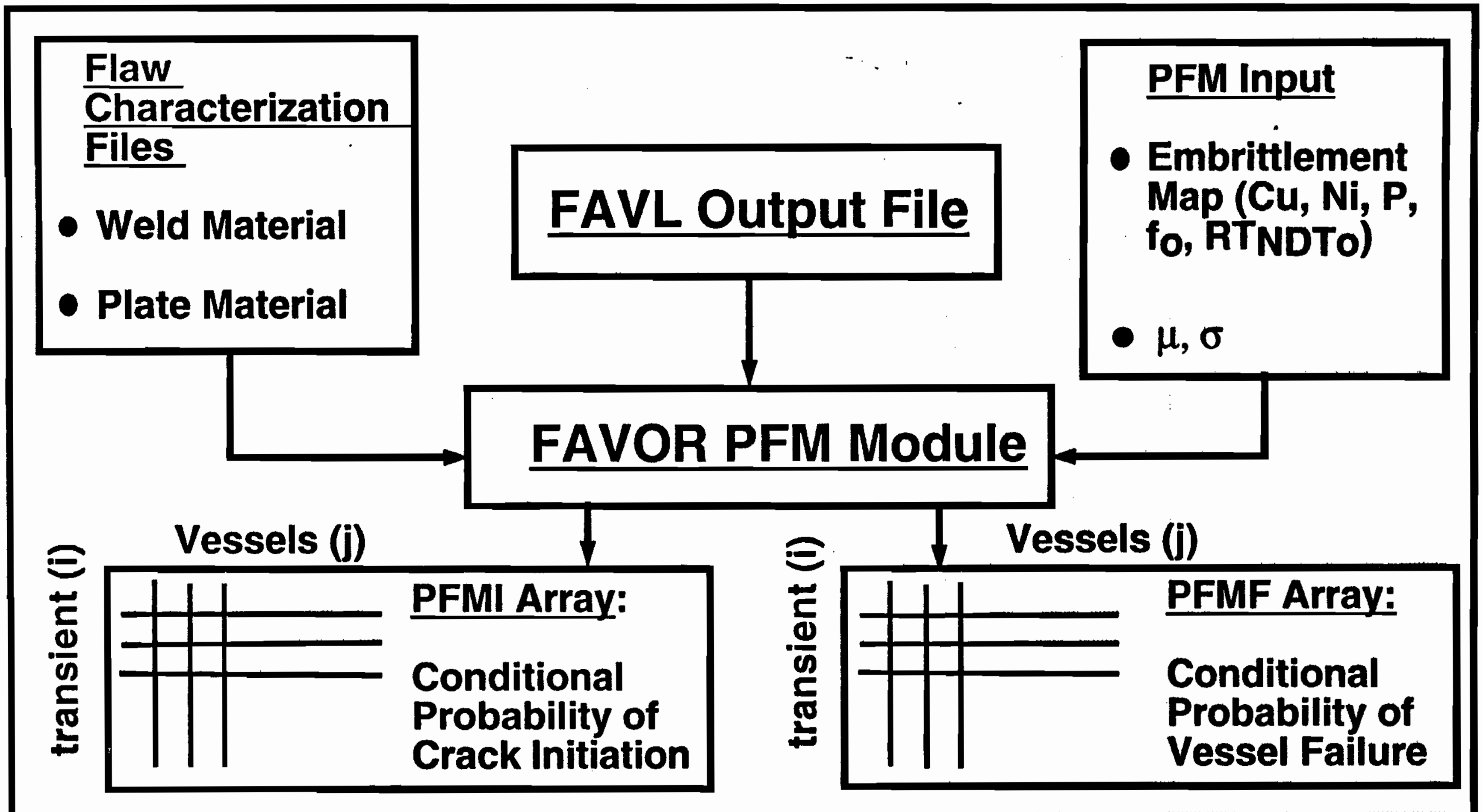
FAVOR Load Generator (FAVL)

One-dimensional axisymmetric finite-element analyses are performed to calculate RPV loads for each transient

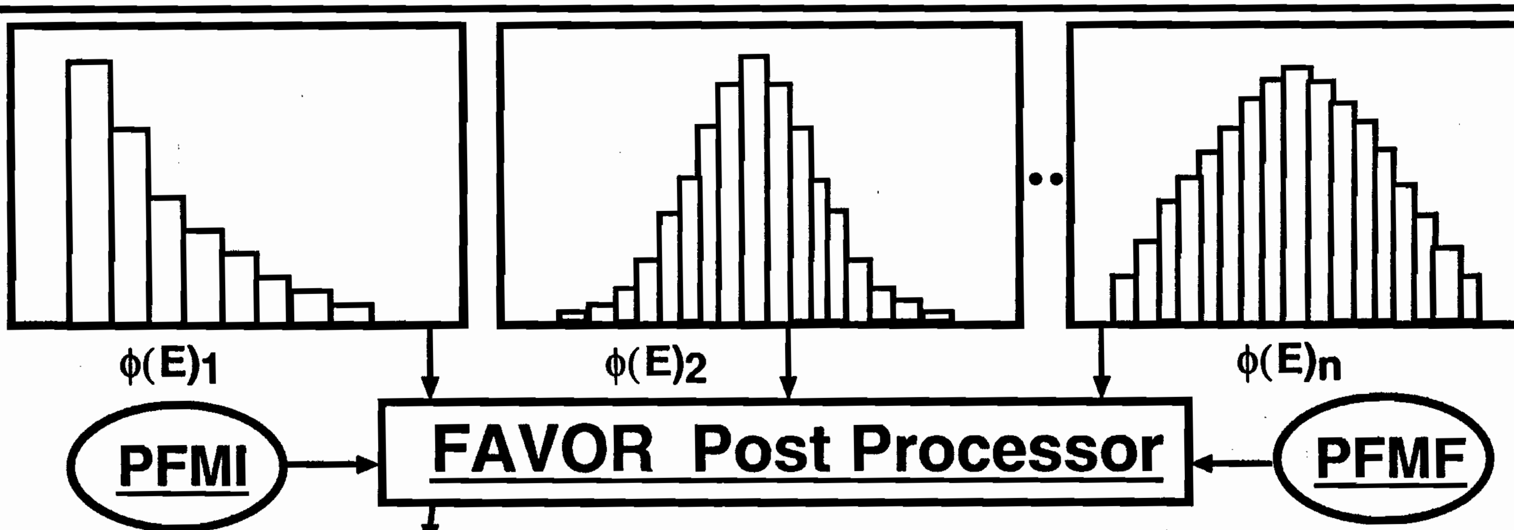
Output File from FAVOR Load Generator

- temperature $T(r,t)$
- circumferential stress $\sigma_H(r, t)$
- axial stress $\sigma_A(r, t)$
- SIF (inner surface-breaking flaws) $K_I(a, l, t)$

The FAVOR PFM Analysis Module Generates Arrays Containing Conditional Probabilities of Initiation (PFMI) and Failure (PFMF) for Vessel(j) Subjected to Transient(i)



The FAVOR Postprocessor Module Integrates the Uncertainties of the Transient Initiating Frequencies with the PFMI and PFMF Arrays to Generate Distributions for the Frequencies of RPV Fracture and RPV Failure



$\phi(F)$ (Frequencies of RPV fracture/failure)

For each vessel:

1. Sample initiating frequencies, $\phi(E)_1, \ldots \phi(E)_n$
2. Combine $\phi(E)$ with PFM results, $\phi(F)_i = \Sigma\phi(E)_T \bullet P_{FM}(T,V)$
3. Generate histogram for $\phi(F)$ from resulting array of $\phi(F)_i$

Near-Term Schedule for Development of the FAVOR Code has been Defined

- Current schedule specifies FAVOR to be ready for PTS re-evaluation analyses on March 1, 2001

- In the interim period:
  - models are being finalized
  - finalized models are being implemented
  - scoping studies are being performed

ITEMS OF INTEREST

476th ACRS MEETING

OCTOBER 5-7, 2000

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
476th MEETING
OCTOBER 5-7, 2000

NRC NEWS

UNITED STATES NUCLEAR REGULATORY COMMISSION

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No. S-00-19 September 13, 2000

[PDF Version (42 KB)]

NRC'S REGULATORY APPROACH: OIG'S ROLE IN A TIME OF CHANGE

Keynote Address
DR. RICHARD A. MESERVE
CHAIRMAN
UNITED STATES NUCLEAR REGULATORY COMMISSION

at the

OIG ANNUAL INFORMATION AND PLANNING CONFERENCE

September 12, 2000

INTRODUCTION

Good morning. Thank you for inviting me to your Annual Planning Conference. I am pleased to have this opportunity to talk with you about the theme for this conference, "NRC's Regulatory Approach," and the important role that OIG can play in assuring the integrity of our implementation of that approach.

As you know, we are in a dynamic period in the NRC, as we move from a prescriptive, deterministic regulatory framework to performance-based rules that are informed by assessments of relative risk. We are early in this transition and we face a daunting task because the transition will involve a fundamental change in our approach. And we must accomplish this evolution without compromising our fundamental mission of protecting the health and safety of the public.

OIG plays an essential part in the NRC's regulatory processes, and OIG's contributions will prove even more important in this time of change. Traditionally, an Inspector General audits agency programs and operations to look for instances of waste, fraud, and abuse, thereby promoting the most effective and efficient use of an agency's resources. This function is fundamentally important because we must be good stewards of the financial resources available to us.

Even more important from my perspective, however, is OIG's role in assuring that the NRC conducts its business according to principles of regulatory best practice. The Commission and its staff must make independent, objective decisions based on technically competent, unbiased assessments. Our decisions must be reached through open processes. And we must conduct inspection and enforcement activities in a manner that is efficient, impartial, and fair. OIG's reviews of our performance against these standards helps to assure that we as an agency are always improving, even as we meet our statutory and regulatory responsibilities. This role is singularly important in this period of transition. I will return to this aspect of OIG's role later in my remarks.

First, though, I would like to provide an overview of the fundamental change in the NRC's regulatory philosophy. I will focus for the most part on power reactor regulation, but we should not lose sight of the fact that our regulatory purview extends well beyond that area and that the change to risk-informed regulation stretches across the entire range of NRC's regulatory responsibilities.

WHERE WE WERE: DETERMINISTIC, PRESCRIPTIVE REGULATION

The foundation of the nuclear reactor regulations was developed, for the most part, in the early days of the civilian use of nuclear power by the Atomic Energy Commission. With little operating experience, the regulatory structure focused to a large extent on plant design, reflecting the perception that a conservative approach to plant engineering would provide large margins of safety. The philosophy of "defense in depth" - many layers of diverse and redundant systems designed to prevent accidents, if possible, and mitigate the consequences of any events that did occur - became a fundamental precept of our regulatory requirements. Assumptions were made about the threats posed by various types of events, such as large-break loss-of-coolant accidents. The regulations resulting from these beliefs and assumptions mandated specific types of analyses and established quantitative acceptance criteria for the results of those deterministic analyses. The acceptance criteria were prescribed so as to ensure, as much as possible, large safety margins.

The goal of this process was to assure plant safety. But our knowledge was not so extensive as to provide a firm understanding of which plant systems and processes were truly significant for safety. As a result, a conservative engineering approach was applied across the board. When a serious event occurred, such as the Browns Ferry fire, the response was to develop additional, prescriptive regulations to deal with the causes and effects of the problem. While this approach was not inappropriate in its time given the state of technical knowledge of these complex systems, it could create two problems. First, it could lead to rules requiring actions that imposed costs that were not always commensurate with the benefits of improved safety. Second, an attitude developed that severe accidents, with consequences beyond those with which the plant was designed to cope, were almost impossible.

The incident at Three Mile Island in 1979 shattered that confidence. The NRC, however, followed previous practice and developed an extensive list of new prescriptive regulations, related largely to the course of events at TMI.

Beginning in the middle 1980s, several factors combined to set the stage for the current change in regulatory approach. First, for several reasons, utilities stopped placing new plant orders, and even canceled previously ordered plants, so that by the late 1980s, design and construction of nuclear power plants was coming to a stop with little prospect for new projects in the near term. As a result, the NRC's focus shifted from design and construction of new plants to safe operation of the existing fleet. Second, based on the growing body of operating experience, both we and the industry began to accumulate important insights to distinguish those aspects of plant design and operation that are truly significant for safety from those that are not. Third, the techniques of quantitative risk assessment improved, and the community became more familiar with these techniques. Fourth, a general recognition arose in academic and government circles, not just in the NRC, that prescriptive, deterministic regulation was economically inefficient and that performance-based regulation could achieve the desired results at lower cost to society as a whole.

WHERE WE ARE: MOVING INTO RISK-INFORMED REGULATION

From much of what has been written and said about risk-informed regulation at the NRC, you might think that this is a relatively recent idea. In fact, the concept has been with us almost since the NRC's creation as regulatory successor to the AEC in 1975. In that year, the Reactor Safety Study, better known as WASH-1400, was published by the NRC. This was the first attempt in the U.S. to apply the technique of quantitative probabilistic risk assessment, PRA, to the evaluation of reactor safety. The report generated a great deal of controversy, particularly over the resulting estimates of risks of nuclear accidents and the associated uncertainties in those estimates. Despite the controversy, however, the potential value of PRA as a tool for gaining insights into reactor safety was widely recognized.

As I noted, the Three Mile Island accident served as a rude awakening for both the industry and the NRC. The accident, however, lent credence to some of the results of WASH-1400, which predicted that events such as the one that occurred at TMI were more likely, and could pose more of a challenge to overall plant safety, than the lower-probability design-basis accidents. Both of the major post-mortem studies of the TMI event -- one sponsored by the NRC, the other commissioned by President Carter -- recommended the use of probabilistic risk assessment where appropriate in helping to focus regulatory attention on risk-significant issues.

The NRC responded, gradually, in a number of ways. For example, the Office of Research funded further development and refinement of risk assessment techniques, and eventually undertook a follow-up study of reactor risk, published as NUREG-1150. The Office of Analysis and Evaluation of Operational Data, established after TMI to help develop regulatory insights from plant operation, used risk assessment techniques to assess the significance of events at operating plants for safety. And the Commission published major policy statements on severe accidents and safety goals based in part on risk insights. In 1988, the NRC requested that operating plants perform assessments of their vulnerabilities to severe accidents, assessments that were called "individual plant examinations." While licensees were not required to use risk assessment techniques in these examinations, their use was strongly encouraged. The increasing familiarity with and confidence in probabilistic risk assessment techniques ultimately led, in the mid-90s, to a determination by the Commission that the agency should begin to evolve toward a risk-informed regulatory approach.

As I indicated previously, and want to emphasize again, "risk-informed" does not mean that risk is the only factor to be used in regulatory decision-making. Rather, it is one of the factors that should be considered in our deliberations. We need to be fully cognizant of the fact that risk assessment techniques are subject to significant limitations, such as in the area of modeling human performance. While risk assessment ideally takes account of uncertainties in our knowledge base, we still employ conservative approaches to account for the practical limitations in the techniques. The concept of defense in depth, with redundancy and diversity in safety systems, and a balance between accident prevention and mitigation of consequences remain central to our regulatory approach. In these aspects, too, risk assessment can help by shining light on areas with the greatest significance for safety.

Currently, we have what effectively is a hybrid regulatory structure. Risk-informed decision-making is employed by the staff in assessing license amendments, in assessing inspection results, and in dealing with specific regulatory requirements, such as technical specifications and in-service inspection. The deterministic foundation still exists, however, and in many areas the body of regulation remains relatively prescriptive. We are now examining how to update these regulations in an evolutionary process. That process is likely to be a long one, requiring continuous adaptation and improvement.

I'd like to take a moment now to speculate on what we may have when that process is complete.

WHERE WE ARE GOING

As you know, the NRC has adopted a strategic plan that articulates four primary objectives: to maintain safety; to improve public confidence; to make our regulatory processes more effective, efficient and realistic; and to reduce unnecessary regulatory burden. Our efforts to risk-inform our regulations need to reflect these objectives.

Our focus must always be on safety, first and foremost. As a result, the risk-informing process will be a two-edged sword. As we apply risk insights to our regulatory structure, we will undoubtedly find rules that are unnecessarily prescriptive and requirements that do not significantly affect plant safety. In these areas, regulatory requirements can be modified, reduced, or made more performance-based and less prescriptive, increasing the efficiency and effectiveness of both licensees and the NRC. However, we may also find areas in which risk-significant systems or processes currently are not adequately addressed by our regulations, and we will need to develop new rules to cover these areas. We must be open to both possibilities and be prepared to act on both.

In addition to the task of looking at individual rules, the risk-informing process gives us the opportunity to take a fresh look at the ways in which our regulations relate to one another. Given the way in which our regulatory structure and processes have evolved since the early days of the AEC, it should not be surprising if we found that the requirements in one rule may overlap with the requirements in other rules. Our efforts should include evaluations of these sorts of regulatory interactions, so that our final products are clear, consistent, and stable. Moreover, as we inform the technical bases of our rules with better understanding of risks, we should look at requirements that are based on technology that has been improved or superseded. If the technology basis cannot be replaced by performance specifications, then at least the technical bases should be updated to reflect the state of the art.

Our oversight process is well on its way to becoming more risk-informed. For example, in addition to inspections, the new reactor oversight process includes the use of objective performance indicators to evaluate plant operations. Findings from inspections and performance indicators are processed to determine their significance for safety, the results of which are then used to guide future oversight activities and, if necessary, enforcement actions. Work is in progress to develop new performance indicators that are clearly focused on risk and are leading indicators of emerging problems.

My vision for the final product of this complex process is a regulatory structure that is more aligned with safety, more internally consistent, and easier for our licensees to understand and our staff to implement. I believe that the overall regulatory burden will, in fact, be reduced without sacrificing safety. I want to emphasize again, however, that while consideration of risk is an important element of the NRC's work, it is not the only factor. Mindful of the limitations of current risk assessment methods, we do not strive for a risk-based regulatory environment. Informed by insights into risk, the concepts of defense in depth and a conservative approach to design and operation will continue to be part of our regulatory paradigm, as long as they are needed to assure safety.

THE ROLE OF OIG

I now want to turn to my view of the role of the Inspector General's office in the process of transition to a new regulatory approach.

As I mentioned at the start of this talk, OIG fulfills two roles. The first, concerned with discovery of instances of waste, fraud, and abuse, is clearly an important one. We cannot and should not tolerate inappropriate conduct by NRC employees. I am convinced, however, that the vast majority of the NRC staff undertake their jobs with the highest regard for personal and professional integrity, and that cases of intentional wrongdoing are few and far between. Nonetheless, we welcome OIG's vigilance.

The second role of OIG is to monitor the performance of our regulatory responsibilities. I want to stress that I do not see this as a means to determine who should be blamed when problems arise. Rather, it is an acknowledgment of the fact that, as humans, we do not always do our jobs as well as we might, that mistakes occasionally are made despite our best efforts, and that constructive analysis can only be beneficial to us all. There will always be room for improving performance. OIG's reviews can provide important insights into ways to improve our regulatory processes, ensuring that they are conducted in accordance with our principles and policies. This willingness to be self-critical is an essential element in improving the confidence of the public, including our stakeholders, in our performance.

Indeed, OIG scrutiny is an aspect of the fact that our regulatory processes must be performed, as much as possible, in the open. Our stakeholders comprise a broad and diverse group, including: the regulated industry; the public, which is often represented by various public-interest groups; the Congress; and the technical community. We must solicit from all input to our processes and carefully consider their views as we carry out our duties. This attention to conducting an open process may be time-consuming, but openness will lead to better decisions and is an essential factor in improving and maintaining public confidence in the NRC. Our encouragement of knowledgeable OIG scrutiny of our activities is part of a philosophy of openness that must be a core NRC operating principle.

CONCLUSION

In conclusion, I want to recognize that the evolution from prescriptive, deterministic standards to risk-informed performance-based regulation will be a challenging one for NRC staff and the regulated industries. Accomplishing this objective will take time and require us to learn new skills and approaches to our work. Complicating the task is the current dynamic environment of the nuclear power industry. While we cannot predict how that industry will change over the next decade in response to the economic deregulation, we must be adaptable to whatever changes occur. We cannot, however, modify our mission or slight our principles. We depend on OIG to inform us on whether we are discharging our responsibilities correctly and to give us guidance on how continuously to improve.

Once again, thank you for the opportunity to share my views with you. I look forward to working with you as we strive to meet these important challenges.

Thank you.

NRC NEWS

UNITED STATES NUCLEAR REGULATORY COMMISSION

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No. S-00-18 August 31, 2000

[PDF Version (176 KB)]

The Honorable Greta Joy Dicus
Commissioner
U.S. Nuclear Regulatory Commission

"Radiological Emergency Planning"

at the

Harvard School of Public Health

August 23, 2000

Good morning ladies and gentlemen. I am very pleased to be here with you today to provide some of my perspectives on current radiological emergency planning (REP) issues --- and given that I was a former student at the Harvard School of Public Health I think I find it much more comforting to be the lecturer rather than the lectured. Did I mention there would be a test at the conclusion of my session?

Back in 1997, I was here to speak on emergency planning (EP) for this course and I'm glad to report that since then the NRC, our other government partners (FEMA, State, and Local governments), and our private sector partners (utilities and industry representatives) have been able to work out improvements in this area and have identified many more opportunities to focus resources in the right areas to make EP more effective, as well as, to inject the risk-informed perspective/mindset into EP. Today, I would like to provide my perspectives as a Commissioner and spend some time discussing the concept of lead Federal Agency and the NRC's incident response focus, our participation in incident response exercises, the NRC's State Outreach program, realism in scenarios, recent events related to EP and some new initiatives in EP such as, the One Voice initiative, the implications of decommissioning, the new reactor oversight program and EP, and the use of potassium iodide in EP.

I have been directly involved in EP for the past 15 years; therefore, my comments are both a product of practical experience as a responder, planner, and a policy maker. Over the past four years as a Commissioner I have gained many insights regarding the issues associated with -- and the importance of EP from a national policy perspective. And I can tell you that because of my unique opportunity to participate in EP activities for the Arkansas Nuclear One power plant as a State regulator and now as a

Federal regulator, I have a great interest in involving ALL stakeholders in the Commission decision making process related to EP.

NRC RESPONSE AND PARTICIPATION IN EXERCISES

Lead Federal Agency Concept

The nature of the emergency, the licensee, the materials involved, and the facilities involved are determinants utilized in the Federal Radiological Emergency Response Plan to designate the Lead Federal Agency (LFA). The LFA is responsible for leading and coordinating all aspects of the Federal response. The LFA is responsible for providing information on the status of the overall Federal response, specific LFA response activities, and the status of onsite conditions. In situations where a Federal agency owns, authorizes, regulates, or is otherwise deemed responsible for the facility or radiological activity causing the emergency and has authority to conduct and manage Federal onsite actions, that agency normally will be the LFA. Generally, the LFA is expected to: (1) ensure that State's needs are addressed, for example, if requested by the State or local authorities, the LFA may advise them on protective actions for the public; (2) the LFA approves the release of official Federal offsite monitoring data and assessments; (3) the LFA provides other available radiological monitoring data to the State and to the Federal Radiological Monitoring and Assessment Center. There are five federal agencies that could be designated as the LFA in response to an incident originating from the use of nuclear materials; however, the decision is generally based on the organization's normal responsibilities.

The NRC is the LFA for any emergency at a nuclear facility licensed by the NRC or an Agreement State or any emergency involving radioactive materials licensed by the NRC or an Agreement State. The Department of Energy (DOE) is the LFA for any emergency at one of its facilities or involving transportation of DOE materials. The Department of Defense (DOD) is the LFA for any emergency at one of its facilities or involving transportation of DOD materials. NASA is the LFA for emergencies involving domestic satellites that involve NASA space missions. DOD is the LFA for domestic satellites that involve DOD space missions. The EPA is the LFA for those emergencies at a nuclear facility not licensed, owned, or operated by a Federal Agency or an Agreement State or for those emergencies involving materials not licensed or owned by a Federal Agency or an Agreement State. EPA is also the LFA for emergencies resulting from a foreign or unknown source. For example, if a radioactive source is found -- something that we have too much experience with I'm afraid, and ownership is not readily known the EPA would assume the LFA role. In the event of a significant foreign event, the EPA would monitor such an event with the focus of protecting the health and safety of United States citizens. For emergencies other than these, the Federal Agencies would confer to determine the LFA for that particular event. In all of these situations, the Federal Emergency Management Agency (FEMA) is the coordinating Agency.

NRC Role in Incident Response

Next, I would like to speak to the responsibilities that the Commission faces in its incident response role. The NRC Chairman (and any of the other Commissioners can be delegated this responsibility) is the senior NRC authority for all NRC response activities and is the Director of the NRC's executive team during event response. The Chairman may act alone or on behalf of the Commission in an emergency and the Chairman is responsible to the President for all agency actions.

The NRC's executive team maintains an oversight role during event response and wants to maintain the broad picture of the event assessment and progression --- and in doing so is generally concerned with determining the status in five key areas.

The first question we want to answer is obvious -- how serious is the accident? In making this assessment for a nuclear facility we gather information through our resident staff, automated information systems, and the licensee to determine what has occurred, if there were radiological or chemical releases, when the threat of release is expected to end, and the relative severity of damage if it has occurred. Next, we are interested in knowing how effective the licensee's response is. In reaching this assessment the NRC verifies if the event was classified correctly, if the State and local officials have been notified, what recovery actions have been identified, and what protective actions have been recommended. Third, we are interested in the State's response. So, we are looking to see if the State has issued any protective action decisions, has the implementation of protective actions been effective, and how many people were affected in issuance of the protective actions. We then want to determine the status of the NRC and coordinated Federal response. In making this assessment, we consider the NRC response mode, the status of the NRC site team and interactions with other federal agencies such as the FEMA, the Department of Energy, the Environmental Protection Agency, the Department of Agriculture, the Health and Human Services Administration, and the National Oceanic and Atmospheric Administration depending on the nature of the incident we are responding to. And lastly, we want to determine how information has been disseminated to the public. In making this determination we want to know if press releases have been issued by the NRC, State, local governments, or the licensee, whether the joint information center has been established, and whether the NRC News Center has been activated. I'm sure at times licensees have experienced frustration with the NRC's requests during incidents, and I would encourage active dialogue to ensure our interaction is at the appropriate level, but I would hope that licensees, State, and local governments also see the value in helping us meet our mission while enhancing public confidence in our ability to regulate the nuclear industry in support of the Congressional policy that nuclear-generated power will be part of our energy mix.

NRC Participation in Exercises

Next, I would like to review NRC participation in exercises. I have been and remain a proponent of significant involvement of Federal agencies in exercises. A number of years ago the NRC revised the exercise rule that eliminated the requirement for the "off year" or annual onsite exercise. While this reduced the required frequency for exercising the licensee's onsite emergency plan from annual to biennial performance, it preserved the requirement for the biennial full participation exercise. This rule requires licensees to ensure that emergency response capabilities are maintained between exercises by conducting drills, at least one of which must involve some of the principal functional areas of onsite capabilities. The rule also requires licensees to continue giving those State and local governments that are in the plume exposure pathway the opportunity to participate in these drills.

With respect to the biennial drill requirement, I would also mention that FEMA has initiated efforts to provide additional flexibility to offsite authorities to improve, streamline, and enhance the efficiency and effectiveness of their emergency preparedness programs by providing an option of foregoing one of the biennial exercises in a 6-year cycle and allow the demonstration of reasonable assurance by alternative means. The NRC is assisting FEMA by initiating efforts to change emergency preparedness regulatory requirements to accommodate this initiative. Right now the thinking is that the alternative demonstrations could be such activities as FEMA evaluated radiological focus drills, functional drills involving some of the key areas of offsite response, and a post plume phase only (ingestion pathway) exercise. As part of the change, offsite authorities would be required to negotiate with FEMA the alternative means to demonstrate reasonable assurance in the biennial period in which the exercise is not conducted. These efforts were one of the outcomes of a FEMA initiated strategic review of its radiological emergency preparedness program (REP) for commercial nuclear power plants which began in 1996.

While exercises currently will only be evaluated every other year at a site, the NRC remains committed to conduct and participate in emergency exercises. Because of the year 2000 concerns with computer software -- 1999 was a very busy year for the NRC in that we participated in a larger than normal number of exercises -- including a few unique exercises conducted in part so that we could verify our readiness to deal with the possibility of concurrent events stemming from the Y2K problems as we entered the 21^{st} century.

In 1999, the NRC participated in five full scale exercises (Dresden, Limerick, San Onofre, and HB Robinson, Y2K Preparedness), five regional-based exercises with three reactor licensees and two fuel facilities, one table top on Y2K readiness (with participation by Calvert Cliffs), and five ingestion team exercises. In 2000, NRC headquarters has participated in two reactor exercises and has one more is scheduled in October. In addition, NRC headquarters also recently participated in a first of a kind exercise at Nuclear Fuel Services fuel facility in Tennessee in cooperation with the FBI.

Finally, I would like to leave you with an opinion that I shared at a recent NRC Y2K table top exercise. I'm sure that those of you who have been intimately involved with the issue have found it to be a challenge. But, I believe, it was also an opportunity. On the Federal level, the coordination and cooperation between Federal agencies on the Y2K issue are a foundation upon which the Federal government is building for future cooperative efforts. Much of the effort being spent on the Y2K problem will help Federal agencies better respond to emerging unconventional threats to the United States, such as terrorist acts. The NRC has purchased satellite phones for all of our nuclear power plants as part of our Y2K contingency plan, and many utilities are also investing in upgraded communications systems. As a result, if a tornado were to destroy the commercial telephone lines into a site, as well as our own direct access lines, as it did last summer during a tornado at Davis Besse, we will still be assured of communications with the site. These are just a few examples of how the Y2K effort will pay off long after we stand down from the increased staffing from our operation centers on New Year's Day.

Lost Source Exercise

In 1996, the NRC's Office for Analysis of Operational Data reported (these numbers are for reportable incidents) 130 incidents where there was a loss of control of NRC licensed material, and 133 similar incidents of agreement-state licensed material. More recent data for 1999 was that there were 98 incidents regarding loss of control for NRC licensed material and 116 for agreement-state licensed material. Up to 1996, exercises for these types of incidents had not been conducted, although there had been many incident responses to real incidents, so you can see why the idea for conducting an exercise for recovery of a lost source began to take hold. As a result, in September and October of 1997 the EPA acting as the lead federal agency, NRC Region I, and two States conducted two lost source exercises. These exercises were conducted to demonstrate an emergency response and source recovery operation involving the private sector, county government and State government, and utilizing federal assistance from multiple federal agencies. The results of the exercise are documented in NUREG-1634. After reviewing the report I saw several statements made by participants which confirmed my belief that realism of scenarios is all too important to ensure an effective incident response program in that many participants stated they learned how to deal with managing isolation and recovery of the source which could not have been achievable through class room exercises or walk thoroughs. More recently, in 1999 another lost source exercise involving a scrap yard was conducted in North Carolina. All of these exercises involved coordination among agencies that had not occurred previously.

It is also important to note that even in the years since this first exercise, there are still incidents involving lost sources including over exposures, and while we gained valuable experience from these exercises, because of issues like turnover of emergency response personnel, we still need to keep planning and drilling and looking for opportunities to ensure we maintain effective incident response programs at the local through Federal level.

Outreach and Training

The NRC maintains a comprehensive State Outreach Program for incident response which provides information, training, and opportunities to exchange ideas between State, utility, and Federal agency representatives.

This program was initiated a number of years ago to support an effective incident response program to improve the States' understanding of how the NRC, as a lead federal agency, will coordinate the Federal response to a severe accident at a nuclear facility. In managing this program the NRC annually conducts training with State, utility, and other Federal emergency responders ---- and when scheduling permits attempts to schedule the training close to scheduled site emergency exercises to reenforce and solidify the experience gained. A secondary objective remains to improve and enhance the working relationships among emergency responders to power plant accidents and provide an opportunity for State representatives to develop a greater understanding of the resources available from the Federal government to assist a State. Through this program many States have obtained copies of the NRC's dose assessment software and are also using it for their decision making.

In 1999 emergency response representatives from 15 States, including local authorities and utility staff, participated in training as part of this program. In 2000 the NRC is slated to conduct outreach with representatives from more than 20 States. Right now I'm told there are plans to conduct the next session in Florida around the 20th of September. Dates for future sessions can be obtained through the NRC web site. Additionally, in part due to the realization that there is always going to be turnover in the emergency responder organizations, it is my understanding that the NRC will be continuing this program --- and based on my experience working for the State of Arkansas I can attest to the value this kind of effort provides local and State emergency responders.

Realism in Scenarios

Next, I would like to talk about something I have labored over for several years. Realism in Scenarios.

Over the years that I worked with the emergency response efforts of Arkansas, I came to understand the sequences of events that were necessary to drive an emergency response exercise at a Nuclear Power Plant to the General Emergency classification. These sequences were all too often as extraordinary as they were predictable. However, while my perceptions regarding emergency response scenarios have remained unchanged over the past few years, as a Commissioner, I have gained a greater appreciation regarding the limitations facing drill conductors.

Never-the-less, I still believe that in order to achieve realism in scenarios, realizing there is normally only a short amount of time to run through the whole scenario, it is important to clearly articulate the specific objectives to be accomplished during the exercise. One way to improve realism would be to focus on a smaller number of objectives. However, regardless of the scope of the exercise it is equally important to rigorously test those objectives and make them as realistic as possible. Exercising the radiological emergency response plan is a time and resource consuming activity for all organizations involved. It will not be a small task to improve the utility of these exercises, factor in potential opportunities for success paths, still meet exercise objectives, and observe increasing budget limitations. But I believe this is preferable to ensure emergency responders do not become complacent, question the utility of their efforts, and potentially experience "negative training" as a result of their participation. I strongly encourage the scenario developers to continue to strive for quality in their efforts and the overall safety objective of preparing emergency responders for a, however unlikely but, potential event at a nuclear power plant.

Indian Point 2

As I'm sure many of you must be aware, in February this year an Alert was declared as a result of a steam generator tube rupture in one of the steam generators at the Indian Point 2 nuclear power plant in New York State. Now this incident has generated a significant level of press coverage and the NRC and the utility, Con Ed, have been interacting very frequently regarding the causes for the tube rupture and the emergency preparedness weaknesses that were manifested during the licensee's response to the event. A significant result of this event has been larger than normal congressional interest. As a result, the Commission has received numerous requests from local, State, and Federal officials to order the plant to remain shut down until its steam generators are replaced. And perhaps a part of this larger than normal interest stems from an NRC internal assessment regarding our approval of tube inspection extension request for the steam generator that experienced the tube rupture. However, I think that the emergency preparedness lesson learned from this event is that conducting exercises which rigorously test drill objectives is proven and effective way to discover if there are areas in need of improvements, and we should strive to diversify our objectives to the extent possible so that we can ensure all aspects of the emergency preparedness programs are throughly tested.

Japanese Fuel Facility Criticality Accident

As you know in September 1999, a criticality accident occurred at the Japanese Tokai-Mura fuel cycle facility. As a result of the Tokai-Mura criticality accident, the President requested the NRC to conduct a review of U.S. commercial nuclear fuel cycle facilities, to ensure that a similar accident could not occur. The NRC has spent a significant amount of resources studying the accident to determine if there were any lessons learned for our fuel facilities and issued its final assessment in April this year. First off, I would just mention cooperation from the Japanese authorities was instrumental to us in completing our assessment. Essentially, the accident occurred because technicians at the plant achieved criticality while working with highly enriched U-235 in an unfavorable geometry. While the NRC staff has concluded that the possibility for a similar event would not be likely in the U.S. because of the regulatory measures established for U.S. fuel facilities, I think there were still some very important emergency preparedness lessons to be learned.

Following the accident the local mayor had to make the hard choice of issuing evacuation orders without the benefit of government guidance or advice. Adding to the resulting confusion from the accident, was the fact that local authorities issued their protective action orders about four hours after being notified by the company of the accident. The company also took a while to notify the government of the accident, and didn't warn local emergency responders that they were facing a criticality accident. The National Emergency Preparedness for Nuclear Disaster Law in Japan did not include fuel fabrication facilities. Thus, the plant did not have plans regarding communication of general information to the public or emergency responders. Approximately 310,000 people were ordered to remain inside their homes and everyone living within 350 meters were evacuated. And while, the IAEA fact finding mission concluded that the accident did not involve widespread contamination of the environment and that there was little risk off site once the accident was brought under control, the public perception will be a difficult obstacle to overcome for the Japanese. As a result of these difficulties in communicating effectively with the public, the incident got a lot more publicity than expected. TMI taught us a lot about communicating with the public and this recent event shows how important working out the details ahead of time for communicating with the public is essential to maintaining public confidence.

NEW INITIATIVES IN EP

ONE Voice Initiative

Based upon lessons learned during the Y2K rollover and the Tokai-Mura criticality accident in Japan, the NRC initiated a plan to begin discussions to enhance communication and coordination among the 17 member Federal agencies of the Federal Radiological Preparedness Coordinating Committee (FRPCC) so that the Federal government speaks in a consistent manner following radiological events AND efficiently and effectively disseminates information between Federal agencies regarding these events, especially those occurring in a foreign country. The information is very important because quite frequently, following an international event, the NRC or DOE would be very interested in determining if there were any implications for related U.S. facilities. In approving the plan, the Commission emphasized its belief that the Federal Government needs to speak with "one voice" during international nuclear related emergencies. Initially under the plan, discussions were to occur among the FRPCC concerning improvements in communications and coordination among Federal agencies in responding to peacetime radiological emergencies under the Federal Radiological Emergency Response Plan. Additionally, the initiative will address a broad range of alternatives such as: 1) decentralizing the approach in which each agency responds to inquiries using a common base of information; 2) centralizing the approach in which the Lead Federal Agency is responsible for all external communications; 3) developing an approach in which the White House is responsible for all external communications; 4) establishing an approach in which the FRPCC itself is responsible for all external communications; or 5) establishing a graded approach where responsibility for communication would change as the scope or intensity of the emergency situation changes or as public concerns escalate. Under this initiative it would be desirable that the FRPCC seek routine involvement by a White House agency in its activities and in individual agencies' emergency exercises when the scenario, if real, likely would draw significant media attention. While the NRC's mission is clearly focused on oversight of U.S. nuclear materials and their uses, it is important to remember that international events have a direct reflection on public confidence here; therefore, I would underscore the importance of our involvement in international activities -- which contributes to enhance the public's perception by ensuring accurate and consistent information on foreign events is disseminated.

Effects on Decommissioning on Emergency Planning Requirements

As more licensee's have decommissioned their nuclear power plants the realization that changes in the emergency preparedness requirements has become more evident . In this regard it has been recognized for some time that EP regulatory requirements do not take into consideration the risk reductions over time for permanently shutdown nuclear power plants. In the past any relief for a decommissioned plant from EP regulatory requirement has been obtained on a case-by-case basis through the exemption process. We know that after a reactor is permanently defueled, the traditional accidents that dominate operating plant risk are no longer applicable. During decommissioning the primary safety concern involves the fuel stored in the spent fuel pool. In particular, if the spent fuel pool water is lost and there is sufficient decay heat, the fuel rods could heat up to where the oxidation of the zirconium fuel cladding becomes self-sustaining and leads to a zirconium fire. Under this scenario, if the accident progressed this far, there could be an offsite release which might lead to offsite evacuation. Because of the potential for the zirconium fire, it is believed that EP requirements are still required until the possibility of such an accident is sufficiently low. The NRC staff has studied this issue and generally concluded after a period of about five years this scenario would be unlikely; however, after one year (because of decay heat of fuel) there are sufficient bases for measured changes to the emergency plans.

For this reason the NRC has been working towards developing a framework which includes a risk-informed integrated rulemaking plan to better focus the requirements associated with a decommissioned nuclear facility and thereby reduce unnecessary regulatory burden -- which has an additional outcome in that it allows for more effective and focused use of NRC and licensee resources in those areas warranting attention. This plan specifically outlines how to make EP regulatory requirements consistent with plant status. Earlier this summer the NRC staff provided the Commission with its recommendations for a generic approach for reducing the unnecessary regulatory burden associated with EP requirements which were developed for operating reactors. I clearly support this effort and have expressed my desire to the staff to aggressively pursue this effort; however, if that is to occur it is important for industry to work with the NRC to support this goal. Therefore, I am looking forward to later this year when the NRC staff forwards further recommendations to the Commission which specifically address comments on the plan from the nuclear industry.

Reactor Oversight Program

Many of you I'm sure have heard about changes to the NRC's reactor oversight program over the past few years these changes have lead to the creation of performance indicators as an additional tool for measuring licensee performance and allow for more effective and efficient use of NRC and licensee resources. And while I know that Mr. Miller will be covering the performance indicators for emergency preparedness among other insightful topics, I would be remiss if I didn't offer my opinions and thoughts on these changes.

Emergency Preparedness is the final barrier in the defense in depth approach to safety that NRC regulations provide for ensuring the adequate protection of the public health and safety. Emergency Preparedness is a fundamental cornerstone of the Reactor Safety Strategic Performance Area. The objective of this cornerstone is to ensure that actions taken by the emergency plan would provide adequate protection of the public health and safety and the environment during a radiological emergency. And while I am very supportive of this program, I also believe that we should periodically evaluate the appropriateness of the performance band thresholds for the cornerstone as future risk insights are developed. As the NRC and industry moves towards implementation of a more performance-oriented assessment framework, the success of the process is dependent upon the licensees' implementation of its programs, and the NRC's verification of the licensees's performance. Therefore, I feel it is imperative that the NRC utilize appropriate inspection and evaluation resources and expertise to properly verify and assess by independent means, the effectiveness of licensee performance in this area.

Potassium Iodide - KI

In June of last year, the NRC issued a proposed rule, on the consideration of KI in emergency planning, to revise the emergency planning regulations to require that the use of KI be considered as a protective measure for the public as a supplement to evacuation and sheltering as appropriate. If you will recall the administration of KI before or very soon after inhaling or ingesting radioiodine will greatly reduce the uptake of radioiodine by the thyroids of children as will as of adults, thus reducing the thyroid dose and the subsequent risk of thyroid cancer and other thyroid diseases. The final rule package is currently being considered by the Commission. The NRC staff is also working with FEMA and other Federal agencies on a revision of the KI Federal policy. The NRC also plans to issue a revised draft guidance document on the use of KI following reevaluation by the FDA of its 1982 guidance on exposure action levels and proper dosage of KI. Also before the Commission is a NRC staff recommendation that KI be distributed through the National Pharmaceutical Stockpile (NPS).

Until, very recently I was the only Commissioner that had voted on these issues. First off, I want to reiterate my belief that evacuation provides the best protection to the public to a large release of radioactive material; however, being a former State emergency responder, I realize that things don't always go as planned and the extra measure of protection from the use of KI provides supplemental protection - defense in depth if you will. Sometime after I had voted to approve the staff's recommendation to use the NPS to distribute KI following emergencies, FEMA's director, Mr. James Witt expressed his disagreement with using the NPS in a letter to the Chairman in June of this year. One other issue that remains part of this debate is funding. That is, who will fund the stockpiling of KI.

Obviously, there is going to be a lot more work before the NRC, FEMA, and the States agree on the use of and the appropriate distribution mechanism for KI. I am optimistic we will get there -- even the Congress is beginning to become more active in this issue as evident by the recent statements made by representative Phil English (R-PA) that he plans to introduce legislation to require the FEMA to develop a plan - in cooperation with NRC - for stockpiling potassium iodide tablets within a 50-mile radius of a nuclear power plant.

CONCLUSIONS

Although much has been done to address the emergency preparedness issues that confront the nuclear industry, we need to continue to look ahead to ensure that regulatory framework reflects the challenges we face regarding our changing missions and budget as well as the economic pressures being faced by the nuclear industry. And we must not lose sight of the fact that while the regulatory requirements are long established in this area we must continually reflect on what we could do better -- work on maintaining through drills the cooperation among emergency responders to ensure that public safety is maintained -- develop realistic scenarios -- work on ensuring our communications with the pubic is effective. The benefits for doing so are enhanced public confidence which is worth every bit of the effort. Effective incident response also helps ensure that nuclear energy remains a viable alternative for this Nation as directed by Congressional policy.

In my opinion, the NRC can help with the efforts to maintain a workable framework for emergency preparedness through our regulatory efforts, and achieve a high degree of credibility demanded by the pubic in arriving at its decisions in a fair and open process.

Thank you for your attention, I would be pleased to answer any questions you might have at this time.

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NRC NEWS

UNITED STATES NUCLEAR REGULATORY COMMISSION

OFFICE OF PUBLIC AFFAIRS, REGION I

475 Allendale Road, King of Prussia, Pa. 19406

No. I-00-65 August 31, 2000

CONTACT: Diane Screnci, (610)337-5330/ e-mail: dps@nrc.gov
Neil A. Sheehan, (610)337-5331/e-mail: nas@nrc.gov

NRC ISSUES INDIAN POINT 2 STEAM GENERATOR INSPECTION REPORT

The Nuclear Regulatory Commission has issued a report to Consolidated Edison Company of New York detailing the findings of a special inspection that reviewed the cause of the February 15th steam generator tube failure at the Indian Point 2 nuclear power plant. The team inspection was conducted from March 7 through July 20 at the Buchanan, N.Y., facility and focused on Con Ed's performance during its 1997 inspection of the plant's four steam generators.

The NRC team has preliminarily concluded that the overall direction and execution of the 1997 steam generator in-service examinations were deficient in several respects. Deficiencies in the steam generator inspection program resulted in the company's failure to adequately account for conditions which adversely affected the detectability of, and increased the susceptibility to, tube flaws. The team concluded that these failures resulted in tubes with flaws being left in service following the 1997 inspection.

Under the NRC's revised reactor oversight process, the agency assesses the inspection findings and characterizes their risk significance by color, specifically green, white, yellow or red. (A green finding results in normal NRC oversight, while white, yellow, or red assessments are considered progressively more serious and receive commensurately greater oversight.) That process assessed the potential impact of running the plant for an operating cycle with the steam generators in a degraded condition. The NRC determined the issue to be of potentially high risk significance. As such, the staff has preliminarily characterized the findings as "red."

While the NRC staff has identified these areas of concern, it is important to note that a review of the February 15th event by an NRC Augmented Inspection Team (AIT) earlier this year found that the plant's licensed operators appropriately responded to the situation, that plant equipment performed as expected and that there were no public health and safety consequences associated with the event itself.

The NRC will meet with Con Ed at a Regulatory Conference, tentatively scheduled for September 26, to discuss the finding. At the conference, which will be held in the NRC's Region I office in King of Prussia, Pa., Con Ed will have an opportunity to provide NRC staff with additional information, including its position on the significance of the issues discussed in the report. This information will be used by the NRC in determining its final characterization of the issues.

The inspection report is posted on the NRC's web site at: http://www.nrc.gov/NRC/REACTOR/IP/index.html.

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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. 00-129 August 29, 2000

NOTE TO EDITORS:

NRC EXTENDS DEADLINE TO SUBMIT NOMINATIONS FOR ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Nuclear Regulatory Commission (NRC) has extended its deadline to October 16 for accepting applications of qualified candidates seeking appointment to two vacancies on its Advisory Committee on Reactor Safeguards (ACRS).

The ACRS was established by Congress to provide the NRC with independent expert advice on matters related to licensing and the safety of existing and proposed nuclear power plants.

A resumé describing the educational and professional background of the candidate, including any special accomplishments, professional references, current address and telephone number should be provided. Criteria used to evaluate candidates include education and experience, demonstrated skills in nuclear safety matters, and the ability to solve problems. Candidates must be citizens of the U.S. All candidates will receive careful consideration. An indication of the candidate's ability and willingness to devote the time required (approximately 60-100 days per year) should also be provided. Copies of resumés of nominees should be sent to the Office of Human Resources, ATTN: Robin Avent, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555-0001.

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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. 00-141 September 14, 2000

NRC TO BROADCAST COMMISSION MEETING LIVE OVER THE INTERNET

The Nuclear Regulatory Commission will broadcast live over the Internet the second public meeting as part of a pilot program to test "media streaming" technology. The Commission meeting, featuring a proposal to revise special requirements for certain structures, systems and components at nuclear power plants based on their safety significance, will begin at 9:30 a.m. on September 29.

Over the next eight months (from now through March 26, 2001), the NRC will broadcast up to 20 open Commission meetings as a means of improving and expanding communications with the public. All "streamed" meetings will be archived and available to Internet users worldwide at http://www.nrc.gov/live.html.

Commission meetings can be viewed from a personal computer, thus eliminating travel costs to NRC headquarters. To observe Commission meetings, users need a computer equipped with a sound card and speakers, access to the Internet, and Real Networks Player software (a free version is available for download from the http://www.nrc.gov/live.html web page). Detailed instructions are provided at the web site for accessing meetings, as well as a toll-free telephone number and e-mail address for assistance.

The first Commission meeting to be broadcast was unsuccessful due to technical difficulties experienced during the nationwide telephone strike. Various fixes and backup provisions have been made which should help ensure smooth transmission over the Internet for the September 29 meeting. If a technical problem develops, a message will appear on users' screens notifying them of technical difficulties.

The web page provides viewers an opportunity to provide comments on the broadcasts. The agency will use this feedback in determining the value of providing this service in the future. Meeting transcripts and a complete listing of Commission meetings will continue to be available on NRC's home page at http://www.nrc.gov/NRC/PUBLIC/meet.html#COMMISSION .

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 15, 2000

NOTE TO: William D.Travers, EDO
Carl J. Paperiello, OEDO
Frank J. Miraglia, Jr., OEDO
John W. Craig, OEDO
Sam Collins, NRR
William Kane, NMSS
Janice Dunn-Lee, OIP
John Larkins, ACRS
William M. Beecher, OPA
Dennis K. Rathbun, OCA
Annette L. Vietti-Cook, SECY
Hubert J. Miller, RGN-1
Luis A. Reyes, RGN-II
James E. Dyer, RGN-III
Ellis W. Mershoff, RGN-IV

FROM: Ashok C. Thadani, RES M V Federline for

SUBJECT: 28TH WATER REACTOR SAFETY MEETING (WRSM)

The Office of Nuclear Regulatory Research will hold its 28th WRSM at the Bethesda Marriott Hotel, Bethesda, MD on October 23-25, 2000. We invite you to participate.

Copies of the preliminary agenda for the 28th WRSM are attached. Please distribute them to interested staff members in your offices.

Attachment: As stated

Monday, October 23, 2000

8:00 **Plenary Session (Grand Ballroom)**

Opening remarks and welcome
Ashok C. Thadani, Director, Office of Nuclear Regulatory Research

Keynote speaker:
Richard Meserve, Chairman, NRC

9:00 Break

9:15 **Expert Panel: *Twenty-Five Years Since the Reactor Safety Study — The Legacy and the Lessons***

Panel Members: G. Apostolakis (MIT), A. Birkhofer (GRS), R. Budnitz (FRA), R. Denning (BCL), B.Garrick (ACNW), H. Lewis (UCSB), J. Murphy (NRC), W. Vesely (Consultant), A. Thadani, Panel Moderator

12:00 Luncheon Sponsored by MIT - Congressional Ballroom

1:30 **Plenary Session (Grand Ballroom)**
Guest speaker, Nils J. Diaz, Commissioner, NRC

2:00 **Expert Panel: *Challenges in the Future for Risk-Informed Regulation***
Panel Members: M. Federline (NRC), D. Helwig (ComEd), D. Lochbaum (UCS), D. Powers (ACRS), R. Zimmerman (NRC), W. Travers (NRC), Panel Moderator

3:30 Break

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1 PRA Today: Risk-Informed Regulation

Chaired by: J. Johnson (NRC), S. Floyd (NEI)
Rapporteur: T. King
Objective: Discuss activities, benefits, objectives and rationale for risk-informed regulation.

- 3:45 Introduction, J. Johnson (NRC)
- 3:50 Risk Informing Part 50 Technical Requirements, M. Cunningham (NRC)
- 4:10 Use of PRA Results in Regulatory Decision-Making, G. Holahan (NRC)
- 4:30 Transition to Risk-Informed Regulation, R. Bari (BNL)
- 4:50 Industry Perspectives on the Role of PRA, S. Floyd (NEI)
- 5:10 Facilitated Discussion

2 Dry Cask Storage and Transportation of Spent Nuclear Fuel

Chaired by: A. Murphy (NRC), A. Machiels (EPRI)
Rapporteur: M. Mayfield
Objective: Identify technical issues and research efforts associated with assuring the integrity of dry cask storage and transportation of spent nuclear fuel.

- 3:45 Introduction, A. Murphy (NRC)
- 3:50 Dry Cask Storage and Transportation - Regulatory Perspective, A. Murphy (NRC)
- 4:05 Dry Cask Storage and Transportation - Industry Perspective, A. Machiels (EPRI)
- 4:20 Inspection of the Castor V/21 Cask and Contents, R. Kenneally (NRC), J. Kessler (EPRI)
- 4:40 Research Supporting Effective Implementation of Burnup Credit in the Criticality Safety Assessment of Transport and Storage Casks, C. Parks, J. Wagner, M. DeHart (ORNL), D. Ebert (NRC)
- 5:00 Facilitated Discussion

12:00 - 1:30 Luncheon sponsored by MIT
Guest speaker John Ahearne, Former NRC Chairman *(Seating for Speech only at 12:45)*

6:30 - 8:00 Poster Session
Sponsored by National Labs

8:30 Plenary Session (Congressional Ballroom)

Guest speaker: Greta Joy Dicus, Commissioner, NRC

3 High Burnup Fuel

Chaired by: R. Meyer (NRC), R. Yang (EPRI)
Rapporteur: F. Eltawila
Objective: Describe new research to develop or confirm regulatory criteria and evaluation models for high burnup fuel and new cladding alloys.

- 9:00 Introduction, R. Meyer (NRC)
- 9:15 Fission Gas Release Measurements in Relation to ANS Standards Modeling of Radiological Releases, E. Kolstad, T. Turnbull, W. Wiesenack (Halden)
- 9:40 Short-Time Creep and Rupture Tests on High Burnup Fuel Rod Cladding, W. Goll, (Siemens AG), E. Toscano (ITF Karlsruhe), H. Spilker (GNB mbH)
- 10:00 Definition and Status of the CABRI International Program with a Sodium Loop and a Water Loop, J-C. Melis, J. Papin (IPSN)
- 10:15 High Burnup BWR Fuel Response to Reactivity Transients and a Comparison with PWR Fuel Response, T. Fuketa, et al. (JAERI)
- 10:40 The History of LOCA Embrittlement Criteria, G. Hache (IPSN), H. Chung (ANL)
- 11:05 High-Temperature Steam Oxidation of Zircaloy Cladding from High Burnup Fuel Rods, Y. Yan, et al. (ANL)
- 11:30 Facilitated Discussion

4a PWR Sump Blockage and Containment Coatings Service Level I Safety Concerns

Chaired by: A. Serkiz (NRC), T. Andreychek (Westinghouse)
Rapporteur: M. Mayfield
Objective: Describe the analytical and experimental methods used to assess safety concerns and discuss industry participation in ongoing assessments.

- 9:00 Introduction, Overview of Principal Findings and Industry Involvement, M. Mayfield, A. Serkiz (NRC), T. Andreychek (Westinghouse)
- 9:15 Panel Discussion: Identifying Key Considerations to Resolve Safety Concerns, Recent Findings, and Outstanding Research Needs: PWR Sump Blockage (M. Marshall, NRC)
  Industry Coatings PIRT Findings (J. Cavallo, CCC&L)
  Service Level I Coatings Behavior (A. Serkiz, NRC)
  PWR Applicability (T. Andreychek, Westinghouse)
  Licensee Views (C. Harrington, K. Jacobs, PWR Industry)
- 10:00 Facilitated Discussion

4b Digital Instrumentation and Control

Chaired by: J. Calvert (NRC), R. Wood (ORNL)
Rapporteur: M. Mayfield
Objective: Identify technical issues and research efforts associated with digital I&C issues

- 10:30 Introduction, J. Calvert (NRC)
- 10:45 Overview of NRC Digital I&C Research Program, J. Calvert, T. Jackson (NRC)
- 11:30 Facilitated Discussion

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5 Thermal Hydraulic and Severe Accident Analysis for Reactors and Spent Fuel

Chaired by: C. Tinkler (NRC), D. Modeen (NEI)
Rapporteur: F. Eltawila
Objective: Present NRC's recent activities and initiatives to improve analytical capabilities and apply new methodologies to better quantify safety margins in support of regulatory decision making.

- 1:30 Introduction, C. Tinkler (NRC)
- 1:35 Application of a CFD Code for T/H Analysis of Spent Fuel Pool Accidents, C. Boyd (NRC)
- 2:05 Evaluation of Uncertainty in Steam Generator Tube Thermal Response During Severe Accidents, S. Arndt (NRC), D. Knudson (INEEL)
- 2:30 USNRC Thermal Hydraulics Program, J. Uhle, C. Gingrich (NRC)
- 3:00 Break
- 3:15 Improved Radiological Consequence Assessment for Dry and Wet Storage of Spent Fuel, J. Schaperow (NRC)
- 3:50 Consolidation of Severe Accident Code Capabilities into MELCOR, R. Gauntt (SNL)
- 4:20 Facilitated Discussion

6 Integrity of the Reactor Coolant Pressure Boundary

Chaired by: M. Kirk (NRC), R. Hardies (BG&E)
Rapporteur: M. Mayfield
Objective: Determine the reasons for regulatory concern when innovative technologies are applied to address problems associated with the integrity of the reactor coolant pressure boundary and identify candidate strategies to address these concerns.

- 1:30 Introduction, M. Kirk (NRC)
- 1:45 Research Perspectives on Evaluation of Reactor Pressure Vessel Integrity, J. Muscara (NRC)
- 2:20 Current Issues in the Regulation of Steam Generator Tube Integrity, TBA (NRC)
- 2:55 Break
- 3:10 EPRI Materials Reliability Program: Master Curve Activities, S. Rosinsky (EPRI), R. Hardies (BG & E)
- 3:45 NRC Review of Technical Basis for Use of the Master Curve in Evaluation of Reactor Pressure Vessel Integrity, M. Kirk (NRC)
- 4:20 Facilitated Discussion

12:00 Luncheon sponsored by Elsevier Science
Guest speaker to be announced *(Seating for Speech only at 12:45)*

8:30 **Plenary Session (Grand Ballroom)**

Guest speaker: Jeffrey S. Merrifield, Commissioner, NRC

9:00 **Expert Panel: *The Future Role of Nuclear Power and the Need for Nuclear Regulatory Research***

Panel Members: R. Budnitz (FRA), D. Lochbaum (UCS), A. Marion (NEI), T. Marston (EPRI) K. Mossman (ASU), Edward McGaffigan, Jr., Commissioner, NRC, Panel Moderator

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7 Reactor Decommissioning

Chaired by: C. Trottier (NRC), P. Genoa (NEI)
Rapporteur: T. King

Objective: Discuss plans and rationale to address decommissioning issues and future needs.

- 1:00 Introduction - Current NRC Initiatives, C. Trottier (NRC)
- 1:10 Needed Research to Support Decommissioning — An Industry Perspective, P. Genoa (NEI)
- 1:30 Improved Codes for Assessing Compliance with License Termination Rule, R. Cady (NRC), S.Y. Chen (ANL)
- 1:50 Survey Methodology for Volumetrically Contaminated Material, E. Abelquist (ORISE), C. Gogolak (EML)
- 2:10 Entombment as a Decommission Option, TBA (NRC)
- 2:30 Facilitated Discussion

8 Regulatory Effectiveness

Chaired by: J. Rosenthal (NRC), K. Ainger (ComEd)
Rapporteur: F. Eltawila

Objective: Discuss NRC research activities supporting the assessment and improvement of regulatory effectiveness and related industry activities.

- 1:00 Introduction, J. Rosenthal (NRC)
- 1:05 Regulatory Effectiveness: What It Is and What It Shows for Station Blackout Rule, B. Raughley (NRC)
- 1:30 Licensee Proposals to Reduce Unnecessary Burden, K. Ainger (ComEd)
- 1:55 High-Level Guidelines for Performance-Based Activities, P. Kadambi (NRC)
- 2:20 Relationship Between Deregulation of the Electric Power Industry and Reduction of Unnecessary Burden, E. Quinn (DOE Task Force)
- 2:45 Facilitated Discussion

3:30 **Plenary Session (Grand Ballroom)**

Rapporteur Panel: C. Ader, Director, Program Management, Policy Development and Analysis Staff
F. Eltawila, Acting Director, Division of Systems Analysis and Regulatory Effectiveness
T. King, Director, Division of Risk Analysis and Applications
M. Mayfield, Director, Division of Engineering Technology

SESSION SCHEDULE & ROOM LOCATIONS

MONDAY	8:00 am	Plenary Session and Expert Panel	Grand Ballrooms B,C, D & E
	12:00 pm	MIT-Sponsored Lunch	Congressional Ballroom
	1:30 pm	Plenary Session and Expert Panel	Grand Ballrooms B,C, D & E
	3:45 PM	Session 1	Grand Ballrooms B & C
		Session 2	Grand Ballrooms D & E
	6:30 pm	National Labs Sponsored Poster Session	Congressional Ballroom
TUESDAY	8:30 am	Plenary Session	Congressional Ballroom
	9:00 AM	Session 3	Grand Ballrooms B & C
		Session 4A	Grand Ballrooms D & E
	10:30 am	Session 4B	Grand Ballrooms D & E
	12:00 pm	Elsevier Science-Sponsored Luncheon	Congressional Ballroom
	1:30 PM	Session 5	Grand Ballrooms B & C
		Session 6	Grand Ballrooms D & E
WEDNESDAY	8:30 am	Plenary Session and Expert Panel	Grand Ballrooms B,C, D & E
	11:30 am	Lunch — On Your Own	See Hotel for Restaurant Specials
	1:00 PM	Session 7	Grand Ballrooms B & C
		Session 8	Grand Ballrooms D & E
	3:30 pm	Rapporteur Panel	Grand Ballrooms B,C, D & E

11:30 LUNCH

(2)

Union of Concerned Scientists

Presentation to ACRS on *Nuclear Plant Risk Studies: Failing the Grade*

In August 2000, UCS released *Nuclear Plant Risk Studies: Failing the Grade*. Almost immediately, we heard folks argue that our study was flawed because we had relied on 'obsolete' results from the Individual Plant Examinations (IPEs) that were nearly a decade old. We would have preferred to use current results from the updated Plant Safety Assessments (PSAs), but that information is not publicly available. We note that the NRC also lacks access to the PSA results. The site specific worksheets for the significance determination process (SDP) "were developed based on your Individual Plant Examination (IPE) submittal that was requested by Generic Letter GL 88-20."[1] Thus, the NRC deems it acceptable to rely on outdated results when classifying the safety significance of a present-day inspection finding yet the agency considers it unacceptable for us to use those same results. Curious.

We have also heard some folks argue that our evaluation—especially the case studies—was flawed because the IPE results should be different because the plants themselves are different. In fact, one critic of our report essentially stated that our case studies were meaningless because we had compared reactors at different sites, albeit of similar design. This critic told me that there were sufficient differences between reactors at the same site (e.g., Browns Ferry Units 2 and 3) to cause the risk numbers to be different.

We looked into this criticism using the only data that is publicly available—the IPE results. The NRC has a database of IPE results on its website.[2] We sorted the results from highest to lowest overall core damage frequency (CDF) and printed out the results. <SLIDES 1A, 1B & 1C> With the exceptions of Beaver Valley 1&2, Salem 1&2, Indian Point 2&3, St. Lucie 1&2, and Hatch 1&2, all of the sites having multiple units of similar design reported exactly the same risk for all units. If Indian Point 2's risk is different from Indian Point 3's risk because of different design features, equipment performance, or procedures, then why is Turkey Point 3's risk exactly the same as Turkey Point 4's risk? Why is Palo Verde 1's risk exactly the same as Palo Verde 2's risk and exactly the same as Palo Verde 3's risk? Why is Sequoyah Unit 1's risk exactly the same as Sequoyah Unit 2's risk, but different than Watts Bar's risk? Why is North Anna Unit 1's risk exactly the same as North Anna Unit 2's risk, but different than Surry Unit 1's risk and Surry Unit 2's risk? Apparently, the NRC views being the same as okay and being different as okay too. Curious.

We have also heard some folks argue that our concern about the risk assessments neglecting design bases problems is over-stated because although plenty of design bases problems have been discovered, none had any safety significance.

We looked into this criticism. The NRC recently issued a draft report on design bases problems.[3] <SLIDE 2> Figure 22 provides the percentage of licensee event reports with design bases issues that were classified as accident sequence precursor events between 1990 and 1997. A declining trend is shown, but

[1] Letter dated January 3, 2000, from Jefferey F. Harold, Project Manager, Nuclear Regulatory Commission, to A. Alan Blind, Vice President - Nuclear Power, Consolidated Edison Company of New York, Inc., "Site Specific Worksheets for Use in the NRC's Significance Determination Process."

[2] http://www.nrc.gov/NRC/NUREGS/SR1603/index.html

[3] Ronald L Lloyd, John R. Boardman, and Sada V. Pullani, Nuclear Regulatory Commission, "Cause and Significance of Design-Basis Issues at U.S. Nuclear Power Plants," Draft May 2000.

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the fact remains that the percentage was non-zero each and every year. Given that nuclear plants are designed to a single failure criterion, evidence of significant design bases problems coupled with knowledge that design bases problems are neglected in the risk assessments should prompt more than a Rhett Butler response from the NRC.

In our report, we concluded—among other things—that "the NRC is guessing when it makes safety decisions using the results from incomplete and inaccurate probabilistic assessments." We feel that the material presented in our report firmly justifies that conclusion. And we call your attention to additional NRC documents that we gathered during our research, but did not use in the report.

In September 1999, the NRC issued a reliability study on the isolation condenser system in BWRs.[4] <SLIDE 3> Figure 6 from this report compares the system unreliability numbers used in the IPEs to the systems' actual operating performance. In all cases, the average system unreliability value used in the IPE is lower than the average unreliability value from actual plant-specific operating experience. In other words, isolation condenser system is less reliable in real-life than assumed in the IPEs.

The following month, the NRC issued a draft reliability study on the high pressure coolant injection (HPCI) system in BWRs.[5] <SLIDE 4> Figure ES-2 compares the system unreliability numbers used in the IPEs to the systems' actual operating performance. In all cases, the average system unreliability value used in the IPE is lower than the average unreliability value from actual plant-specific operating experience. In other words, the high pressure coolant injection system is less reliable in real-life than assumed in the IPEs. At one site (Hope Creek), the uncertainty bands applied to the IPE value and to operating data do not even overlap.

The NRC also issued a draft reliability study on the reactor core isolation cooling (RCIC) system in BWRs.[6] <SLIDE 5> Figure 13 compares the system unreliability numbers used in the IPEs to the systems' actual operating performance. In all cases, the average system unreliability value used in the IPE is lower than the average unreliability value from actual plant-specific operating experience. In other words, the reactor core isolation cooling system is less reliable in real-life than assumed in the IPEs. At seven (7) of the thirty (30) sites, the uncertainty bands applied to the IPE value and to operating data do not even overlap. The uncertainty band applied to the IPE value does not encompass the average operating data value at any one of the 30 sites.

The story is totally different for safety system reliability in PWRs. The NRC issued a reliability study on the auxiliary/emergency feedwater (AFW/EFW) system in August 1998.[7] <SLIDE 6> Figure 9 compares the system unreliability numbers used in the IPEs to the systems' actual operating performance. Unlike the BWR results, the actual operating experience is <u>not</u> to the right of the IPE data. Nope, the NRC changed to a vertical axis for unreliability. In all but three cases, the average system unreliability value used in the IPE is lower than the average unreliability value from actual plant-specific operating experience. In other words, the auxiliary/emergency feedwater system is less reliable in real-life than assumed in the majority of the IPEs.

The recurring theme of these NRC reports is that actual safety system reliability is biased lower than that assumed in the IPEs. Trying to ascertain the source of this bias, we examined NRC reports on component

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[4] Nuclear Regulatory Commission, NUREG/CR-5500 Vol. 6, "Reliability Study: Isolation Condenser System, 1987-1993," September 1999.
[5] Nuclear Regulatory Commission, NUREG/CR-xxxx, "Reliability Study Update: High Pressure Coolant Injection (HPCI) System, 1987-1998," October 1999.
[6] Nuclear Regulatory Commission, NUREG/CR-xxxx, "Reliability Study Update: Reactor Core Isolation Cooling (HPCI) System, 1987-1998," October 1999.
[7] Nuclear Regulatory Commission, NUREG/CR-5500 Vol. 1, "Reliability Study: Auxiliary/Emergency Feedwater System, 1987-1995," August 1998.

iability. We wanted to see if the biases existed at the component level and were then rolled up into the ?m reliabilities, or if the biases were introduced elsewhere.

NRC recently issued a draft reliability study on motor-driven pumps.[8] <SLIDE 7> Figure 4 and 5 .i several other figures as well—these two are illustrative of the findings across the board) compares motor-0driven pump failure on demand probabilities used in IPEs to the components' actual operating formance. Unlike the system reliability comparisons, these figures reveal that actual component rformance scatters around the failure probabilities assumed in the IPES with about as many data points ove the assumed probabilities as below. In other words, the actual component failure on demand rates sonably approximate those assumed in the IPEs. One potential cause for the system-level biases is ninated.

e suspect, but cannot yet conclusively prove, that a contributing cause is undefined definitions. <SLIDE Each of the NRC system reliability studies cited above contains an illustration of the failures used in e engineering analysis. This figure shows that there are reported inoperabilities (Round Thing A) from hich a subset of failures are classified (Round Thing B) from which a subset of failures are counted Round Thing C). The system-level bias might result from the industry's Round Thing C being smaller an the NRC's.

ven if the system-level biases discussed above could be thoroughly eliminated, the results from the risk ssessments would still form an unsuitable foundation for making regulatory decisions, particularly ecisions involving a line drawn between acceptable and unacceptable conditions. The risk assessments i[illegible]plete. Among other failings, the risk assessments focus exclusively on core damage frequencies n[illegible]here are credible events that can result in serious problems without core damage. Spent fuel ol accidents are just one example in this category.

he NRC keeps issuing reports that clearly document non-conservatisms in the risk assessments. Yet the gency is moving towards risk-informed regulation knowing that 'PRA quality' is an oxymoron. We elieve that the NRC should suspend the move to risk-informed regulation and take steps to address oncerns about risk assessment content and availability. After the questions about risk assessments are esolved, the NRC could resume the move towards risk-informed regulations. Unless, of course, the gency elected to devote those resources to improving safety rather than merely maintaining safety.

[8] Nuclear Regulatory Commission, NUREG-1715 Vol. 2 Draft, "Component Performance Study — Motor-Driven Pumps, 1987-1998."

US Nuclear Plant Core Damage Frequencies from IPEs

CORE DAMAGE FREQUENCY	PLANT NAME
1.17E-03	SURRY 1&2
4.62E-04	TURKEY POINT 3&4
3.30E-04	WATTS BAR 1&2
3.20E-04	H.B. ROBINSON 2
2.40E-04	CALVERT CLIFFS 1&2
2.14E-04	BEAVER VALLEY 1
2.00E-04	SUMMER
1.92E-04	BEAVER VALLEY 2
1.90E-04	HADDAM NECK
1.70E-04	SEQUOYAH 1&2
1.30E-04	FARLEY 1&2
1.15E-04	POINT BEACH 1&2
1.00E-04	NRC's Objective
9.00E-05	PALO VERDE 1,2,&3
8.80E-05	DIABLO CANYON 1&2
8.74E-05	GINNA
7.97E-05	COOPER
7.40E-05	MAINE YANKEE
7.40E-05	SURRY 1&2 INTERNA
7.16E-05	NORTH ANNA 1&2
7.00E-05	SHEARON HARRIS 1
6.70E-05	SEABROOK
6.65E-05	KEWAUNEE
6.60E-05	DAVIS-BESSE
6.35E-05	SALEM 2
6.26E-05	D.C. COOK 1&2

SLIDE 1A

CORE DAMAGE FREQUENCY	PLANT NAME
6.25E-05	SALEM 1
5.85E-05	CALLAWAY
5.80E-05	CATAWBA 1&2
5.80E-05	PILGRIM 1
5.72E-05	COMANCHE PEAK 1&
5.61E-05	MILLSTONE 3
5.40E-05	BIG ROCK POINT
5.07E-05	PALISADES
5.00E-05	PRAIRIE ISLAND 1&2
4.90E-05	VOGTLE 1&2
4.80E-05	BROWNS FERRY 2
4.74E-05	LA SALLE 1&2
4.67E-05	ARKANSAS NUCLEAR
4.63E-05	HOPE CREEK
4.49E-05	TMI 1
4.40E-05	INDIAN POINT 3
4.27E-05	SOUTH TEXAS PROJ
4.20E-05	WOLF CREEK
4.00E-05	MCGUIRE 1&2
3.42E-05	MILLSTONE 2
3.40E-05	ARKANSAS NUCLEAR
3.13E-05	INDIAN POINT 2
3.10E-05	NINE MILE POINT 2
3.09E-05	BYRON 1&2
3.00E-05	SAN ONOFRE 2&3
2.74E-05	BRAIDWOOD 1&2
2.70E-05	BRUNSWICK 1&2
2.60E-05	MONTICELLO
2.60E-05	CLINTON

SLIDE 18

CORE DAMAGE FREQUENCY	PLANT NAME
2.60E-05	ST. LUCIE 2
2.36E-05	HATCH 2
2.30E-05	OCONEE 1,2,&3
2.30E-05	ST. LUCIE 1
2.23E-05	HATCH 1
1.85E-05	DRESDEN 2&3
1.75E-05	WNP 2
1.72E-05	GRAND GULF 1
1.70E-05	WATERFORD 3
1.55E-05	RIVER BEND
1.53E-05	CRYSTAL RIVER 3
1.36E-05	FORT CALHOUN 1
1.32E-05	PERRY 1
1.10E-05	MILLSTONE 1
7.84E-06	DUANE ARNOLD
5.70E-06	FERMI 2
5.53E-06	PEACH BOTTOM 2&3
5.50E-06	NINE MILE POINT 1
4.30E-06	VERMONT YANKEE
4.30E-06	LIMERICK 1&2
4.00E-06	ZION 1&2
3.69E-06	OYSTER CREEK
1.92E-06	FITZPATRICK
1.20E-06	QUAD CITIES 1&2

SLIDE 1C

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Draft NRC
Report

Figure 22, since it is expected that DBIs will continue to be reported by licensees, and that some of them will be potentially safety significant.

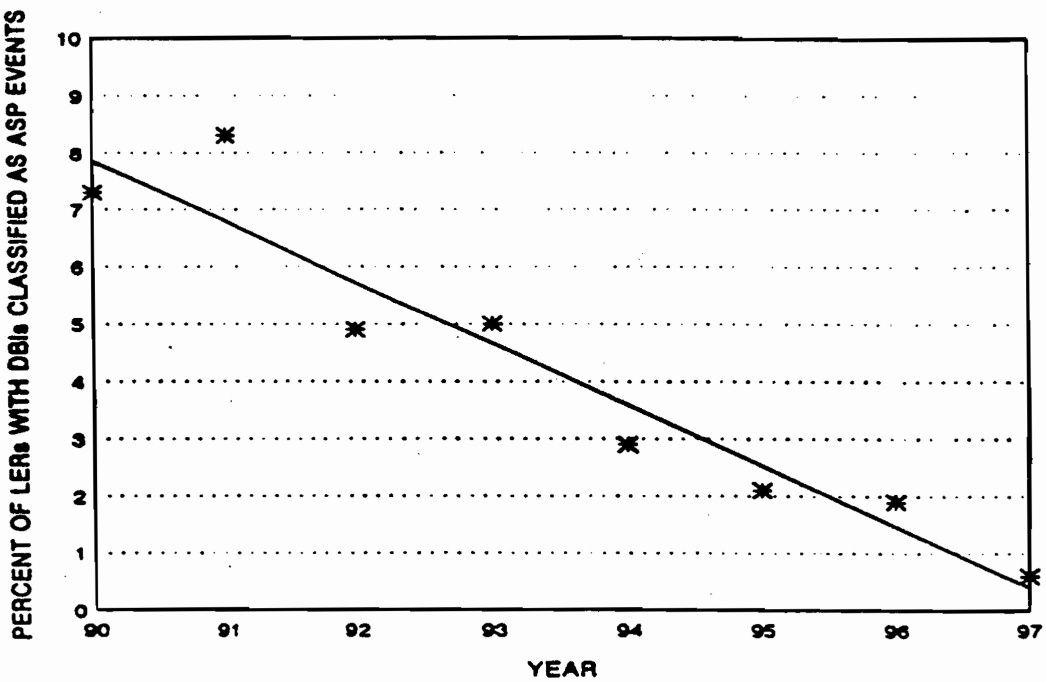
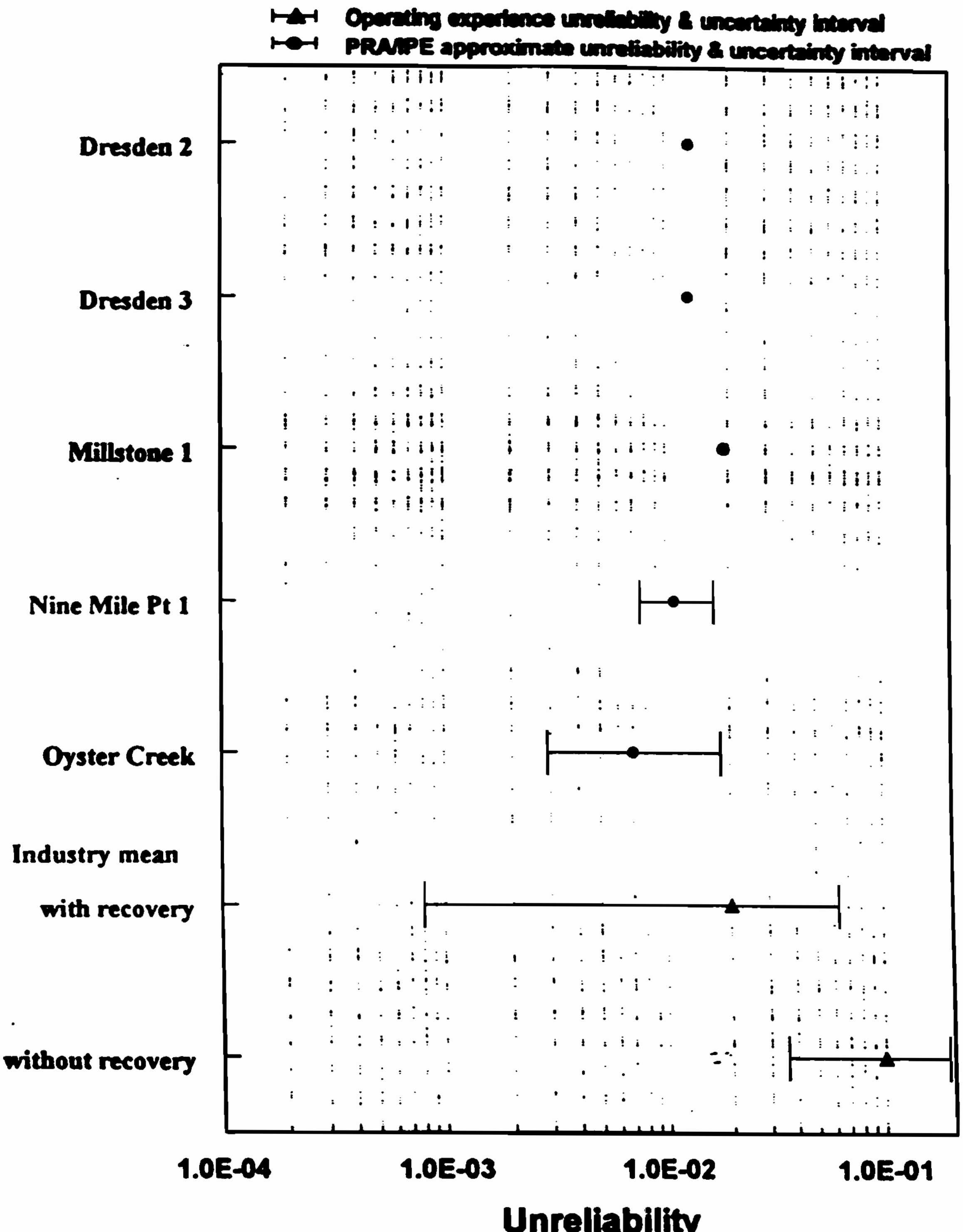


Figure 22 Percent of licensee event reports with design-basis issues classified as accident sequence precursor events

Accident Sequence Precursor Events Results (1997)

A search using the ORNL ASP database, showed that three (approximately 0.6 percent) of the 512 DBIs reported in 1997 had a CCDP of at least 1.0×10^{-6}. For all the LERs for 1997, the ASP computer search algorithm selected 797 for engineering review as potential precursors. Of these, 48 LERs (55 including revisions) were determined to be potentially significant. Of these 48, 31 LERs were rejected after detailed analysis, 2 LERs were determined to be impractical to analyze, and 8 LERs (5 events) were documented as "interesting" events. Review and analysis of the events described in the remaining seven LERs led to the identification of five (0.25 percent) of the 1975 total number of LERs as ASP events as shown in Table 4. All five ASP events involved older PWRs located in either Regions I or II.

SLIDE 2



ISOLATION CONDENSER

Figure 6. Plot of IC train unreliabilities approximated from PRA/IPE information and estimates of IC train unreliability (with and without recovery) calculated from the operational experience data. (For some plants the information documented in the PRA/IPEs was insufficient to generate uncertainty intervals.)

SLIDE 3

lower than the industry-wide estimate based on the 1987–1998 experience. However, this comparison is based on data from the original IPE submittals. A plot of these estimates is shown in Figure ES-2. Section 3.3 provides the results and insights for comparison with PRA/IPE results.

The leading contributors to HPCI system unreliability based on the fault tree developed for comparison to PRA/IPEs and using the 1987–1998 experience are the failure of the injection valve to reopen (cycling the injection valve for subsequent for reactor pressure vessel water level control) and failure to run of the injection system, 55% and 42%, respectively.

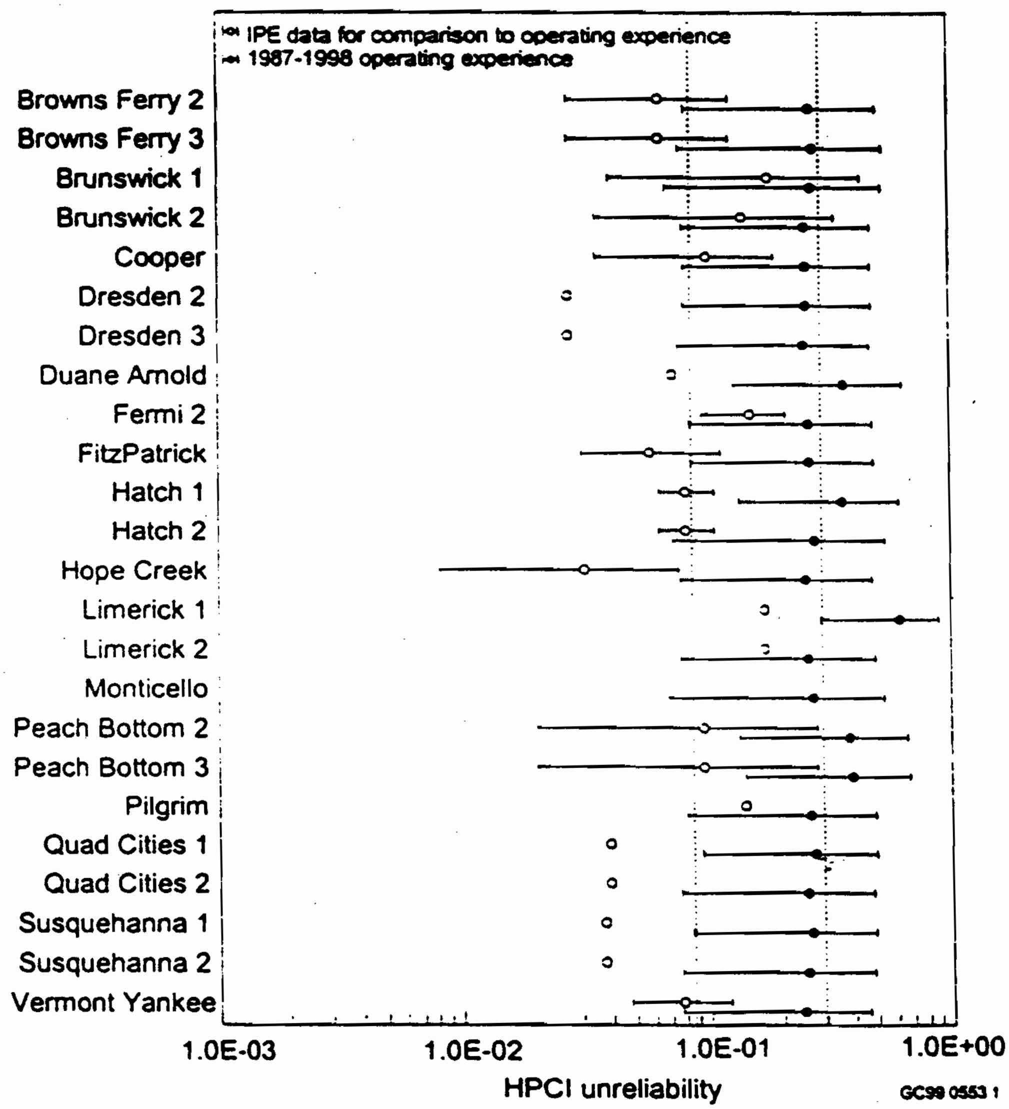


Figure ES-2. Plant-specific estimates of HPCI system unreliability for a PRA comparison based on the 1987–1998 operating experience and compared to estimates calculated using component failure probabilities found in the PRA/IPEs. (The dashed lines represent the corresponding industry-wide averages.)

SLIDE 4

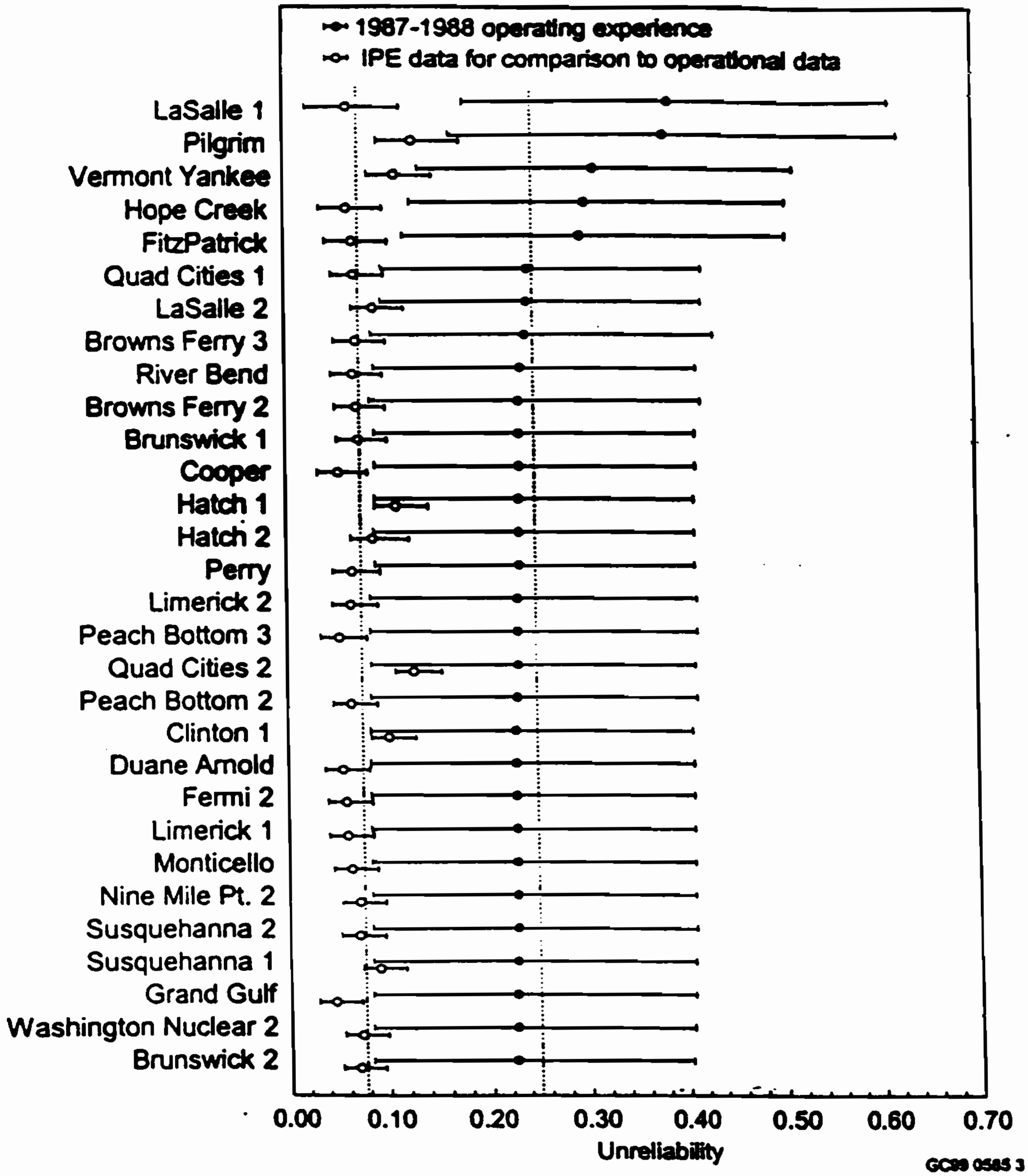
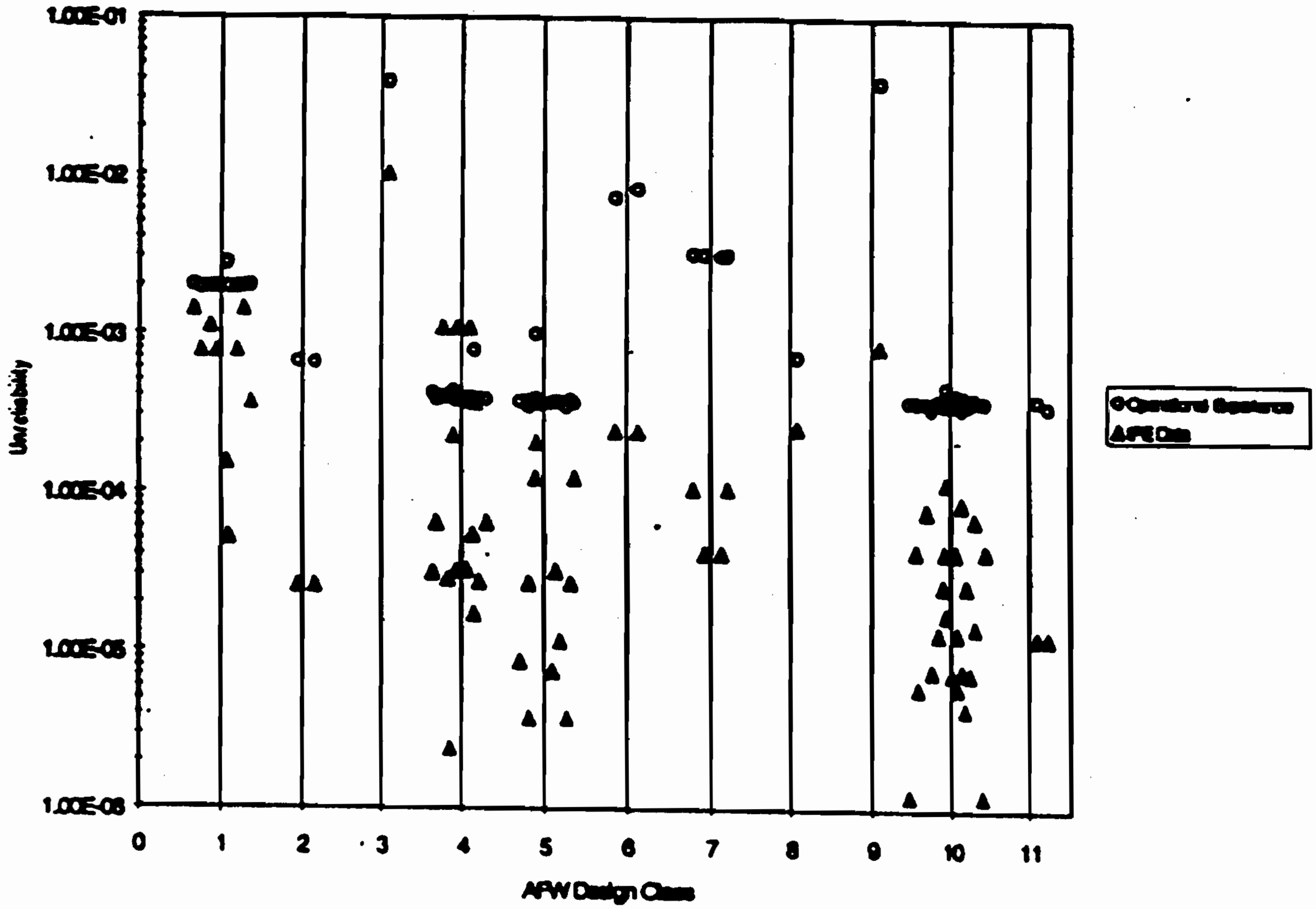


Figure 13. Plant-specific estimates of RCIC system unreliability calculated using the 1987–1998 operating experience compared to the unreliability estimates calculated using the PRA/IPE data. The estimates were calculated using the PRA comparison fault tree model shown in Figure C-1 of Appendix C.

SLIDE 5



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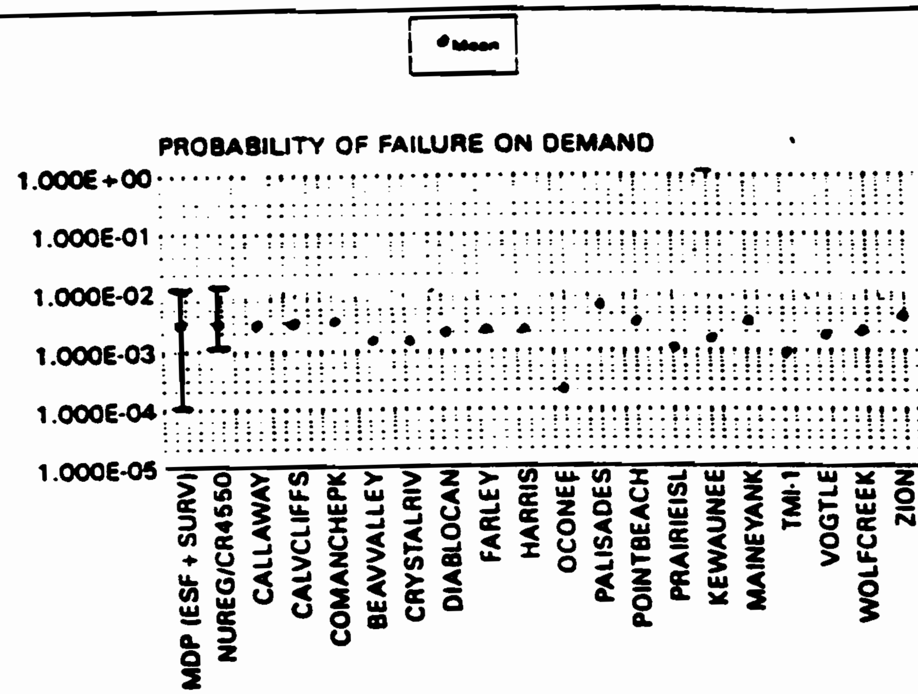
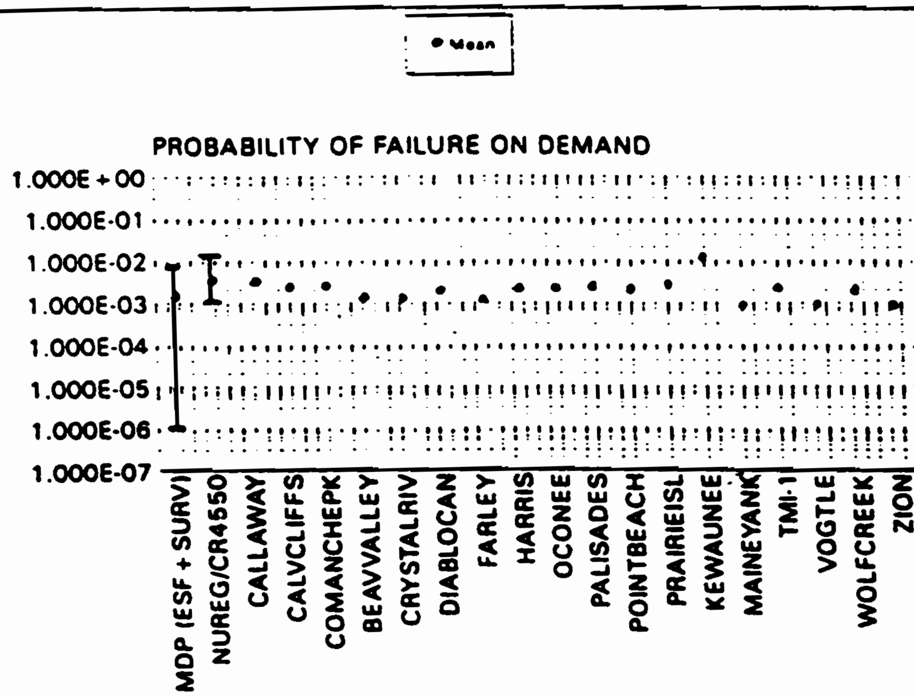
SLIDE 6



PWR HPI SYSTEM MDP PROBABILITY OF FAILURE ON DEMAND - COMPARISON WITH VALUES USED IN IPEs
FIGURE 4



PWR CCW SYSTEM MDP PROBABILITY OF FAILURE ON DEMAND - COMPARISON WITH VALUES USED IN IPEs
FIGURE 5

SLIDE 7

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Vol. 2
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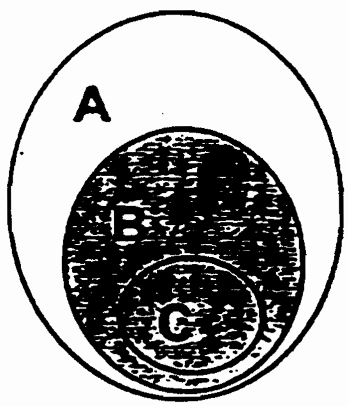

A Represents all the inoperabilities identified from the SCSS database search.

B Represents the inoperabilities that are classified as failures.

C Represents the subset of failures for which the demand counts could be determined or estimated, countable failures.

Figure 15. Illustration of the failures used in the engineering analysis (shown in shaded regions).

Proposals

- Request NRC RES and ACRS observation of peer review process
- Develop closure mechanism for peer review
- Develop template for NRC review of Option 2 submittals
- Develop PRA summary description for inclusion in FSAR for plants implementing option 2

NEI

Considerations

- A PRA standard cannot obviate NRC review for a specific application
- The results of the industry peer review process can be used to focus NRC review
- Industry recognizes that some existing PRAs need improvement to support regulatory reform

NEI

Considerations

- Industry supports development of a PRA standard
- PRA is inherently judgmental
- An overly prescriptive standard cannot assure quality - a high quality peer review will always be required

NEI

Industry PSA Peer Review Process

- Intent: Use peer review results to facilitate focused NRC review of Option 2 applications
  - April 24 NEI letter to NRC
  - Request NRC review in concert with NEI categorization guideline
  - Application-specific review

NEI

3

Role of the Industry PRA Peer Review Process to Support Quality PRA Applications

By
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Vice President
ERIN Engineering and Research, Inc.

Advisory Committee on Reactor Safeguards
October 5, 2000

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Discussion Topics

- Industry PRA Peer Review Process Overview
- Case Study of Certification's Role in Recent Risk-Informed Decision Making at ComEd's PWR plants
- Impact of Peer Review Process to Support decision making

Key Elements of Industry PRA Peer Review Process

- Team of 6 or 7 reviewers with broad expertise in PRA and detailed knowledge of plants and plant PRAs in same owners group; use of common personnel to enhance consistency
- Structured process of PRA elements and sub-elements and checklists to perform a review
- Two to three person months of effort by review team including document review, onsite review, and preparation of final report
- Interactions with PRA team on-site to help interpret PRA documentation

Key Elements (cont'd)

- For each PRA element the following results are obtained:
  - consensus grading of each sub-element and overall element grade (1=IPE, 2=risk ranking, 3=risk informed, 4=risk based applications)
  - documented strengths and weaknesses in the form of Fact and Observation Sheets including those that impact grading: (A = important, immediate update; B = important, defer to next update; C = desirable for applications, D = editorial, S = Superior treatment)
  - recommendations on items for PRA updates, upgrades, and applications
  - consensus on priorities for resolution of each high priority F&O (A,B,S)

Case Study: ComEd Byron and Braidwood EDG AOT Extension

- Decision in 1998 to pursue risk informed EDG AOT extension to 14 days at Byron and Braidwood.
- Performed major PRA upgrade to reflect changes in PRA technology and plant changes since IPE submittal
- WOG PRA Certification Peer Review performed for Braidwood in September 1999 prior to EDG LAR submittal
- PRA model and documentation enhancements to address Category A and B F&O's made before the LAR submittal
- PRA certification cited in LAR submitted to NRC in January 2000 together with summary of risk evaluation and PRA results following RG 1.174 & RG 1.177

Case Study: Continued

- Follow-up WOG PRA Peer Review on Byron (Sister Plant) in June 2000 confirmed that Category A and B issues identified for Braidwood had been addressed
- Response to NRC RAI's on PRA Peer Review Findings and internal flooding risk addressed July 2000
  - Summary of A and B F&Os and how resolved by ComEd prior to LAR preparation and submittal
  - Summary of plant changes to reduce flooding risk and additional PRA evaluations to confirm this reduction
- NRC 9/1/00 SER accepted LAR's use of PRA to perform risk evaluations per RG 1.174 and RG 1.177
- ComEd has completed additional sensitivity analyses to confirm assumptions made in LAR regarding impact of plant modifications to reduce flooding risk.

Risk Management Insights

- Strategy employed to reduce RCP seal LOCA risk identified through ComEd participation in WOG PRA Certification
- Plant modifications to address flooding risk and RCP seal LOCAs effective in managing risk in relation to RG 1.174 and RG 1.177 guidelines
- Plant modifications and flooding risk confirmed via PRA updates not to have any impact on LAR conclusions
- Acceptable delta and incremental risk metrics dependent on compensatory actions (avoidance of concurrent maintenance on risk significant items during extended EDG AOT)
- Configuration Risk Management program plays an important role to minimize temporary risk impacts of longer AOTs

Results of Braidwood PRA Peer Review 9/99

PRA Element		Grades	F&O					
			A	B	C	D	S	Subtotal
IE	Initiating Events	3 (C)	1	2	4	1	1	9
AS	Accident Sequence Evaluation	3 (C)		3	7	2		12
TH	Thermal Hydraulic Analysis	3 (C)		3	2			5
SY	System Analysis	3	1	4	4	1	1	11
DA	Data Analysis	3 (C)		2	3	1	1	7
HR	Human Reliability Analysis	3 (C)	3	3	6			12
DE	Dependencies	3 (C)		2	1	3	1	7
ST	Structural Response	3						
QU	Quantification	3	2		8			10
L2	Containment Performance	3 (C)			2			2
MU	Maintenance & Update	3 (C)		1	2		1	4
	Averages and Subtotals	3 (C)	7	19	39	8	5	78

Resolution of Category A Issues

Category A Issues	Resolution
Lack of documentation for loss of service water initiator recovery	Documentation added to system notebook; no impact on PRA
Modeling of service water cross-tie HRA did not address EDG design feature creating dependence	PRA model was revised to remove credit for action for applicable sequences
HRA update did not include operator interviews or simulator input	HRA updated to reflect input by operator training; sensitivity studies confirmed no impact on EDG LAR
Time window assumption for HP recirculation switchover challenged	PRA model revised to incorporate new time window prior to LAR
Justification for time windows for bleed and feed actions inadequate	HRA was updated and justification provided prior to LAR

Resolution of Category A Issues (cont'd)

Category A Issues	Resolution
Uncertainty and sensitivity analysis not finished at time of review	Quantitative uncertainty analysis and extensive sensitivity analyses completed prior to LAR
Cumulative effect of several A and B issues believed to warrant update	A major PRA update and requantification completed prior to the LAR; issues not impacting EDG AOT issue added to PRA update action tracking system

Additional issues raised in Category B,C, and D F&O's	Many were reflected in PRA model changes or sensitivity analyses prior to LAR; rest placed into PRA update action tracking system

PRA Peer Review Database

ComEd PSA Assumption Database - [Braidwood Certification Observations Table]

File Edit View Insert Format Records Tools Window Help PRA Reports PRA Forms BRW Peer Review Form URE Entry Form

BRAIDWOOD PRA PEER REVIEW CERTIFICATION

Element	Subelement
IE	IE-15

FO #	Significance
IE-03	A

URE #
BB-1999-41

Closure Status
Closed

Closure Date
09/21/1999

Responsible Analyst:
Karl N. Fleming

Observation:

Recovery of dual loss of service water initiating event (DLSX)
The DLSX initiating event frequency is dominated by common cause failure to run of the SX pumps. The Initiating Event Notebook identifies a recovery factor of 0.98 (i.e., 98% mapped to success, and 2% left as the DLSX initiating event frequency and mapped directly to core damage). There is no basis or documentation for this recovery factor. The DLSX initiating event is the dominant scenario (55%) in the current results (excluding the passive dual SX flood).

The necessary technical basis for applying 98% recovery value was documented in ERIN Memo # S1349901-1071, dated 9/21/1999. Need to update the initiating event notebook to reference this memo properly, and this will be done as a part of closure of Action Item # 14879.

Level of Significance:

A

Possible Resolution:

Provide justification for the recovery of the DLSX initiating event.

Plant Response or Resolution:

The basis for assumed recovery action has been documented by ERIN Memo #S1349901-1071, from Karl Fleming, dated September 21, 1999. See the details below. The basis will be incorporated into the IE notebook, and this will be done as a part of closure of Action Item # 14879.

Following are the excerts from the memo:

The current PSA results for Byron and Braidwood consider a Loss of SX initiating event due to common cause failure of the SX pumps. This event would occur during normal plant operation in which there is one operating SX pump and one standby pump for each train of SX. Based on an analysis of the INEEL common cause database, a set of events was identified that could lead to failure of both operating and both standby pumps. This is postulated to result in a dual unit loss of SX initiating event. The frequency was estimated using a model of the SX system for initiating event frequencies and estimates of component failure rates and MGL parameters determined from the screened CCF events.

In the current analysis, it is assumed that there would be opportunities for recovery of the event during the available time to mitigate the event, which would be in part dictated by the RCP seal LOCA event. For this initiator, there would be a high degree of confidence that

Record: 3 of 81

Station response or resolution

Impact of Peer Review

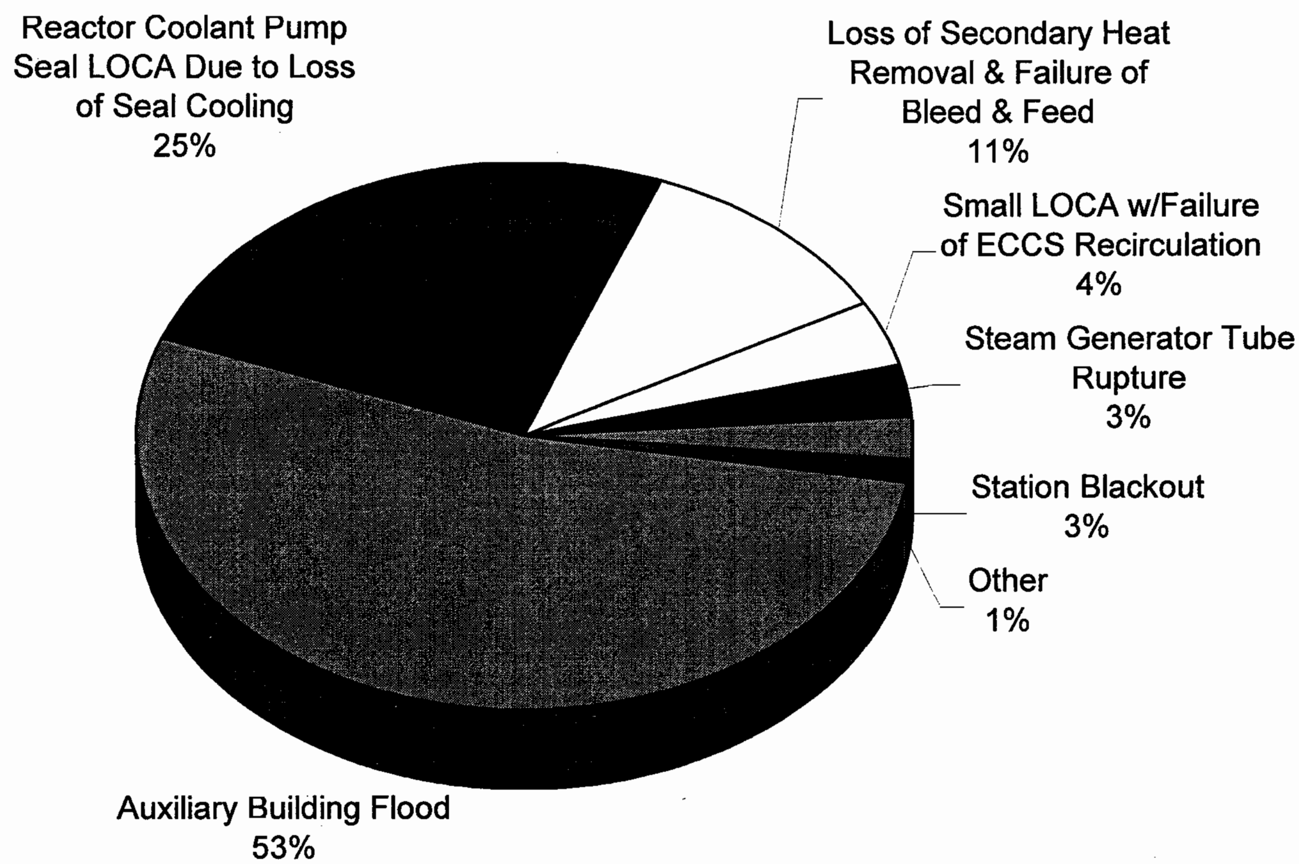
- Grades themselves not used directly
- Category A and B F&O's identified that could impact risk informed decisions in general and the EDG AOT extension specifically; most resulted in PRA enhancements
- Peer review results used to develop and implement a risk management action plan for implementing AOT change
- All F&O's incorporated into ComEd action tracking and PRA update program which requires resolution
- NRC provided with sufficient information to approve the risk evaluation of the EDG AOT extension in a timely manner.
  - Risk evaluation and PRA Summary included in the LAR
  - RAI response summarized resolution of A and B F&O's

Impact of Certification Peer Review Process

- Most important results of the peer review process are:
  - delineation of strengths and weaknesses of existing PRAs for use in applications
  - clear roadmap for how the PRA should be updated or augmented to support specific applications
  - enhanced consistency in treatment of generic issues across the industry as results are incorporated in future updates
- Least important results are the numerical grades

Back-up Slides

Braidwood Level 1 PSA (Internal Events CDF 1.4E-4/yr) (By Accident Class)



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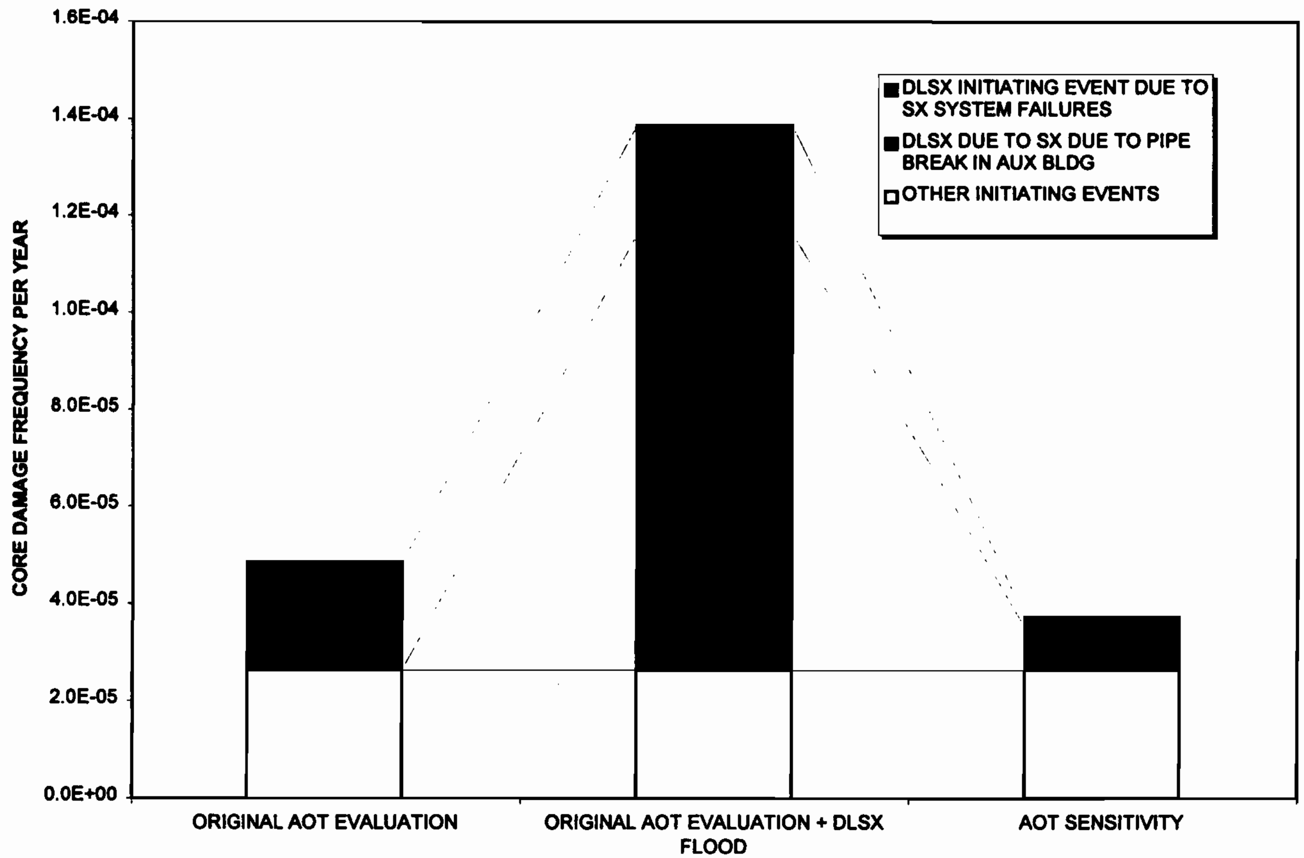
Impact of Floods and Modifications to Reduce Flood Risk on CDF at Braidwood Unit 1



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Key Strengths and Enhancement Opportunities

STRENGTHS	POTENTIAL ENHANCEMENTS
• Comprehensive PRA Update since IPE • Consistent with As-Built, As-Operated Plant • Treatment of CCF models and data • Modeling of multi-unit dependencies • Treatment of System initiators using fault trees	• Offsite power recovery treatment is difficult to track in configuration risk management • Need to update internal flooding study to reflect plant modifications • Operator interviews needed for risk significant operator actions • Consider updating MAAP 3.0B to 4.0 • Other minor technical enhancements and clarifications to documentation • Need for Review of System Notebooks by System Engineer, Operations and Training Departments

PRA Quality Assessment

PRA Elements	Status	Comparative Assessment
Initiating Events	• Complete and up to Date • Frequencies Based on Plant Specific data where feasible • Inadequate basis for DLSX recovery	Strength Enhancement
System Notebooks	• Complete Set of System Notebooks • Additional Evaluation Required for: Room Cooling Technical Basis	Enhancement
Success Criteria	• Complete - generic information • MAAP model (3.0B) exists, but needs V&V • Selected MAAP calculations performed to support success criteria	Strength Enhancement Strength

PRA Quality Assessment (cont'd)

PRA Elements	Status	Comparative Assessment
Accident Sequences	• Event Trees Completely Updated • LOOP Event Tree Modifications Needed to Support Online Risk Management and Future Applications	Strength Enhancement
Data	• Plant Specific data used with Generic data	Strength
Human Reliability Analysis	• Complete update of the HRA using current procedures • Operator Interview work in progress • Consistent with the current state of the art in HRA • Dependencies Explicitly treated	Strength Enhancement Strength Strength

PRA Quality Assessment (cont'd)

PRA Elements	Status	Comparative Assessment
Dependencies	• Common Cause data developed from the latest available NRC data base	Strength
	• Enhanced deterministic room cooling analysis support desirable	Enhancement
	• Limited internal flood update performed to capture postulated risk significant sequences	Enhancement
Structural	• ISLOCA evaluation available	Strength

PRA Quality Assessment (cont'd)

PRA Elements	Status	Comparative Assessment
Quantification	• CAFTA software at the State of the Art Technology • Truncation selected consistent with PSA Applications Guide • Explicit parametric uncertainty quantification	Strength
Level 2	• Simplified Level 2 approach is used • Dependencies between Level 1 and 2 treated in an approximate manner • MAAP assessment available to support plant specific assessment, but out dated	Enhancement

Table 9: Results of Risk Evaluation for Braidwood and Byron – Comparison of Sensitivity Results to Results of Original Evaluation

Risk Metric	Risk Significance Criterion	Risk Metric Results (% of Risk Significance Criterion)			
		Braidwood Unit 1		Byron Unit 1	
		Original Evaluation	Sensitivity Analysis	Original Evaluation	Sensitivity Analysis
ΔCDF_{AVE}	< 1.0E-06/yr	2.40E-07/yr. (24%)	2.41E-07/yr. (24%)	2.22E-07/yr. (22%)	2.23E-07/yr. (20%)
$ICCDP^{*}$	< 5.0E-07	3.61E-07 (72%)	3.62E-07 (72%)	3.63E-7 (73%)	3.64E-07 (73%)
$\Delta LERF_{AVE}$	< 1.0E-07/yr.	1.20E-08/yr. (12%)	1.20E-08/yr. (12%)	8.65E-09/yr. (8.7%)	8.65E-09/yr. (8.7%)
$ICLERP^{*}$	< 5.0E-08	1.82E-08 (36%)	1.82E-08 (36%)	1.54E-08 (31%)	1.54E-08 (31%)

*These values are for Train A EDG which provides the limiting values for the Risk Metrics. Removal from service of Train A EDG has greater risk impact than Train B EDG due to the functional dependence of Train A Auxiliary Feedwater Pumps on AC electric power and use of diesel driven pumps on Train B. This is the reason why RAW values for Train A EDG are always greater than corresponding Train B EDGs.

**Note: Small differences in values between original evaluation and sensitivity evaluation in the above table are due to round-off errors, and the results virtually the same.

(5)

U.S. Nuclear Regulatory Commission

RECENT ACTIVITIES ON ASME REVIS!ON 12 PRA STANDARD

Presented to
Advisory Committee on Reactor Safeguards

Presented by
Mary Drouin, Gareth Parry

October 5, 2000

ASME PRA STANDARD

- June 14, 2000, ASME issued Revision 12 of "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" for public review and comment
- NRC provided comments to ASME on August 14, 2000
- NRC comments were based on using the criteria in SECY-00-0162 and comments provided on Revision 10

STAFF CONCLUSIONS

- Not a standard that addresses PRA quality
- Difficult to use in determining where there are weaknesses and strengths in the PRA results and therefore will have limited use in the decision-making process
- Only provides limited assistance to the staff in performing a more focused review of licensee PRA submittals
- Provides minimal assistance in making more efficient use of NRC resources

MAJOR STAFF COMMENTS

- Section 1, Introduction
  - No specific application fits under a single category, therefore, the different categories are not very helpful
- Section 2, Definitions
  - Many are inaccurate, are not written for the context in which they are used, and are unnecessary
- Section 3, Risk Assessment Application Process
  - Does not provide any requirements and excludes any minimum standard

MAJOR STAFF COMMENTS (cont'd)

- Section 4, Risk Assessment Technical Requirements
  - Flaws in completeness, accuracy, logic, organization and structure of the technical requirements
- Section 5, PRA Configuration Control
  - A strength of the standard
- Section 6, Peer Review
  - Insufficient focus on the need for the reviewers to make value judgments on the appropriateness of the assumptions and approximations made in the PRA, and an assessment of their impact on the results

RECENT ACTIVITIES

- ASME appointed Task Group to look at Revision 12 and provide advice
- Staff proposed set of principles and objectives for the standard which were used by the ASME Task Group
- Task Group met on September 19&20, 2000
- Task Group briefed ASME, NRC's PRA Steering Committee and NEI's Risk-Informed Regulation Working Group on September 21, 2000

PRINCIPLES AND OBJECTIVES

In the risk-informed environment in which NRC and industry are currently operating, PRA results are used as one, but not the only input to a decision-making process. Depending on the specific nature of the application, PRA results can play a more or less significant role. The extent to which the PRA results influence the decision will be impacted by the confidence the decision-makers have in those results. Accordingly, development of a Standard that promotes a consistent determination of the strengths and weaknesses of a PRA will directly impact the ability of decision-makers to efficiently establish a level of confidence in the results. The requirements of such a Standard provide a reference point for determining the strengths and weaknesses and also for evaluating alternative PRA approaches. The Standard should also recognize that in some areas methodology and data enhancements will occur over the next several years.

1. The PRA Standard needs to provide well-defined criteria against which to judge the strengths and weaknesses of the PRA so that decision-makers can determine the degree of reliance that can be placed on the PRA results of interest.

2. The Standard needs to be based on current good practices as reflected in publicly available documents. The need for the documentation to be publicly available follows from the fact that the Standard may be used to support safety decisions.

3. To facilitate the use of the Standard for a wide range of applications, categories can be defined to aid in determining the applicability of the PRA for various types of applications.

4. The Standard needs to be thorough and complete in defining what is technically required and should, where appropriate, identify one or more acceptable methods.

PRINCIPLES AND OBJECTIVES (cont'd)

5. The Standard needs to require a peer review process that identifies and assesses where the technical requirements of the Standard are not met. The Standard needs to assure that the peer review process:
   a. determines whether methods identified in the Standard have been used appropriately;
   b. determines that, when acceptable methods are not specified in the Standard, or when alternative methods are used in lieu of those identified in the Standard, the methods used are adequate to meet the requirements of the Standard;
   c. assesses the significance on the results and insights gained from the PRA of not meeting the technical requirements in the Standard;
   d. highlights assumptions that may significantly impact the results and provides an assessment of the reasonableness of the assumptions;
   e. is flexible and accommodates alternate peer review approaches; and
   f. includes a peer review team that is comprised of members who are knowledgeable in the technical elements of a PRA, are familiar with the plant design and operation, and are independent with no conflicts of interest.

6. The Standard needs to address the maintenance and update of the PRA to incorporate changes that can substantially impact the risk profile, so that the PRA adequately represents the current as-built and as-operated plant.

7. The Standard needs to be viewed as a living document. Consequently, it should not impede research but needs to be structured such that when improvements in our state of knowledge occur, the Standard can easily be updated.

ASME APPOINTED TASK GROUP

From NRC:

- Mike Cheok
- Mary Drouin
- Gareth Parry
- Nathan Siu

From Industry:

- Bob Budnitz
- Dave Bucheit
- Jim Chapman
- Greg Krueger
- Doug True

TASK GROUP CHARGE AS STATED BY ASME

Evaluate Principles and Objectives and provide conclusions and recommendations on the following:

1. Is it possible and/or appropriate for the standard to meet each objective?
2. To what extent does Draft 12 of the standard meet each objective?
3. Identify the critical technical issues associated with as many technical elements as possible.
4. Propose resolutions for the issues identified in (3) and provide examples of changes that could be made affecting structure and organization of the technical elements

GENERAL CONCLUSIONS AS STATED BY ASME TG

- The stated principles/objectives for the standard are appropriate and it is possible to meet them.
- While the content of Draft 12 addresses many of these objectives, problems exist in several areas. These are more specifically identified in the detailed comments.
- Draft 12 should and can be modified to be acceptable to the stakeholders represented by the TG.

DETAILED OBSERVATIONS AND RECOMMENDATIONS AS STATED BY ASME TG

- The current Objective Statements for the technical elements do not provide a clear description of the overall objective for each element and they are not always consistent with the High Level Requirement (HLR) Statements
  - ***Recommendation:*** *provide clear description*
- The HLRs should be logically related to the Objective Statements
  - ***Recommendation:*** *define HLRs that are logically related*
- The SRs should fully implement the HLRs.
  - ***Recommendation:*** *specify minimum set of Srs that fully implement HLRs, particular attention to level of detail for data and quantification*
- In general, the level of detail in the supporting requirements (SRs) is sufficient to capture most of the technical issues required to meet the HLRs. Exceptions to this conclusion are:
  - Data Section is incomplete
  - Quantification section is too detailed.
  - ***Recommendation:*** *see above*

DETAILED OBSERVATIONS AND RECOMMENDATIONS AS STATED BY ASME TG (cont'd)

- The SRs should address certain technical topics which are important to risk and where a consensus methodology does not currently exist
  - A few missing issues need to be identified (e.g., BWR ATWS, Consequential SGTR, dual unit initiators, etc.)
  - ***Recommendation:*** *address risk important technical topics with requirement for documentation of approach, assumptions and significance*
- The clarity of some SRs needs to be improved
  - ***Recommendation:*** *examples -- replace "to the extent necessary to support category x applications," the word 'may' not be used as a lead statement and not use as a permissive, term 'consider' should be defined an usage limited*
- The current definitions for the categories are not clear and are not adequate to help formulate SRs.
  - Specific applications may span categories; therefore categories can not be defined by applications
  - ***Recommendation:*** *need clear definition, TG proposed criteria and definitions*
- Consistency in and between categories and technical elements needs to be improved
  - ***Recommendation:*** *improve consistency, etc., the SRs should define lower limit of acceptability for each category*

DETAILED OBSERVATIONS AND RECOMMENDATIONS AS STATED BY ASME TG (cont'd)

- Section 6 - Peer Review
  - Needs enhancement with respect to methodology and documentation.
  - Should clarify that Peer Review is a process applied to evaluate the PRA and not to review specific applications of the PRA
  - ***Recommendation:*** *examples -- term 'methodology' needs clarification, methodology requirements shoudl be enhanced*
- Section 3 - The Application Process generally describes how the standard could be used in decision making processes involving the application of a PRA
  - More detail would be necessary to make this process work, but it is not appropriate to include this level of detail in the standard at this time
  - ***Recommendation:*** *modify to clarify that it is an overall process for application of a PRA in conjunction with the requirements of the standard*
- Additional references would be useful
  - ***Recommendation:*** *add references that are sources of information to be used for explanation*
- Section 2, Definitions, needs improvement
  - ***Recommendation:*** *provide clear and accurate definitions*

RECOMMENDED FUTURE ACTIONS BY ASME TG

- TG to provide input on several recommendations; examples:
  - Write objective statement for each element
  - Modify the HLRs
  - Identify missing technical topics
  - Define categories
  - Identify suggested references
- Project Team should initiate review and resolution of comments on Section 2, 3, 5 and 6
- “Small” group should organize and edit Section 4 according to principles and objectives

FUTURE STAFF ACTIONS

- Continue to work with ASME and ANS
- Complete “acceptance criteria” for endorsement of ASME or ANS standards

Status of the FAVOR Code Development

Terry Dickson, Richard Bass, and Paul Williams
Heavy-Section Steel Technology Program
Oak Ridge National Laboratory

Advisory Committee on Reactor Safeguards
Pressurized Thermal Shock Screening Criterion Re-evaluation

October 5, 2000
Nuclear Regulatory Commission
Rockville, Maryland

This Presentation Describes the Evolution of an Advanced Computational Tool for RPV Integrity Evaluations : FAVOR

1/ How FAVOR is Applied in PTS Re-evaluation

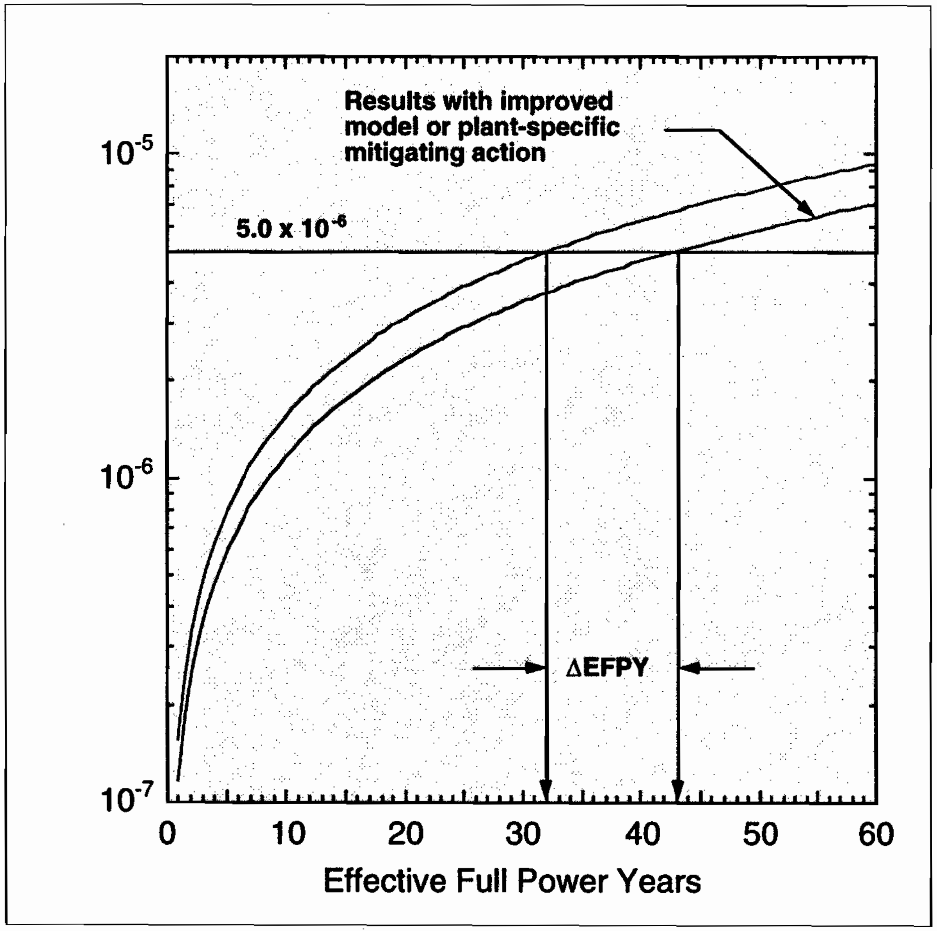
2/ Integration of Evolving Technology into FAVOR

3/ FAVOR Structure

4/ Overall PRA Methodology

5/ PFM Details

Application of FAVOR to PTS Re-evaluation Addresses the Following Two Questions



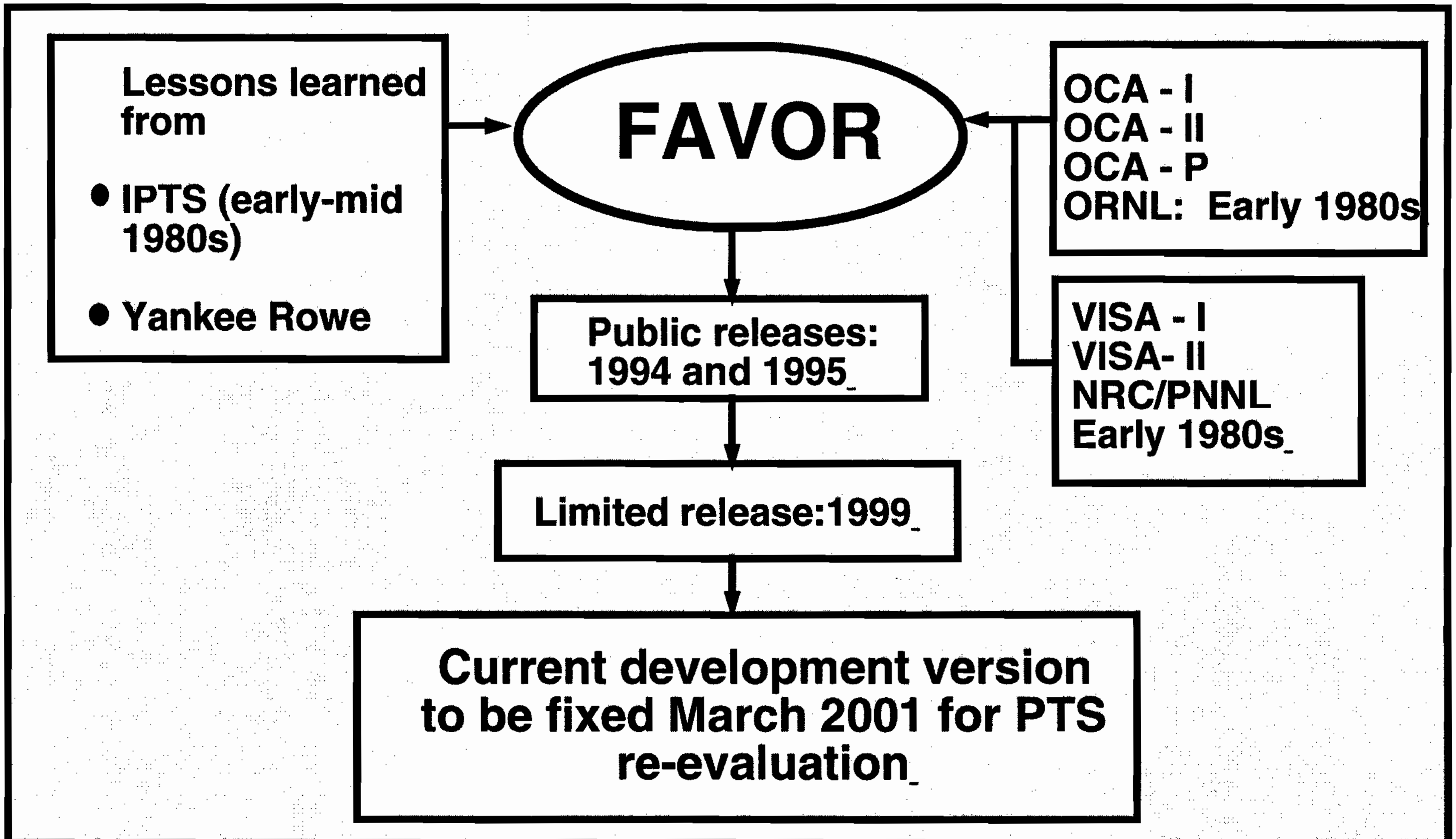
- At what time in operating life does frequency of RPV failure exceed acceptable value (currently 5 x 10e-06)?
- How does integration and application of advanced technology affect the calculated result?

Near-Term Schedule for Development of the FAVOR Code has been Defined

- **Current schedule specifies FAVOR to be ready for PTS re-evaluation analyses on March 1, 2001**

- **In the interim period:**
  - **models are being finalized**
  - **finalized models are being implemented**
  - **scoping studies are being performed**

Development of the FAVOR Code was Initiated in Early 1990s by Combining Best Attributes of OCA/VISA with Evolving Technology



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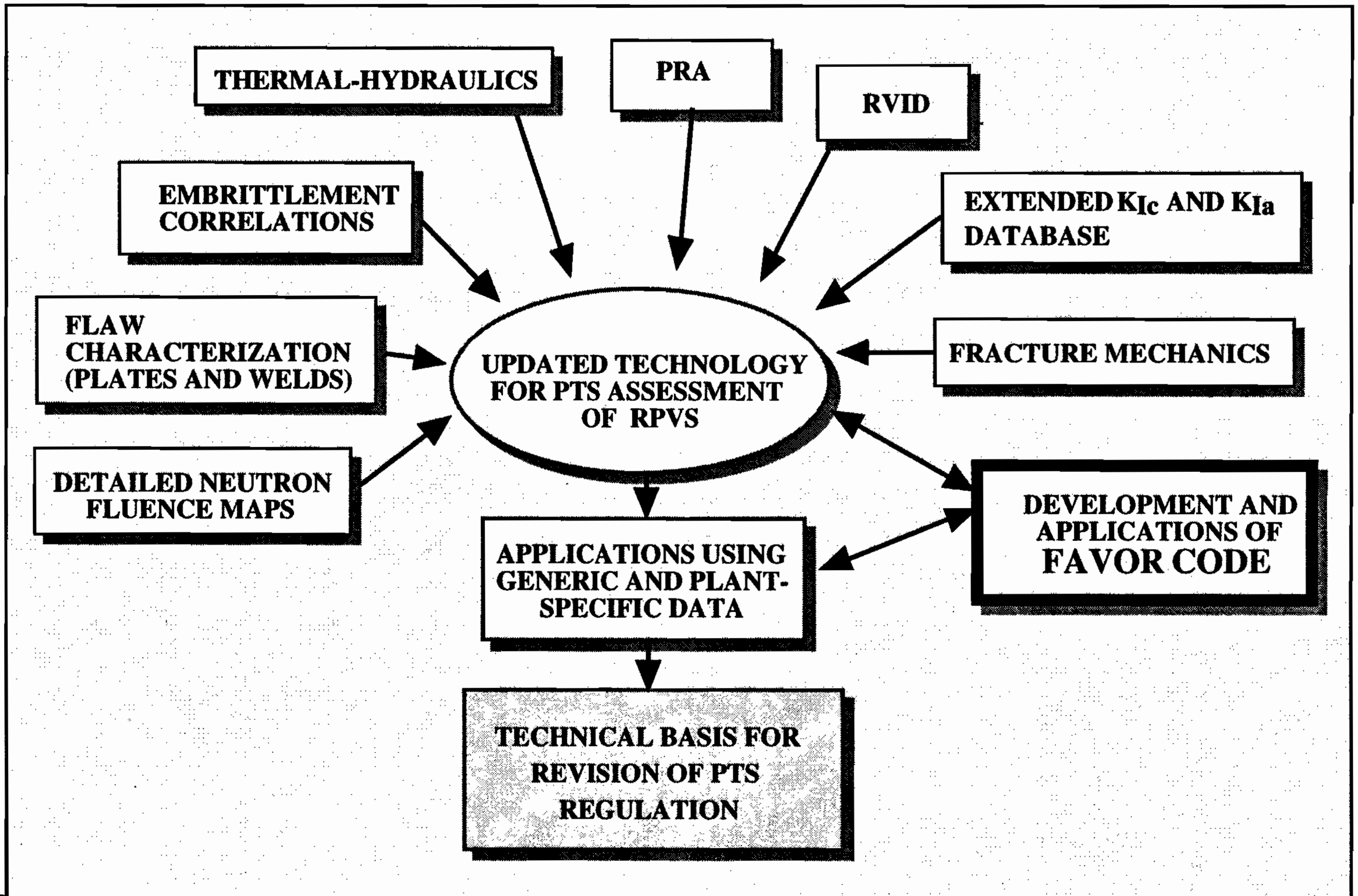
Elements of updated technology are currently being integrated into the FAVOR* computer code to re-examine the current PTS regulations
*(Fracture Analysis of Vessels: Oak Ridge)



Advanced Technology is Integrated into FAVOR to Support Possible Revision of PTS Regulation

- Flaw characterizations from NRC research (plates and welds)
- Detailed fluence maps
- Embrittlement correlations
- RVID
- Fracture toughness models
- Surface-breaking and <u>embedded</u> flaws
- Inclusion of through-wall weld residual stresses
- New PFM methodology

A Significant Improvement Since the Derivation of Current PTS Regulations* is Flaw Characterization

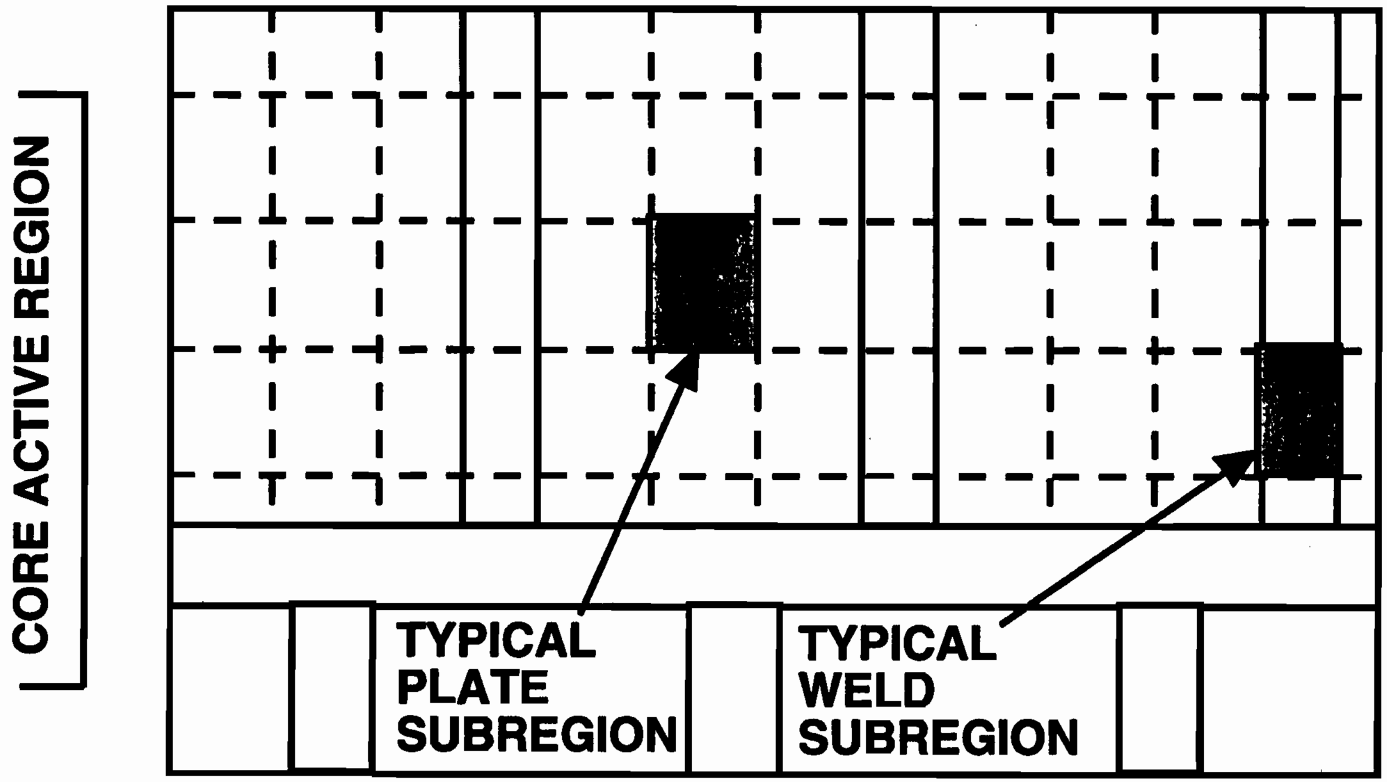
** analyses assumed all flaws were inner-surface breaking flaws*

- Recent NDE and DE of RPV material at PNNL has established an improved technical basis for flaw-related data used as input for PFM analyses

- A significantly higher number of flaws were found than was postulated in PFM analyses from which current PTS regulations were derived; however, all flaws detected thus far are embedded

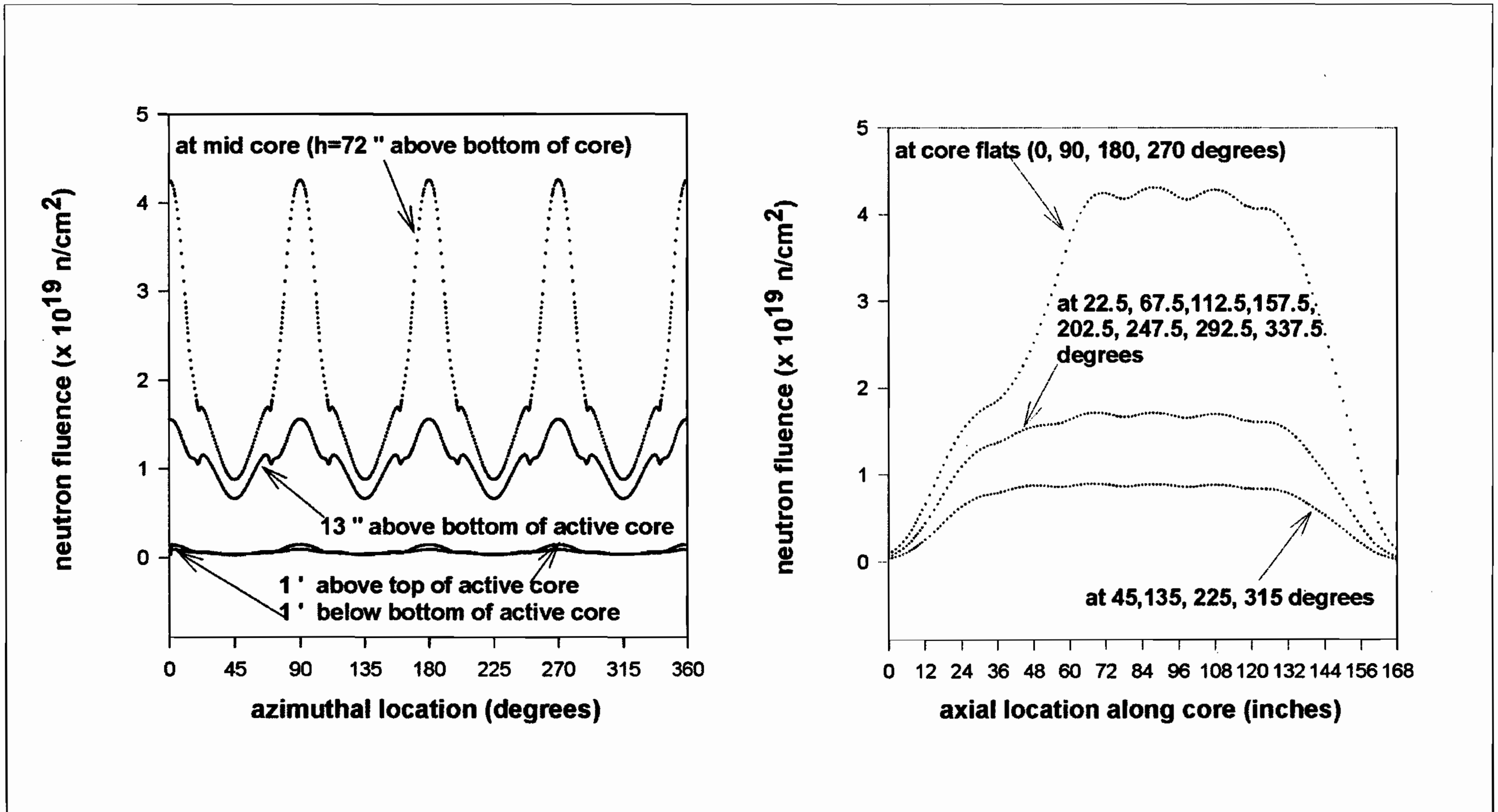
- Application of PVRUF flaw densities to commercial PWR results in over 3500 flaws in 1st 3/8 thickness of RPV wall

FAVOR utilizes a methodology that allows the RPV beltline to be discretized into sub-regions, each with its own distinguishing embrittlement-related parameters. This accommodates chemistries from the RVID and detailed neutron fluence maps

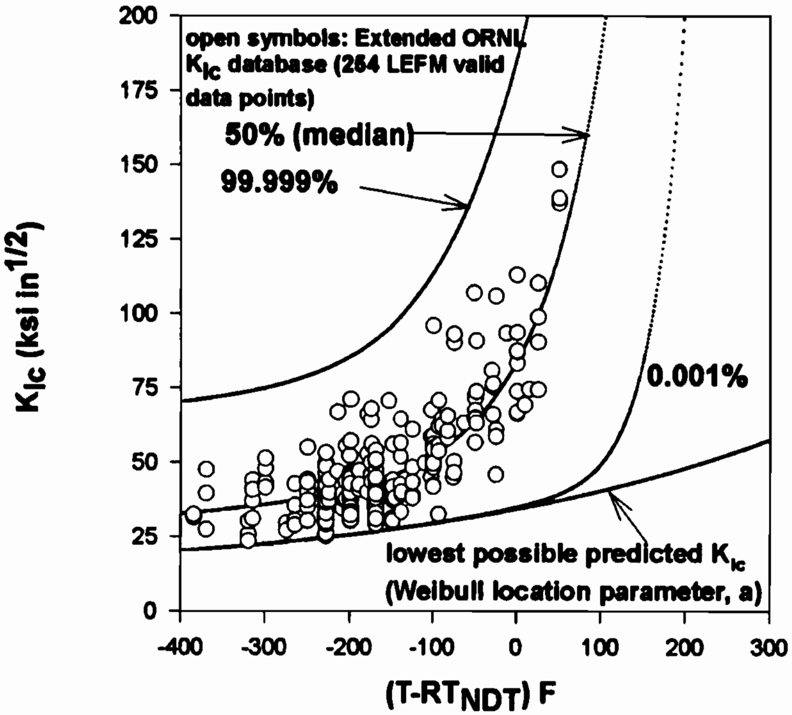


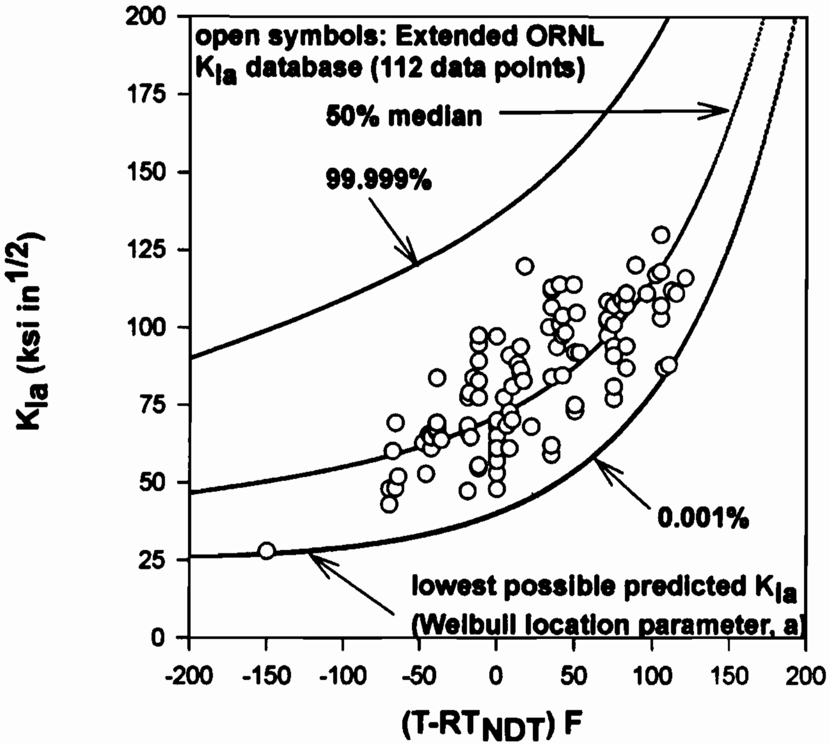
TYPICAL VESSEL BELTLINE LAYOUT

Brookhaven National Laboratory is generating very detailed neutron fluence maps for selected PWRs corresponding to 32 EFPY and 40 EFPY

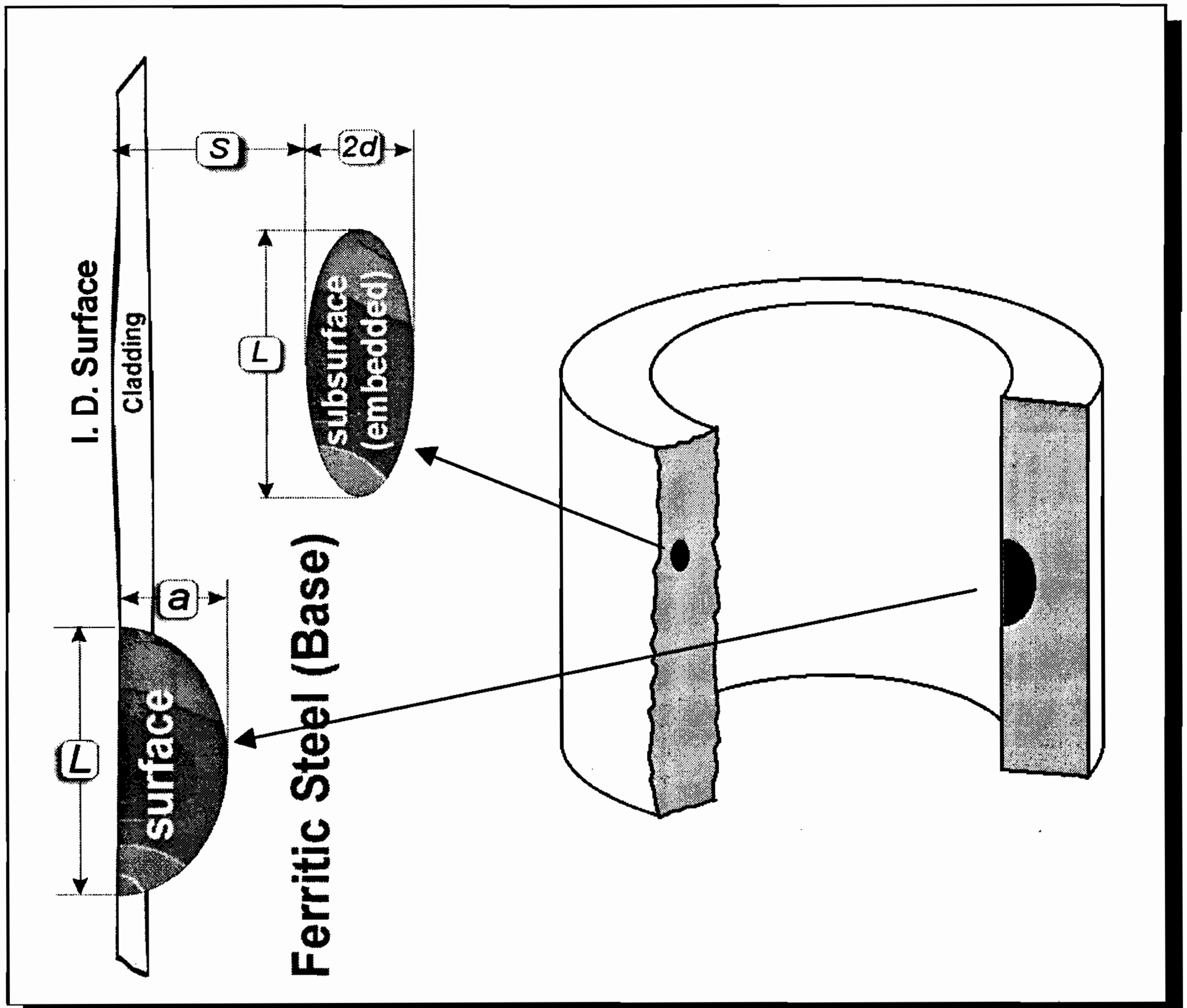


New Statistical Models for Enhanced Plane-Strain Static Initiation (K_{Ic}) and Arrest (K_{Ia}) Fracture Toughness Data Bases were Implemented into FAVOR





The FAVOR PFM Model Now Includes Inner Surface-Breaking and/or Embedded Flaws



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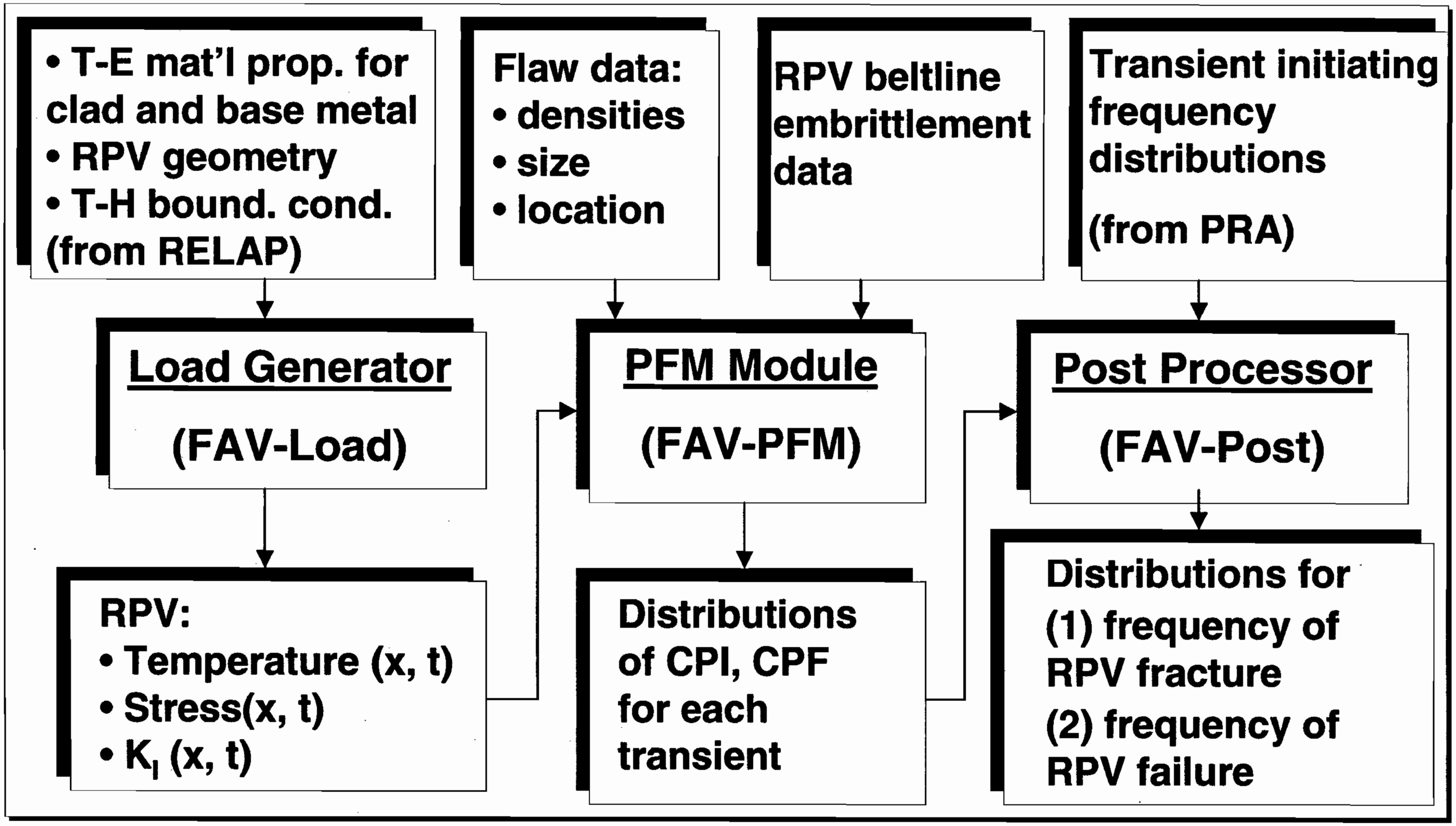
The Current FAVOR Code Consists of Three Separate Modules



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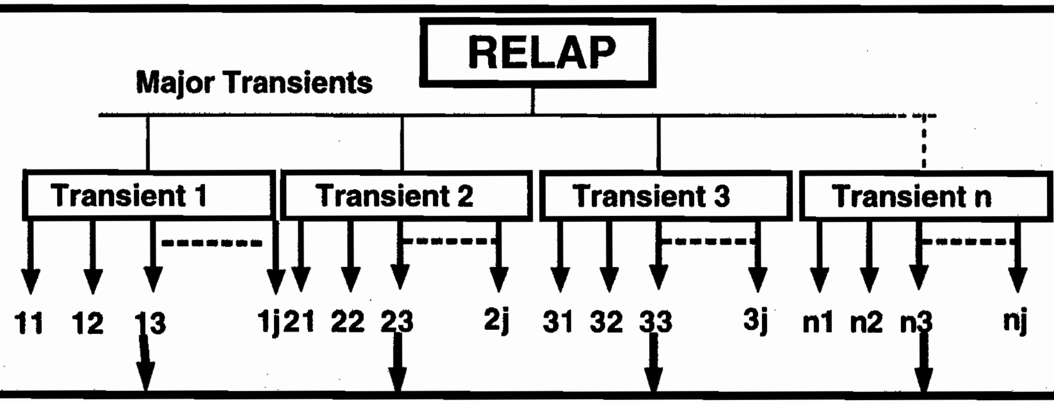
FAVOR Analyses Incorporate Uncertainty Associated with Thermal Hydraulics by Including Variants for Each of the Transients



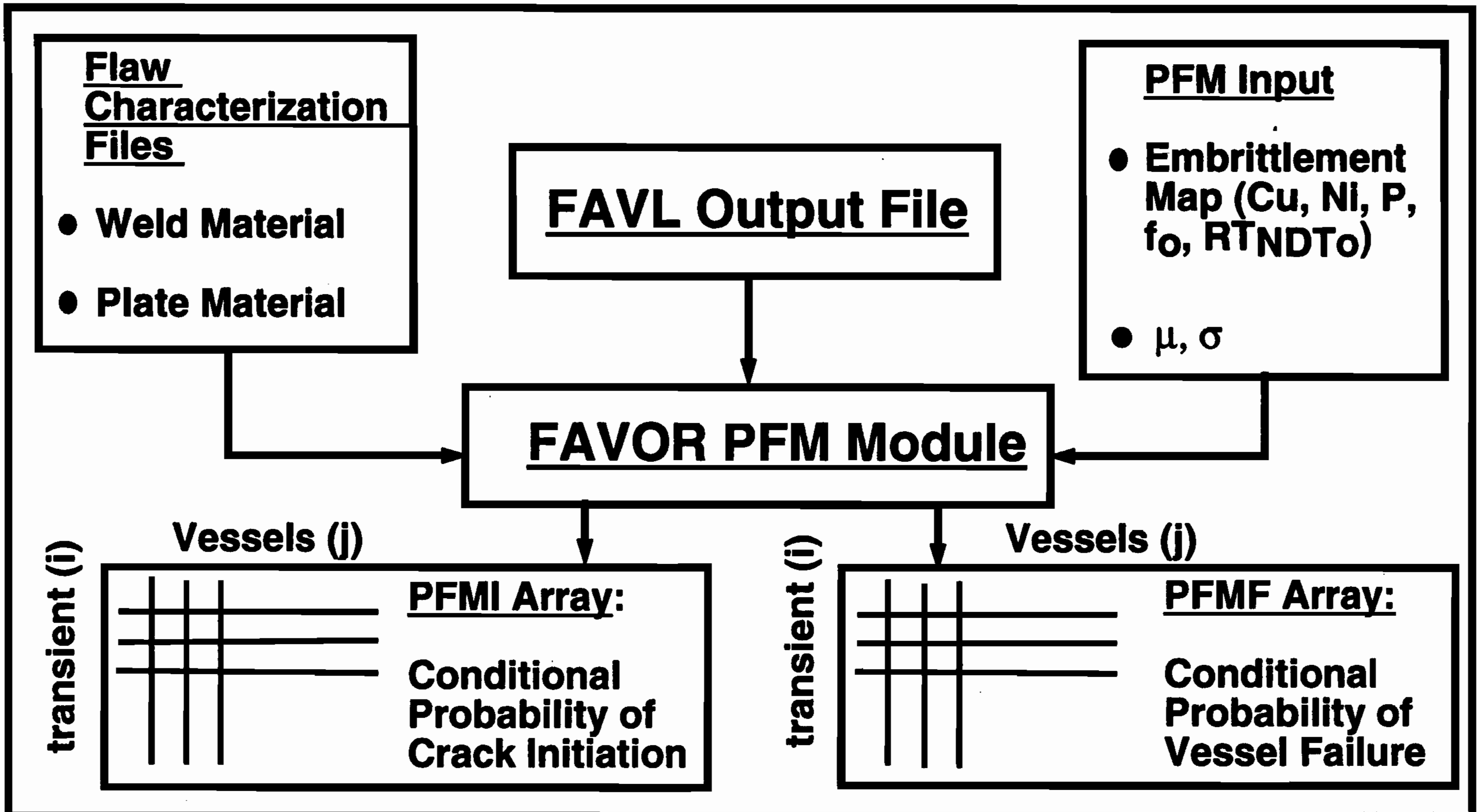
FAVOR Load Generator (FAVL)

One-dimensional axisymmetric finite-element analyses are performed to calculate RPV loads for each transient

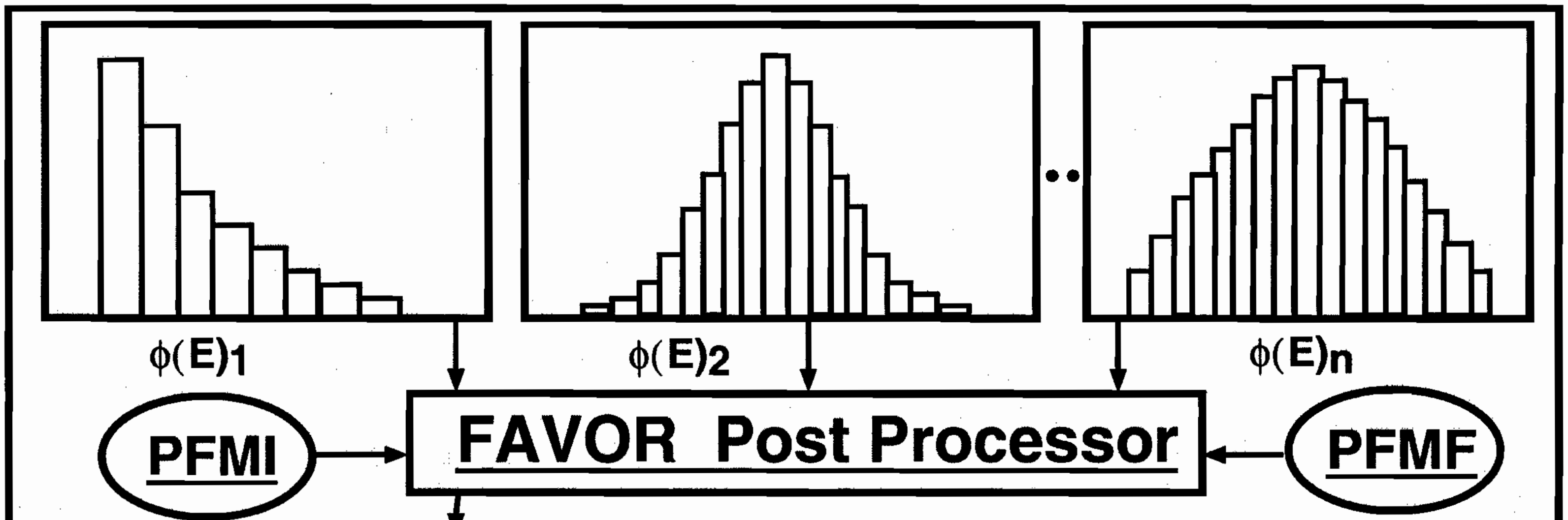
Output File from FAVOR Load Generator

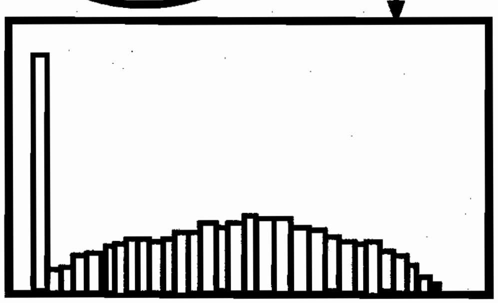
- temperature	$T(r,t)$
- circumferential stress	$\sigma_H(r, t)$
- axial stress	$\sigma_A(r, t)$
- SIF (inner surface-breaking flaws)	$K_I(a, l, t)$

The FAVOR PFM Analysis Module Generates Arrays Containing Conditional Probabilities of Initiation (PFMI) and Failure (PFMF) for Vessel(j) Subjected to Transient(i)



The FAVOR Postprocessor Module Integrates the Uncertainties of the Transient Initiating Frequencies with the PFMI and PFMF Arrays to Generate Distributions for the Frequencies of RPV Fracture and RPV Failure





ϕ(F) (Frequencies of RPV fracture/failure)

For each vessel:

1. Sample initiating frequencies, $\phi(E)_1, \ldots \phi(E)_n$
2. Combine $\phi(E)$ with PFM results, $\phi(F)_i = \Sigma\phi(E)_T \bullet P_{FM}(T,V)$
3. Generate histogram for $\phi(F)$ from resulting array of $\phi(F)_i$

Near-Term Schedule for Development of the FAVOR Code has been Defined

- **Current schedule specifies FAVOR to be ready for PTS re-evaluation analyses on March 1, 2001**

- **In the interim period:**
  - **models are being finalized**
  - **finalized models are being implemented**
  - **scoping studies are being performed**

(7)

Fracture Toughness Distributions and Uncertainty Analysis

→ Status Report to ACRS ←

Mark Kirk
RES/DET/MEB

Rockville, Maryland
September 21, 2000

VG 1

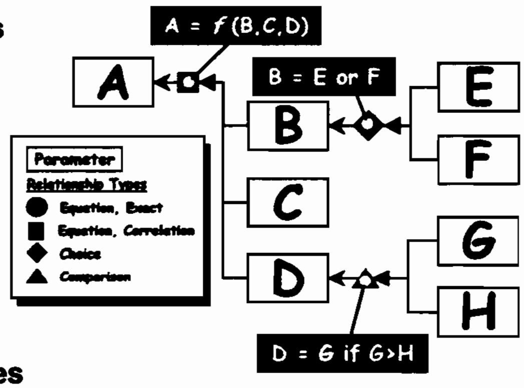
Overview

- Goal and participants in cooperative effort
- Approach
- New data
- Uncertainty framework
- Current results
- Summary and future work

VG 2

Root Cause Analysis

- Describe uncertainties
  - Data
  - Model / equation
  - Expert judgements
- Classify uncertainties
  - Aleatory
    - ✓ Non-reducible
  - Epistemic (state-of-knowledge)
    - ✓ Reducible
- Propagate uncertainties through model



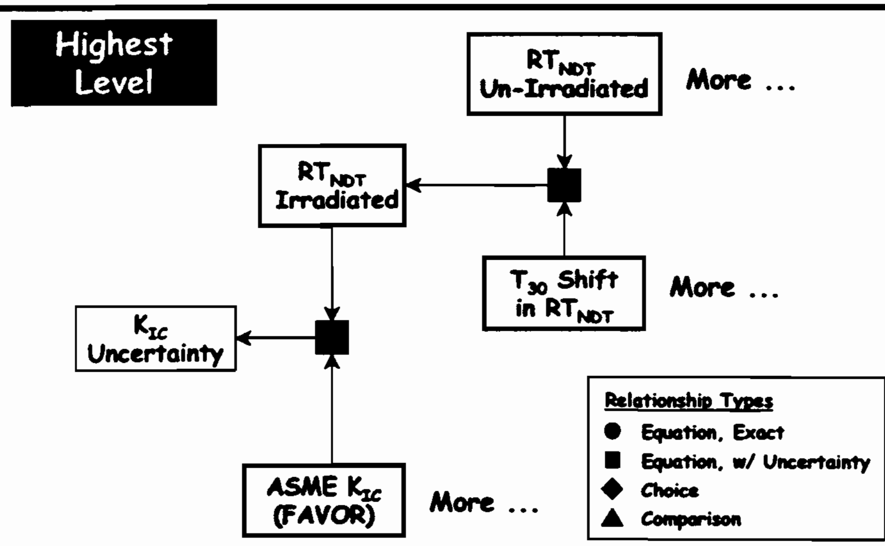
[Natishan & Modarres, 2000]

The BIG Change from the Old Ways

Input uncertainties propagate to output uncertainties via a systematic and critiquable process, rather than margins being prescribed to the analysis *a priori*.

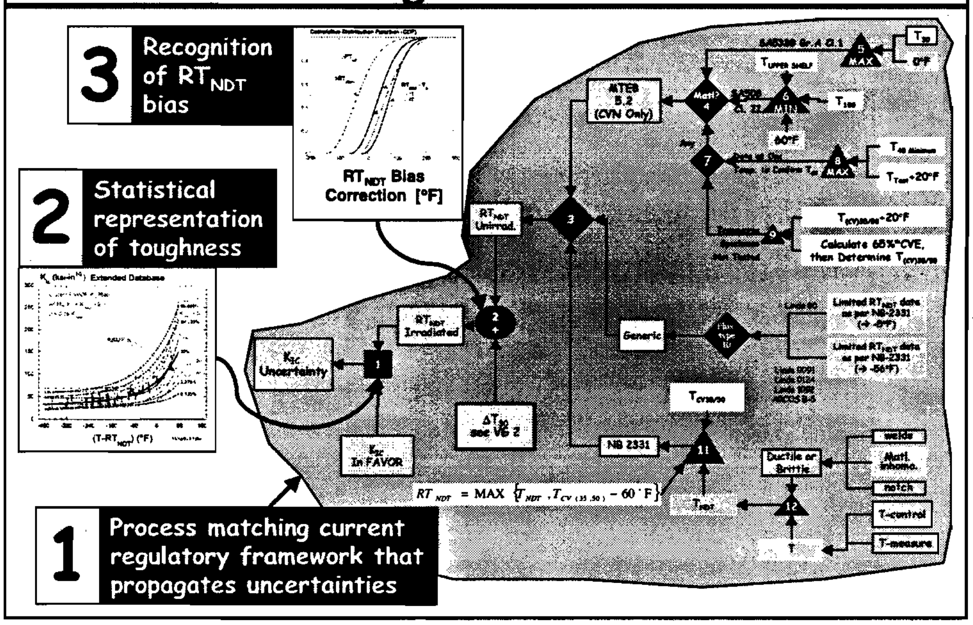
VG 7

Root Cause Analysis of K_{IC} / RT_{NDT} Process

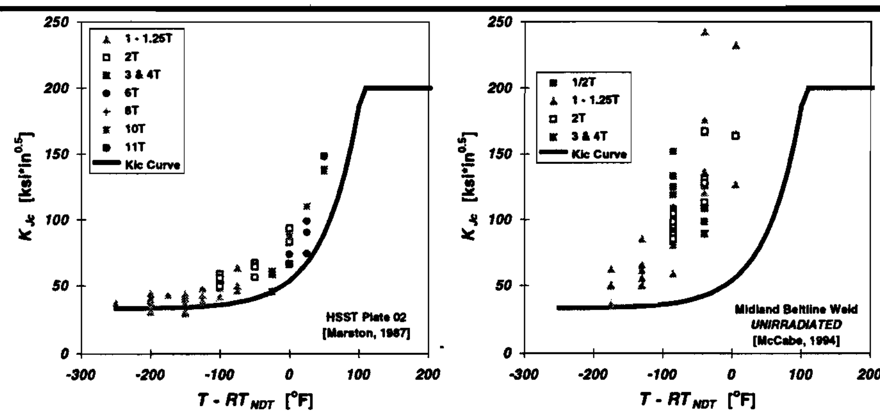


VG 8

Root Cause Analysis of K_{IC} / RT_{NDT} Process → New or Significant Features ←



Bias in RT_{NDT}



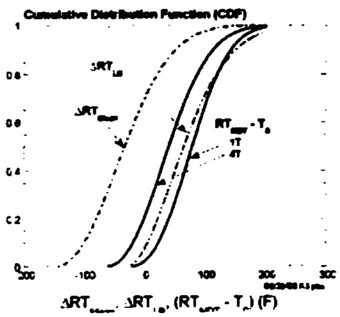
Expected since RT_{NDT} designed to be a bounding estimate of transition temperature, but ...

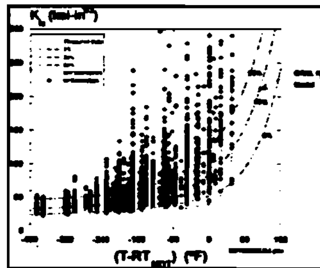
Inconsistent with a PRA approach that relies on "best estimates."

VG 10

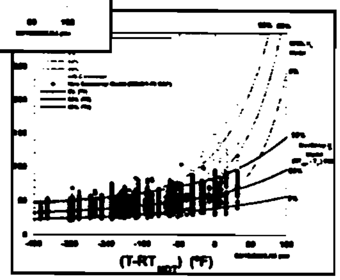
Bias in RT_{NDT}

Candidate Correction Functions





Different Correction Procedures



Appropriate correction and procedure to correct still being developed.

VG 11

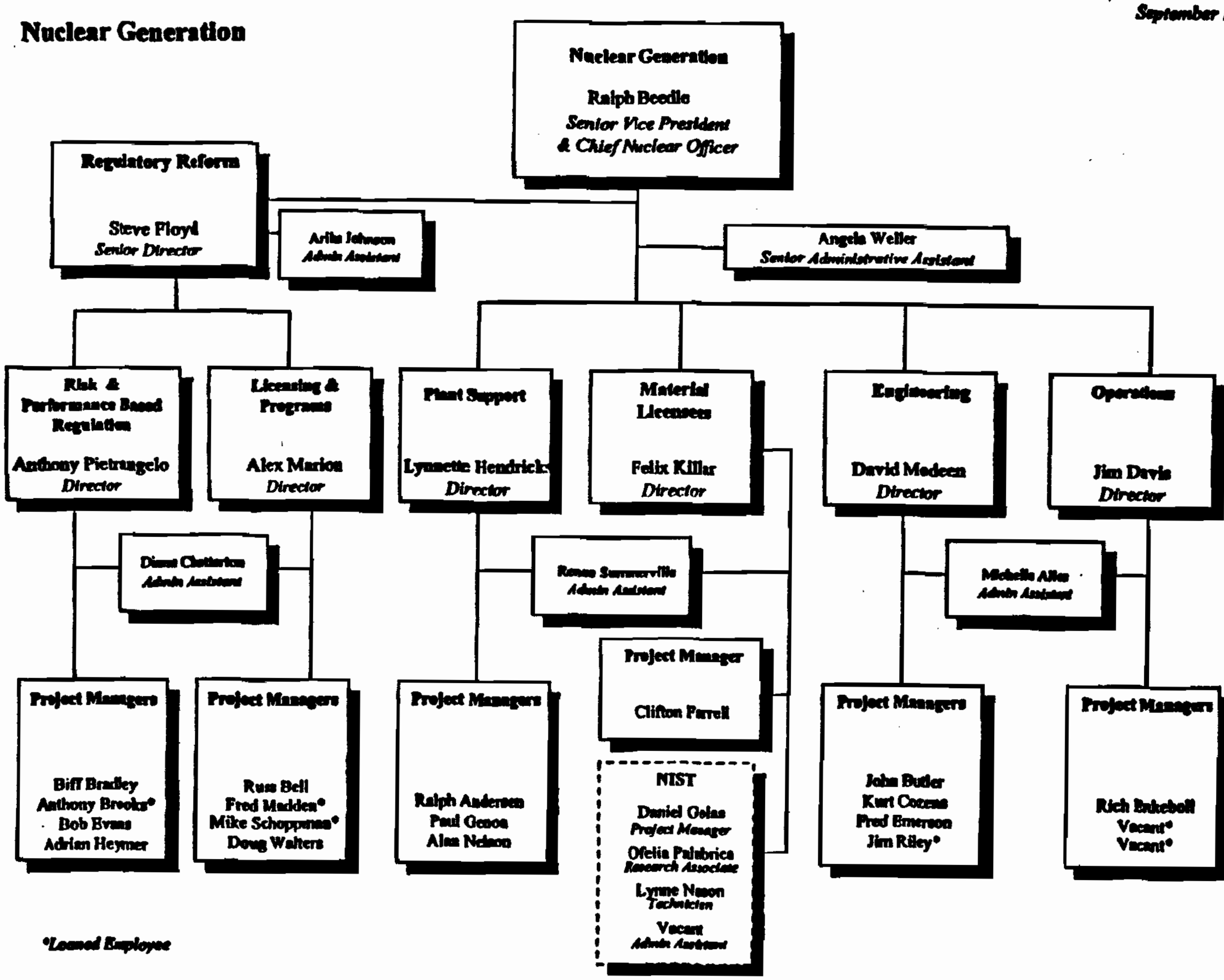
Summary and Future Work

- Completed
  - Statistical transition fracture toughness model
    - ✓ Data collected
    - ✓ Fit completed
  - PRA-uncertainty framework
    - ✓ Current processes understood using root-cause diagram approach
    - ✓ Mathematical models developed
      - ➢ Details of FAVOR implementation discussed / clarified
- On-going
  - Full implementation of uncertainty model into FAVOR
  - Resolution of RT_{NDT} bias correction function and modeling procedure
  - Assembly of input data to run the models

VG 12

Nuclear Generation

September 12, 2000



*Loaned Employee

(8)

United States Nuclear Regulatory Commission

GSI-168

Environmental Qualification (EQ) of Low-Voltage I&C Cables

Presentation To Advisory Committee on Reactor Safeguards

Office of Nuclear Regulatory Research

M. Mayfield, E. Hackett, S. Aggarwal, Division of Engineering Technology

M. Cunningham, Division of Risk Analysis and Applications

October 6, 2000

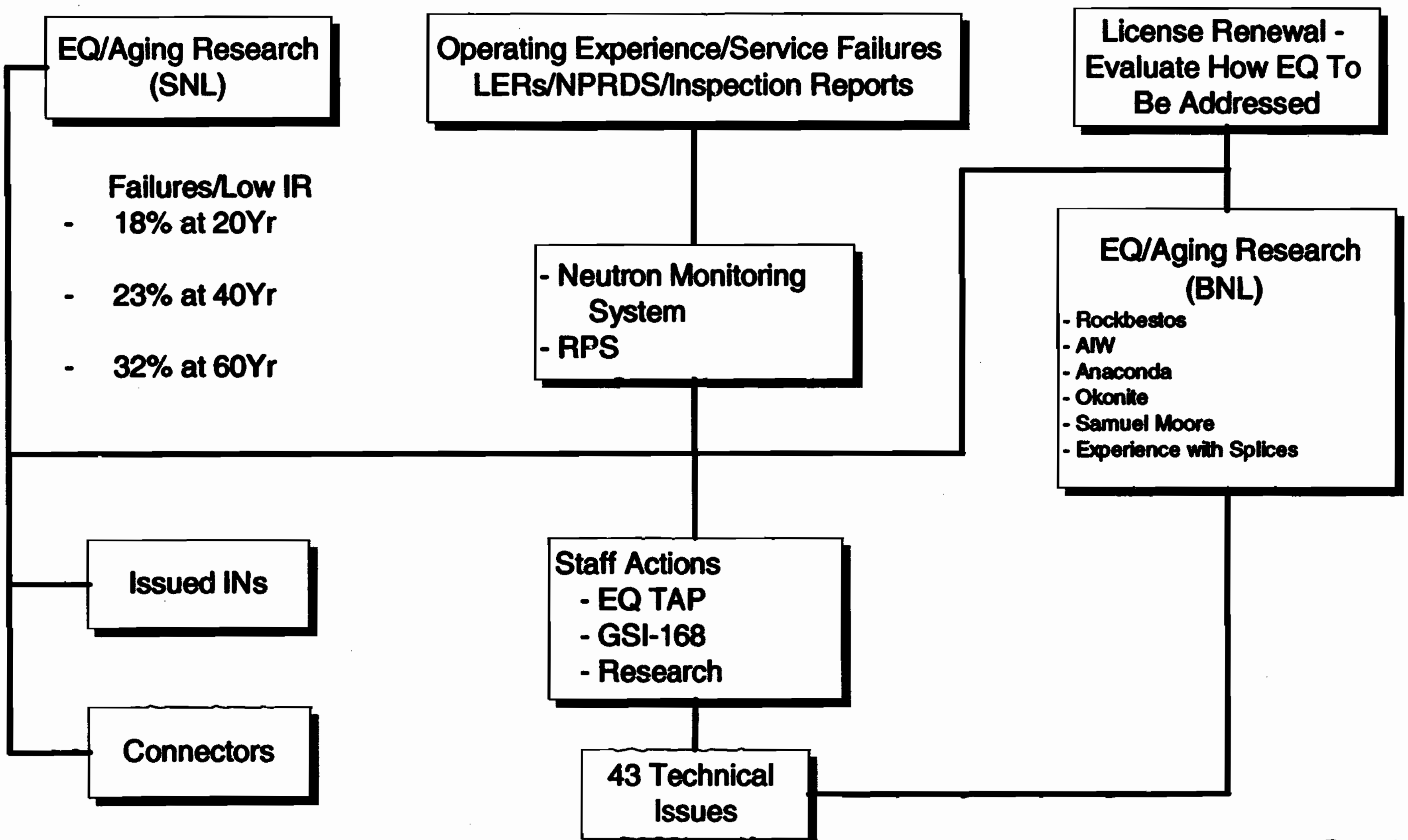
GSI-168
EQ of Low-Voltage I&C Cables

❑ Purpose

- To present to the ACRS technical and regulatory background and research results for resolution of GSI-168

Environmental Qualification (EQ) of Electric Equipment

Background



ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT SANDIA NATIONAL LABORATORY RESEARCH RESULTS

SUMMARY OF TESTED (T), FAILED (F)[a] AND MARGINAL (M)[b] CABLES

Cable Type	20 YR.: T - F/M	40 YR.: T - F/M	60 YR.: T - F/M
Rockbestos XLPE/Neoprene (3/C)	3 - 0/0	3 - 0/0	6 - 1/0
Brand Rex XLPE/CSPE (3/C)	3 - 0/0	3 - 0/0	3 - 0/0
Samuel Moore XLPO/CSPE (3/C)	3 - 0/0	3 - 0/0	6 - 0/0
Raychem XLPE (1/C)	2 - 0/0	2 - 0/0	3 - 0/0
Anaconda Unbonded EPR/CSPE (3/C)	6 - 0/0	6 - 0/0	6 - 0/0
Anaconda EPR/CSPE (1/C)	1 - 0/0	1 - 0/0	1 - 0/0
BIW Bostrad EPR/CSPE (2/C)	2 - 0/2	2 - 0/2	4 - 0/4
BIW Bostrad EPR/CSPE (1/C)	2 - 0/1	2 - 0/2	2 - 0/2
Okonite Bonded EPR/CSPE (1/C)	3 - 0/0	3 - 0/0	4 - 1/0
Samuel Moore Bonded EPDM/CSPE (2/C)	4 - 1/0	4 - 0/0	4 - 2/0
Samuel Moore Bonded EPDM/CSPE (1/C)	2 - 0/0	2 - 0/0	2 - 0/0
Kerite (1/C)	2 - 0/0	2 - 0/2	3 - 0/3
Rockbestos Coaxial (1/C)	2 - 0/2	2 - 0/2	2 - 0/2
Rockbestos SR/Fiberglass (1/C)	2 - 0/0	2 - 0/0	2 - 0/0
Champlain Kapton (1/C) [c]	2 - 1/0	2 - 1/0	2 - 1/0
Totals	39 - 2/5 (18 %)	39 - 1/8 (23 %)	50 - 5/11 (32 %)

(a) Failure criterion = 1 amp fuse in circuit blown.

(b) Marginal criterion = minimum IR lower than 2,500 ohm-1,000 ft. for instrument cable, 500 ohm-1,000 ft. for control cable, or 10^7 ohm-1,000 ft. for coaxial cable.

(c) Failed cables in 40 and 60 tests were damaged prior to accident test.

ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT BROOKHAVEN NATIONAL LABORATORY RESEARCH RESULTS

SUMMARY OF TESTED (T), FAILED (F)[a] AND MARGINAL (M)[b] CABLES

Cable Type (see Note)	20 YR.: T - F/M	40 YR.: T - F/M	60 YR.: T - F/M
Rockbestos XLPE/Neoprene (2/C)	5 - 0/0	5 - 1/3	3 - 1/2
American Insulated Wire Unbonded EPR/CSPE (3/C)	2 - 0/0	-	3 - 3/0
American Insulated Wire Unbonded EPR/CSPE (4/C)	3 - 0/0	-	-
Anaconda Unbonded EPR/CSPE (3/C)	2 - 0/0	5 - 0/5	-
Anaconda Unbonded EPR/CSPE (1/C)	-	2 - 0/0	-
Samuel Moore Bonded EPDM/CSPE (2/C)	4 - 0/2	5 - 2/3	3 - 1/2
Samuel Moore Bonded EPDM/CSPE (1/C)	2 - 0/0	2 - 0/0	-
Okonite Bonded EPR/CSPE (1/C)	2 - 1/0	3 - 3/0	3 - 3/0
Totals (using footnote b criterion for "marginal")	20 - 1/2 (15 %)	22 - 6/11 (77 %)	12 - 8/4 (100 %)
Totals (using footnote c criterion for "marginal")	20 - 1/0 (5 %)	22 - 6/3 (41 %)	12 - 8/4 (100 %)

(a) Failure criterion = test specimen unable to hold 2,400 Vac test voltage during post-LOCA submerged voltage withstand test.
(b) Marginal criterion = test specimen leakage current > 1.2 milliamps during post-LOCA submerged voltage withstand test.
(c) Marginal criterion = test specimen leakage current > 2.0 milliamps during post-LOCA submerged voltage withstand test.

NOTE XLPE and EPR insulated cables only were tested since they represent > 70 % of the cables currently installed in commercial nuclear power stations.

Operating Experience

- 87 Events identified from 1968 - 1992 related to age-related degradations of in-containment cables (EPRI report, July 1994).
- Problems with splices at operating power plants.
- Cable with highest number of reported problems is neutron monitoring systems.
- Cable failures due to high temperature and moisture intrusion (reported in LERs).

Risk Assessment Considerations

- Scoping studies have shown that the risk associated with aged instrumentation and control cables could be significant.
- An accurate assessment of the risk requires data on a number of plant-specific factors.

Information Required for Comprehensive Risk Assessment

- Potentially vulnerable cable systems
  - Specific function
  - Extent of aging
  - Susceptibility to environmentally-induced failure
  - Accident-induced environment
  - Time-dependent reliability
- Operator response to cable failures
  - Partial vs. complete failures
  - Misleading signals
  - Secondary indications

Summary

- Regulations do not require periodic inspection or monitoring of cables. Periodic TS surveillance tests evaluate operability of overall system but cannot evaluate aging and degraded state of cables and pending failures.
- Operating experience indicate some service induced degradation of safety-related cables, attributable to elevated temperature conditions and moisture intrusion.
- Research tests from both SNL and BNL/Wyle suggest some of the tested cable types would not function during a LOCA after 40 years of service. Most of them would not survive after 60 years of service, if they are operated at rated temperatures.

Note: Actual plant service conditions are generally less severe than the original parameters used for qualification.

Summary (cont'd)

- Risk studies give relatively high CDF values, conditioned on all of the cables failing during LOCA. Further, research data would not support a sufficiently low failure rate to reduce the CDF values to an acceptably low level.

- Risk studies are not definitive because of lack of detailed information about cables and operator actions due to misleading information. More data is needed for determining risk significance of cable aging.

Summary (cont'd)

- The research data strongly support that cable aging must be addressed through an aging management program for license renewal periods.

- The staff is still considering the resolution options for the current license term.

(10)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
476th	13	13-1

Title

GSI-168 (PRE-DECISIONAL)

Authors

RES

List of Documents Attached

13

T2 E17

Jit Singh (ACRS)
Jack Strosnider
Dick Wessman

PRE-DECISIONAL INFORMATION

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Generic Safety Issue GSI-168, "Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables"

September 29, 2000 -4:00 p.m.

Table of Contents (will be updated)

1. Background

Safety-related electric cables are used in a wide variety of applications in a nuclear power plant. Some of these cables are medium-voltage cables used to transmit electric power to safety-related electrical equipment; others are low-voltage cables used to transmit signals between instrumentation and control (I&C) devices (e.g., data and control signals) for performing safety functions.

Environmental qualification (EQ) requirements were developed to ensure that safety-related electric cables will perform their safety functions during their service life in the environment in which they operate during normal operation, transient conditions, and accidents. These requirements have evolved as operating experience accumulated and the aging process was better understood. Initially, qualification was based on the "high industrial quality" of electrical components. For plants constructed after 1971, a more formal approach was adopted: qualification was judged on the basis of Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1971, "IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations." Although IEEE Std. 323-1971 did not address aging or life determination issues, the standard did call for a systematic program of analysis, testing, and quality assurance. It specified that qualification may be achieved through type testing, analysis, operating experience, or a combination of these methods.

For plants with NRC construction permits dated July 1, 1974, or later, the Commission endorsed the 1974 version of IEEE Std. 323 as an acceptable standard for demonstrating EQ. At the time of release of this standard, the NRC considered backfitting older plants to meet IEEE Std. 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," but recommended against it because the incremental improvements provided by the new standard were not considered safety significant.

During the EQ rule-making process in 1982, the Commission again had the opportunity to require older plants to meet the latest standards. When the rule (10 CFR 50.49) was finalized, the Commission deemed older qualification methods acceptable (i.e., grandfathered them). The Commission stated that the new rule allowed older plants to use the Division of Operating Reactor (DOR) guidelines and NUREG-0588 (Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, 1981) Category II to qualify electric equipment. The Commission also recognized that the industry had just invested significant human resources and millions of dollars in conforming plants to the requirements of the DOR guidelines and NUREG-0588 Category II. Requiring the older plants to meet the newer EQ requirements (NUREG-0588 Category I) would invalidate the industry's efforts to comply with the staff's previous directions (to meet the requirements of the DOR guidelines and NUREG-0588 Category II).

Since older plants were not required to upgrade to the latest EQ requirements, the staff in effect sanctioned the use of three different sets of EQ requirements: (1) the DOR guidelines, which generally apply to equipment installed in plants that became operational before 1980; (2) the Category II criteria of NUREG-0588, which apply primarily to plants that became operational after 1980 and had originally committed to the requirements of IEEE Std. 323-1971; and (3) the Category I criteria of NUREG-0588 and Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Revision 1-1984),

 PRE-DECISIONAL INFORMATION

which apply to plants committed to meeting the requirements of IEEE Std. 323-1974 and to replacement equipment.

During license renewal activities in the late 1980s and early 1990s, the staff revisited the issue of EQ of safety-related electrical equipment, particularly the issue of whether EQ requirements for older plants were adequate for license renewal.

In SECY-93-049, "Implementation of 10 CFR 54 Requirements for Renewal of Operating Licenses for Nuclear Power Plants, March 1, 1993," the staff informed the Commission that, in evaluating the technical adequacy of the EQ requirements for license renewal, the staff had identified generic issues that might require backfits even for plants that were not renewing their licenses. Subsequently, in a June 28, 1993, memorandum from Samuel J. Chilk to James M. Taylor, the Commission directed the staff to treat EQ of electrical equipment in operating reactors as a potential safety issue and to periodically inform the Commission of the staff's progress in assessing the issue. On July 1, 1993, the staff submitted its EQ task action plan (EQ-TAP) as Enclosure 3 of the third quarterly report on fire protection issues. The purpose of the EQ-TAP was to evaluate and resolve existing environmental qualification concerns and to identify and resolve any other EQ concerns.

The EQ-TAP involved reviewing EQ-related information and conducting EQ-related research to enable the staff to (1) assess existing differences in the EQ requirements for older and newer plants, (2) assess the adequacy of accelerated aging practices for demonstrating equipment qualification, and (3) identify and resolve any other EQ issues. The plan included meetings with the industry, an EQ program review, data collection and analysis, a refined PRA, research on aging and condition monitoring, and options for resolving EQ concerns. These activities were modified as more information became available through research and a review of industry operating experience.

The Commission was informed of the status of the EQ-TAP by memoranda dated April 8, 1994, November 16, 1994, June 27, 1995, August 22, 1996, November 15, 1996, and February 5, 1998. In summary, the staff developed an EQ research program plan in 1995 to test cables, assess cable condition-monitoring methods, and develop and update an EQ database. The program was limited to low-voltage I&C cables since they were judged to be the most susceptible to aging degradation resulting in misleading information to control room operators. The information from the program was to be used to determine whether research should be done on other electric equipment. One of the initial steps in the program plan was to review information already available to avoid duplications and unnecessary research. In a memorandum to the Commission on November 15, 1996, the staff concluded, in part, that it would be prudent to have some form of condition monitoring and feedback mechanism during the current and the license renewal period.

In April 1996, the NRC issued a two-volume technical report, NUREG/CR-6384, "Literature Review of Environmental Qualification of Safety-Related Cables," which documented past and ongoing EQ research on safety-related cables. This report identified 7 major issues related to EQ. These 7 major issues were broken down into 43 sub-issues. Of the 43 sub-issues, 19 were considered unresolved. Public meetings to discuss the results of this review yielded additional data and industry reports on EQ, which were later reviewed in the research program. Based on draft NUREG/CR-6384, subsequent meetings (including a public workshop in

November 1993- see NUREG/CP-0135), reviews, and the availability of funding, six issues were identified that needed further review: (1) the Arrhenius methodology (i.e., the use of Arrhenius methodology to compare naturally aged cables with cables aged under accelerated conditions simulating the conditions to which the naturally aged cables were exposed), (2) activation energy estimates (i.e., whether activation energies used in past qualification tests were valid), (3) cable designs (i.e., multiple versus single conductor), (4) bonded versus unbonded jacket design, (5) in situ condition monitoring methods and (6) whether condition monitoring predicts accident survivability. Note that the issue concerning differences that exist in EQ requirements for older versus newer plants was addressed as a part of the literature review and site visits. The testing program concentrated on three most popular types of in-containment cables currently used in U.S. nuclear power plants.

Since the concerns identified in the EQ-TAP were either resolved by reviewing EQ-related literature or were being addressed by the cable research program, and since the cable research program was a long-term effort, the staff decided to close the EQ-TAP in 1998 and transfer the long-term research activities to Generic Safety Issue (GSI) 168, "Environmental Qualification of Low-Voltage Instrumentation and Control Cables." The staff would then inform the Commission of its progress on this issue through updates to NUREG-0933, "Prioritization of Generic Safety Issues."

2. Scope of GSI-168, "Environmental Qualification of Low-Voltage Instrumentation and Control Cables"

While preparing the EQ-TAP in 1993, the staff initiated GSI-168 to address any EQ-related generic safety issues identified during the EQ-TAP reviews. NUREG-0933 describes GSI-168 as follows:

> As discussed in SECY-93-049, the staff reviewed significant license renewal issues and found that several related to environmental qualification (EQ). A key aspect of these issues was whether the licensing bases, particularly for older plants whose licensing bases differ from newer plants, should be reassessed or enhanced in connection with license renewal or whether they should be reassessed for the current license term. The staff concluded that differences in EQ requirements constituted a potential generic issue, which should be evaluated for backfit independent of license renewal.
>
> In the staff's development of an interoffice action plan to address upgrading EQ requirements for older plants during the current licensing term, the staff evaluated the technical adequacy of EQ requirements. As part of this evaluation, the staff reviewed tests of qualified cables performed by SNL, under contract with the NRC. The purpose of these tests was to determine the effects of aging on cable products used in nuclear power plants. After accelerated aging, some of the environmentally-qualified cables either failed or exhibited marginal insulation resistance during accident testing, indicating that qualification of some electric cables may have been non-conservative. Although the SNL tests may have been more severe than required by NRC regulations, the test results raised questions with respect to the EQ and accident performance capability of certain artificially aged cables. Depending on the application, failure of these cables during or following design basis events could affect the performance of safety functions in nuclear power plants.

Although none of the EQ issues remaining after the EQ-TAP was closed in 1998 were identified as generic safety issues, GSI-168 was considered the most appropriate management tool for tracking and reporting progress in resolving them.

3. Research Program

In 1995, a research program was implemented with the following goals:

(1) To establish a database on operational performance, test performance, and environmental testing of cable in domestic and foreign operating plants, and to determine which EQ-related technical issues can be resolved with existing information and which need further research.

To develop and implement testing of low-voltage I&C cables to (a) assess the validity of current qualification methods, (b) evaluate the adequacy of accelerated aging methods, (c) evaluate methods for in situ cable inspection and condition monitoring, and (d) provide a technical basis for revising current rules and guidance on environmental qualification of cables, as needed.

As stated earlier, a thorough literature review and analysis was done to resolve as many issues as possible without resorting to extensive cable testing. The literature review and analysis, which is documented in Volumes 1 and 2 of NUREG/CR-6384, and Brookhaven National Laboratory (BNL) technical report TR-6169-9/97 sorted the 43 technical issues into four categories:

Category 1 issues were resolved by past work; new research was not needed: 24 issues.

Category 2 issues were unresolved by past work, but new research was not recommended: 6 issues.

Category 3 issues were unresolved by past work and new research was recommended: 6 issues.

Category 4 issues were unresolved by past work. No new research was recommended but the issues may be elucidated by the research on the Category 3 issues: 7 issues.

The Category 3 issues were:

(1) How do the properties of cables subjected to the accelerated aging techniques used in the original qualification compare with the properties of naturally aged cables of equivalent age?

Prolonged thermal exposure of cables inside nuclear power plants can cause significant degradation of their insulating polymers. These changes in the elastic properties of cable insulation and jacket materials can be considered a precursor to eventual cable failure. To account for this in the qualification process, cable samples are thermally preaged prior to LOCA testing to simulate their expected condition at the end of their service life. To simulate thermal aging for a 40-year service life, cable specimens are typically subjected to elevated temperature conditions in ovens. The aging oven temperatures are calculated based on the Arrhenius methodology using an estimated activation energy. Only limited work has been done to evaluate the application of Arrhenius, which was of interest in the research program. Comparisons of Arrhenius predictions to naturally aged materials have been limited. Therefore, the staff decided to study comparison of naturally aged cables with cables aged under accelerated conditions.

(2) *What are the limitations in using an estimated value of the activation energy to predict the chemical degradation during thermal aging?*

The estimated qualified life of a polymer is very sensitive to the value used for activation energy in the Arrhenius equation. For example, a change in activation energy from 1.0 to 1.3 eV increases the predicted qualified life from 12 to 87 years. The issue of activation energies is not resolved since there is uncertainty in the process used and the values obtained. It would be extremely cost and time intensive to embark on a comprehensive research program to accurately determine activation energies. It was, therefore, decided to use activation energies in our research program as used in the original qualification. However, the staff decided to include tests in our research program to verify activation energies for the cable materials being tested.

(3) *Do multiconductor cables have different failure mechanisms than single-conductor cables, and, if so, are these failure mechanisms accounted for in the qualification process?*

Control cables typically contain # 12-14 AWG conductors and may be single or multi-conductor cables. The number of conductors can influence the failure mechanism of the cable due to the greater self heating effects with increased conductors. In Sandia's previous work, EPR multiconductor cables were found to have a higher propensity to failure during a LOCA compared to identical single conductor cables. This was attributed to severe dimensional swelling, specifically under simultaneous radiation and steam exposure, and raised a question of whether this is a generic problem with multiconductor cables. Several hypotheses were provided to explain why the multiconductor cables failed. It is possible that penetrations used in the Sandia's LOCA test chamber could have caused differential pressure to develop between the inside and the outside of the cable during LOCA testing. Sufficient information was not available to draw a definitive conclusion. Therefore, the staff decided to include testing of single conductor & multiconductor cables in the research program.

(4) *Do cables with bonded jackets have different failure mechanisms than cables with unbonded jackets, and if so, are these failure mechanisms accounted for in the qualification process?*

Another variation in cable construction involves the jacket. The jacket protects the cable's insulation from mechanical damage, chemical attack and fire. In bonded jacket cables, the insulation and jackets are fused together and form a composite insulation. In this construction, the jacket and insulation can not be easily separated and do not move relative to each other, as in unbonded jacket cables. This could have an effect on failure mechanisms. Also, in Sandia's tests, the cables with bonded CSPE jackets cracked through to the conductor and subsequently failed during the post-LOCA voltage withstand test. The cause of bonded jacket cracking during accelerated aging was to surface cracks which developed in the Hypalon jacket and, due to its integral bonding with the insulation, propagated into the insulation material. Since the reasons for failures of cables with bonded jackets in the Sandia tests were not clearly understood, the staff decided to perform additional research. In our research program, Sandia tests were repeated using the same cable materials.

(5) *Are there any effective condition-monitoring techniques for determining cable condition in situ?*

As discussed earlier, there is a wide variation in the construction of cables; including the materials used, the number of conductors, the thickness of insulation and jackets, and the configuration. These variations, together with the environment in which the cables are installed, will affect the rate and degree to which cables will degrade. To ensure the cable will perform its safety function when needed, it is desirable to monitor the conditions of cables. There are many promising condition monitoring techniques. The staff decided to evaluate these techniques and their effectiveness.

(6) *Can condition-monitoring techniques be used to predict LOCA survivability?*

Since cables have a qualified life of 40 years and are not routinely replaced in a nuclear power plant, as noted earlier, it is desirable to determine the condition of the safety-related cables, with the ultimate goal of performing CM being to provide a means of predicting accident survivability for the cable.

Research Approach

The research approach consisted of:

1. Identification of the most popular types of in-containment cables currently used in U.S. nuclear power plants: Approximately 89% the currently operating plants had cables with XLPE insulation in containment, and approximately 73% had cables with EPR insulation.

2. Acquisition of cable samples: Naturally aged cables from two nuclear power plants were obtained. Unaged cables of the same type as the naturally aged cables were also obtained. Some cables were also obtained from Sandia which were tested in previous tests.

3. Baseline Information: Unaged cables with no-preaging were included in the tests as "control specimens." Hold-points were incorporated into the program to allow the condition and performance of the cables to be monitored at preselected intervals throughout the pre-aging and LOCA testing process. Various condition monitoring techniques were used at each hold point to obtain data on cables, as well as to evaluate of those CM techniques for monitoring cable condition.

4. Pre-aging & LOCA testing: Naturally aged cables were used in 3 tests. The accelerated aging parameters were chosen to match those used in the original qualification of cables to simulate 20, 40, and 60 years of qualified life.

Three types of I&C cables were selected: (1) cables insulated with ethylene propylene rubber (EPR) and covered with a bonded Hypalon® (chlorosulfonated polyethylene) individual jacket and a Hypalon® outer jacket; (2) cables insulated with EPR covered with a non-bonded

Hypalon® individual jacket and a Hypalon® outer jacket; and (3) cables insulated with cross-linked polyethylene (XLPE) with no individual jacket and a Neoprene® outer jacket.

In the testing, unaged cables underwent accelerated aging using currently accepted aging models to simulate 20, 40, and 60 years of qualified life. As noted earlier, naturally aged cables from two nuclear power plants were tested. For comparison, unaged cable of the same type as the naturally aged cables received accelerated aging, using currently accepted aging models, to match the service conditions to which the naturally aged cables were exposed.

During the testing, unaged cable specimens received accelerated thermal and radiation aging to the desired equivalent qualified life, then were exposed to high-temperature and high-pressure steam and chemical spray, simulating a design basis loss-of-coolant accident (LOCA). Unaged and naturally aged cables were also exposed to the same LOCA simulations and then their physical, chemical, and electrical properties were compared. The testing was done in accordance with IEEE Std. 323-1974, IEEE Std. 383-1974, and Regulatory Guide 1.89. The accelerated aging parameters were chosen to match those used in the original qualification of the cables. Note that the staff was unsuccessful in obtaining typical environmental conditions (temperature and radiation) in operating nuclear power plants to accurately simulate pre-aging.

The cables went through sequences of pre-aging and LOCA tests to address the EQ issues. In each sequence, one or more of the three cable types being studied were tested. When more than one type of cable was tested, the cables were pre-aged separately, using the original qualification parameters. The LOCA test profile was selected to envelop the profiles used in the original qualification of all the cables in the test. The pre-aging used thermal aging before radiation aging to be consistent with the original qualification protocol. In LOCA tests, radiation exposure preceded steam exposure.

The specimens were individually powered with 28 V dc. A pressure transmitter was connected to the test chamber leads of each long cable specimens to simulate instrumentation loop circuit. The short specimens in the stainless steel baskets were not powered since they were to be used for materials condition monitoring. Each of the powered specimens was monitored for applied voltage, circuit current, and leakage current throughout the LOCA steam exposure simulation. Circuit current was monitored for each conductor to facilitate troubleshooting. Each monitoring circuit was protected by a 1/32 A fuse.

The following documents on this research program were made publicly available for review and comment.

"Acquisition Plan for Non-aged and Naturally Aged Cable Samples From Nuclear Facilities," BNL Technical Report: TR-6168/69-01-9/95. This plan describes how to obtain representative new and naturally aged low-voltage electric cables from operating and decommissioned nuclear power plants, DOE facilities, and cable manufacturers for testing. The plan specifies the criteria for selecting samples and the background data to be obtained with the samples. It also gives special handling instructions to ensure that the selected cable samples are not damaged and do not experience abnormal environmental conditions during and after removal from a facility.

"Quality Assurance Plan,"BNL Technical Report TR-6169-05-95. This quality assurance (QA) plan ensures that the research results are traceable. The QA plan is based on the requirements in Appendix B of 10 CFR Part 50. All work, including the work of subcontractors, was done under this QA plan, which specified the development and approval of detailed test procedures for all testing activities and periodic audits.

"Pre-aging and LOCA Test Plan,"BNL Technical Report: TR-6168/69-04-95. This report describes the basis for the pre-aging and LOCA testing of low-voltage safety-related cables. The preliminary five-phase approach and test matrixes are described in this report. Throughout the test program, the test plan was modified to incorporate public input and recommendations and lessons learned from previously completed tests. Specimens of each type and manufacturer were prepared according to the established procedures. Some of the new or unused long specimens were wound on mandrels during testing; others were tested in straight lengths mounted in Unistrut® channels to simulate a typical installation in a cable tray. Each group of specimens was pre-aged by currently accepted accelerated thermal and radiation aging techniques to simulate the service age of the naturally aged cable samples and 20, 40, and 60 years of qualified life.

"Condition Monitoring Research Plan for Low-Voltage Electric Cables,"BNL Technical Report : TR 6168/69-03-95. This report describes the goals of the condition-monitoring research, the approach taken and the condition-monitoring techniques studied. The technique studies were visual examination, elongation-at-break, oxidation induction time, oxidation induction temperature, Fourier transform infrared spectroscopy, indenter, hardness, dielectric loss, insulation resistance, functional tests, and voltage withstand tests.

4. Results of Accelerated Aging and LOCA Testing

This section summarizes the results of the six test sequences. Details on the performance of these tests are provided in NUREG/CR-XXXX.

Test Sequence 1: Cross-Linked Polyethylene (XLPE) Insulated Cables Aged to 20 Years

The samples tested in this sequence were #14 and #16 AWG cross-linked polyethylene (XLPE)-insulated cables with a Neoprene® overall outer jacket manufactured by Rockbestos, with the trade name "Firewall® III." These cables had 30 mils of XLPE insulation on the individual conductors and a 45-mil overall Neoprene® jacket. The pre-aging parameters for the four groups of specimens in this test sequence were:

- Group 1: No pre-aging (control specimens)
- Group 2: Pre-aging to match naturally aged cable (2.86 hrs @ 248° F + 0.63 Mrad)
- Group 3: Naturally aged cable (10 years old)
- Group 4: Pre-aging to 20 years (648.5 hrs @ 302° F + 26.1 Mrad)

After completion of the accelerated aging, all the specimens in the first three groups appeared to be in good physical condition with good ductility, and no cracking was evident in any of the specimens. However, the Group 4 specimens were severely degraded with excessive cracking in the outer jackets, but no cracking in the conductor insulation. Insulation resistance measurements indicated that, electrically, the insulation on all specimens was in acceptable condition.

The LOCA conditions simulated included exposure to 150 Mrad of accident radiation, followed by exposure to steam at high temperature and pressure (346° F and 113 psig peak conditions; double peak profile), as well as chemical spray. The test duration was 7 days.

Specimens in Groups 1, 2, and 3 exhibited acceptable performance during the accident steam exposure. However, all five test specimens in Group 4 experienced performance anomalies. All specimens experienced leakage currents and fuses were blown off when the leakage current exceeded the fuse rating. In actual plant conditions, the leakage currents measured would be analyzed based on the intended safety application for the specimens to determine if leakage currents were acceptable.

Post-LOCA IR measurements indicated acceptable values for specimens in Groups 1, 2, and 3. However, the Group 4 measurements indicated short circuits in all the test specimens. Because of to severe degradation of the outer jackets on the Group 4 specimens, moisture intrusion occurred through the cracks in the jacket, and through micro-cracks in the insulation, after which it traveled along the conductor and into the splice. A leakage path was then set up from conductor to conductor, or from a conductor to the ground/shield wire in the test lead, which caused the high current readings observed. All cables passed the post-LOCA voltage withstand test (splices were removed from Group 4 specimens prior to this test).

In final analysis, it was concluded that all specimens passed the LOCA test.

Test Sequence 2: Ethylene Propylene Rubber (EPR) Insulated Cables Aged to 20 Years

The samples used in this sequence were #16 AWG, 3/C, 600V cables with 30 mils of EPR insulation and a 15 mil unbonded chlorosulfonated polyethylene (CSPE) (also known as Hypalon®) individual jacket covering the insulation on each conductor. The 45 mil overall outer jacket covering the conductor bundle was also made of Hypalon®. The cables were manufactured by American Insulated Wire (AIW). The pre-aging parameters for the four groups of specimens in this test sequence were:

Group 1: No pre-aging (control specimens)
Group 2: Pre-aging to match naturally aged cable (28.5 hrs @ 250° F + 3.3 Mrad)
Group 3: Naturally aged cable (24 years old)
Group 4: Pre-aging to 20 years (82.2 hrs @ 250° F + 25.7 Mrad).

After completion of accelerated aging, all the specimens appeared to be in relatively good condition with good ductility, and no cracking was evident in any of the specimens. Insulation resistance measurements indicated that all specimens were in acceptable condition electrically.

The LOCA conditions simulated included exposure to 150 Mrad of radiation followed by exposure to steam (340°F and 60 psig, peak conditions; single peak profile) and chemical spray. The test duration was 7 days.

Throughout the LOCA steam exposure, no performance abnormal observations were noted for any of the specimens. All cable specimens passed the post-LOCA voltage withstand test. Leakage currents observed were insignificant.

In final analysis, all specimens passed the LOCA test.

Test Sequence 3: XLPE Insulated Cables Aged to 40 Years

The test specimens were XLPE-insulated cables with a Neoprene® overall outer jacket manufactured by Rockbestos, with the trade name "Firewall® III." (These cables were the same as used in test sequence .1) The pre-aging parameters for the four aging groups in this test sequence were:

Group 1: No accelerated aging (control specimens)
Group 2: Accelerated aging to simulate the exposure of the naturally aged specimens (9.93 hrs @ 248° F + 2.27 Mrads)
Group 3: Naturally aged 10-year-old cable
Group 4: Accelerated aging to simulate 40 years of qualified life (1301.16 hrs @ 302° F + 51.49 Mrads)

After completion of the accelerated aging, all the specimens in the first three groups appeared to be in good physical condition with good ductility, and no cracking was evident in any of the specimens. The Group 4 specimens appeared to be severely degraded with extensive cracking in the jackets; however, no cracking was evident in the conductor insulation. IR measurements indicated that, electrically, the insulation in all specimens was in acceptable condition.

The LOCA conditions simulated included exposure to 150 Mrad of accident radiation followed by exposure to steam (using the same LOCA profile as used in test sequence 1) and chemical spray.

Throughout the test, performance problems were observed for all of the Group 4 specimens, including leakage currents and blown fuses. The specimens in Groups 1, 2, and 3 performed acceptably.

Following the LOCA steam exposure test, a post-LOCA inspection was performed in which in situ IR measurements were made on all specimens before and after opening the test chamber. The results showed generally lower IR values as compared to those taken prior to the LOCA steam exposure.

The Group 4 specimens were then removed from the test chamber and a post-LOCA inspection was performed. The nuclear grade Raychem splices were cut off on both ends of the specimens, after which acceptable IR measurements were obtained for four of the five cable specimens in Group 4. This indicated that the performance problems observed during the steam exposure were caused by problems in the splices, and not the cables. Disassembly, inspection, and testing of the Raychem splices revealed water inside the splices. Cracks were also noted in the insulation inside the splice. It was concluded that moisture intrusion into the splices, together with the insulation faults within the splices, had contributed to the Group 4 performance anomalies. The installation of these splices on cables with cracked and embrittled jackets may not be a typical application of splices. Similar problems were observed in test sequence 1. Although custom engineered splice kits were used for this test sequence, and technicians were given additional training in the installation of these splices, moisture was still observed inside the splices. The high- pressure steam in the LOCA chamber environment forced moisture into the cable through cracks in the jacket, where it was driven along the interior of the jacket, directly into the interior of the splice. Once there, cracks in the insulation allowed the moisture to provide a conductive path between the cable conductors. Since the insulation was fairly brittle after aging, cracking could have occurred during application of the splices or during condition monitoring of the cables when the conductors were handled, even though special precautions were taken to minimize the potential for handling damage.

One of the Group 4 specimens would not hold the full 500 volts even after its splices were removed. Further investigation showed that this specimen was damaged at one of the cable ties used to attach the test specimen to its test fixture. The cause of this failure was judged to be human error in handling the test specimen.

After post-LOCA inspections were completed, a voltage withstand test was conducted on each of the test specimens. In final analysis, all cables performed acceptably except the damaged specimen.

An important conclusion from the results of this test sequence is that cable condition should be considered prior to the application of splices to cables in the field.

Test Sequence 4: Multiconductor Cables

The objective of this test sequence was to determine whether multiconductor cables have any unique failure mechanisms that are not present in single conductor cable.

The test specimens were #16 AWG, 3/C, 1,000V EPR-insulated cables with CSPE individual and outer jackets manufactured by Anaconda. The conductors were insulated with 30 mils of EPR covered with a 15 mil unbonded CSPE individual jacket, and a 45 mil CSPE overall outer jacket. Also included in this test sequence were Samuel Moore cables #16 AWG, 2/C, 600V. The conductors were insulated with 20 mils of Dekoron (which is ethylene propylene diene monomer (EPDM)) with a bonded 10 mil Dekorad (CSPE) individual jacket, and a 45 mil Dekorad overall outer jacket. Each cable was tested in both the multiconductor and single conductor configuration. Single conductor specimens were made by disassembling a multiconductor length of cable. There were no naturally aged cable specimens in this test. The pre-aging groups in this test sequence were:

Group 1: Anaconda and Samuel Moore with no accelerated aging (control specimens)
Group 2: Samuel Moore with accelerated aging to simulate 20 years of qualified life (8.85 hrs @ 250° F + 25.99 Mrads)
Group 3: Anaconda (169.20 hrs @ 302° F + 53.60 Mrads) and Samuel Moore (169.05 hrs @ 250° F + 51.57 Mrads) with accelerated aging to simulate 40 years of qualified life.

After accelerated aging, all cables were in good condition with no cracking evident. The LOCA conditions simulated included exposure to 150 Mrads of accident radiation followed by steam (346° F and a pressure of 113 psig, peak conditions, as used in test sequences 1 and 3) and chemical spray. Note that in their original qualification test, Anaconda cables were qualified using a single-peak LOCA profile, while Samuel Moore cables were qualified using a double-peak LOCA profile.

Throughout the LOCA steam exposure, no abnormal observation was made for any of the test specimens.

The post-LOCA visual inspection of the Group 2 cables showed some degree of degradation to all the specimens. The multiconductor Samuel Moore specimens were still flexible and the outer jackets felt spongy to the touch. Large cracks were noted in the jackets, but, the underlying insulation appeared to be in good physical condition. The CSPE individual jacket on the single conductor Samuel Moore specimens appeared torn, shriveled and dis-bonded from the insulation near the cable ends. Swelling of the insulation and jacket materials was noted for all specimens.

Visual inspection of the Group 3 cables also showed degradation on each of the specimens. The multiconductor Samuel Moore specimens had multiple large cracks in the outer jacket, exposing the individual insulated conductors underneath. The exposed portion of the individual conductor jackets appeared to be in good physical condition. The single conductor specimens appeared similar to those in Group 2, with the CSPE jacket shriveled and dis-bonded on sections that were not attached to the mandrel. The Group 3 Anaconda multiconductor specimens had multiple large cracks and ruptures in the outer jacket. Swelling was noted for all specimens.

During the post-LOCA voltage withstand test, all the Anaconda cables and the Samuel Moore specimens aged to simulate 20 years performed acceptably. However, 2 out of 3 Samuel Moore specimens aged to simulate 40 years could not hold the 2400 V test voltage on one conductor. Subsequent dissection and inspection of the two specimens revealed a single pin-hole in each of the failed conductors. These were judged to be caused by localized degradation of the insulation, which was punctured by the high test voltage applied in the voltage withstand test. The area around the pin-hole was burnt, indicative of an electric discharge. It was, therefore, concluded that the failure was due to localized degradation of the insulation, which caused the high potential test to puncture the insulation on the two failed conductors.

It should be noted that the submerged voltage withstand test is an extremely harsh, potentially destructive test that is performed during qualification testing as a means of providing additional assurance that the cables will perform acceptably during and after exposure to accident conditions, and that the cables are capable of withstanding unexpected over voltages and electrical transients.

The data obtained related to differential swelling of jacket and insulation materials caused by moisture absorption, which could occur during a LOCA, provide evidence that this phenomenon can contribute to rupture or cracking of the materials during steam exposure. If the insulation on the conductors of a multiconductor cable expands faster and to a greater degree than the outer jacket, the resulting stresses imparted on the outer jacket may be significant enough to cause it to rupture. While these results might suggest a potential common cause failure for multiconductor cables, it must be noted that, of the 15 cables pre-aged to 40 years and LOCA tested in this program, 3 experienced performance problems that would impact their ability to perform safety function(s). Therefore, the significance of the differential swelling phenomenon depends on the materials of construction and the cable configuration.

Test Sequence 5: Bonded Jacket Cables

The objective of this test was to determine whether cables with individual jackets bonded to the underlying conductor insulation have any unique failure mechanisms that are not present in cables with an unbonded individual jacket.

The following cable specimens were tested:

- Anaconda 3/C, #16 AWG, 1,000V cable (with shield and ground wire): The conductors were insulated with 30 mils of EPR covered with a 15 mil unbonded CSPE individual jacket and a 45 mil CSPE overall outer jacket.
- Samuel Moore 2/C, #16 AWG, 600V cable (with shield and ground wire): The conductors were insulated with 20 mils of Dekoron (EPDM) covered with a bonded 10 mil Dekorad (CSPE) individual jacket and a 45 mil Dekorad (CSPE) overall outer jacket.

- Okonite 1/C #12 AWG, 600V cable: The conductor was insulated with 30 mils of Okonite (EPR) covered with a bonded 15 mil Okolon (CSPE) individual jacket.

There were no naturally aged cable specimens in this test sequence. The pre-aging groups included in this test sequence were:

Group 1: Specimens from Anaconda (A), Samuel Moore (S) and Okonite(O) with no pre-aging (control specimens)

Group 2: Specimens from A, S, and O with accelerated aging to simulate 20 years of qualified life (A: 84 hrs @ 302° F + 25.69 Mrads; S: 84 hrs @ 250° F + 25.99 Mrads; and O: 252 hrs @ 302° F + 25.79 Mrads)

Group 3: Specimens from A, S, and O with accelerated aging to simulate 40 years of qualified life (A: 169 hrs @ 302° F + 51.35 Mrads; S: 169 hrs @ 250° F + 51.57 Mrads; and O: 504 hrs @ 302° F + 51.49 Mrads)

After thermal and radiation aging, visual examination and elongation-at-break (EAB) measurements confirmed that all specimens in Group 2 were in excellent condition. In Group 3, the Samuel Moore specimens were in good condition and moderately flexible. The Anaconda specimens appeared degraded and were somewhat stiff. The Okonite specimens were brittle (EAB value was less than 5%).

The LOCA conditions simulated included exposure to 150 Mrads of accident radiation, followed by steam (double-peak LOCA profile, as used in test sequences 1 and 3, with a test duration of 10 days) and chemical spray. This profile was chosen to envelop the original qualification test profiles used for all three cables (Anaconda cables were originally qualified using a single-peak profile).

After the LOCA irradiation, no cracking was evident in any of the Samuel Moore or Anaconda specimens. The Okonite specimens in Group 3 had several circumferential cracks in the CSPE individual jacket. Throughout the first portion of the LOCA steam exposure, which included the two transients to peak conditions, no anomalies were noted for any of the test specimens. However, immediately upon the initiation of the chemical spray (at 15 hours into the test), all Group 3 Okonite specimens showed leakage currents in the range of 0.2 mA to 0.6 mA. Upon completion of the chemical spray, the leakage currents ceased. No other anomalies were noted for any of the other specimens.

Subsequent to completion of the steam exposure, a check of the test specimen wiring revealed that the monitoring circuit for the single-conductor Okonite specimens was incorrectly wired. This impaired the capability to monitor the leakage currents for these specimens, therefore, the exact time of failure could not be determined.

Post-LOCA examination of the specimens revealed that the Group 1 cables were in good condition. Group 2 cables showed some degree of degradation on all of the specimens. The Samuel Moore cables were flexible, but, small longitudinal cracks were noted in the jackets. The Anaconda specimens had multiple cracks in the outer jacket, which appeared to be due to swelling of the jacket. Also, one circumferential crack was noted in the outer jacket of one of the Anaconda specimens, exposing the individual insulated conductor jackets underneath. Each of the Okonite specimens was found to have a longitudinal crack along the length of the jacket. One of the specimens had split open, exposing the bare conductor underneath.

Visual inspection of Group 3 cables showed degradation on each of the specimens. The Samuel Moore specimens were still flexible, but, both specimens had a large crack in the outer jacket exposing the individual conductor jackets underneath. The exposed individual jackets appeared to be in good condition. The Anaconda specimens had multiple longitudinal cracks in the outer jackets. A large circumferential crack was noted in the overall jacket of one cable, exposing the individual insulated conductor jackets underneath. All of the Okonite specimens had longitudinal cracking of the composite jacket and underlying insulation that split open along the length of the cable, exposing the bare copper conductor.

After post-LOCA inspections, a voltage withstand test was conducted on each of the cable specimens. All the Samuel Moore and Anaconda cables performed acceptably. For the Okonite cables, 1 of the 2 specimens in Group 2 and all 3 specimens in Group 3 could not hold the 2400V test voltage. These cables were judged to have failed the test.

It was concluded that the failures observed in the Okonite specimens were caused by differential swelling of the bonded CSPE individual jacket and the underlying EPR insulation. Cracking was initiated in the CSPE individual jacket and propagated into the underlying EPR insulation because of the bonding. It should be noted that the Okonite pre-aging conditions used in the original qualification test were severe.

It is also noted that the manufacturer qualified the 1/C bonded jacket cables on the basis of a qualification test performed on larger cables using a "similarity" rationale.

Test Sequence 6: EPR and XLPE Insulated Cables Aged to 60 years

The following cables from four different manufacturers (Rockbestos, AIW, Samuel Moore, and Okonite) were tested in this sequence.

- Rockbestos cable insulated with 30 mils of XLPE insulation and a 45 mil Neoprene® outer jacket, with the trade name "Firewall® III." This cable was the same as the 2/C cable used in test sequences 1 and 3.
- AIW cable insulated with 30 mils of EPR covered with a 15 mil unbonded CSPE individual jacket, and a 45 mil CSPE overall outer jacket. This cable was the same as that used in test sequence 2.
- Samuel Moore cable insulated with 20 mils of Dekoron (EPDM) covered with a bonded 10 mil Dekorad (CSPE) individual jacket and a 45 mil Dekorad (CSPE) overall outer jacket. This cable was the same as that used in test sequences 4 and 5.

- Okonite cable insulated with 30 mils of Okonite (EPR) covered with a bonded 15 mil Okolon (CSPE) individual jacket. This cable was the same as that used in test sequence 5.

The pre-aging groups in this test sequence were:

Group 1: Rockbestos, Okonite, AIW, and Samuel Moore with no accelerated aging (control specimens).

Group 2: Rockbestos (1363 hours @ 302° F + 77 Mrads), Okonite (756 hours @ 302° F + 77 Mrads), AIW (252 hours @ 250° F + 77 Mrads), and Samuel Moore (252 hours @ 250° F + 77 Mrads) with accelerated aging to simulate 60 years of qualified life.

Subsequent to pre-aging, the visual inspections indicated that the Samuel Moore and AIW cables were in acceptable condition. The Neoprene®outer jackets on the Rockbestos specimens in Group 2 were brittle and appeared severely degraded with noticeable cracking and discoloration. The CSPE jacket on the Okonite specimens in Group 2 appeared to be in good physical condition; but, circumferential hairline cracks were found in the jackets of all three preaged specimens.

The LOCA conditions simulated included exposure to either 75 or 150 Mrads of accident radiation followed by steam (double-peak LOCA profile, as used in test sequences 1 and 3, with peak conditions of 346° F and a pressure of 113 psig and a duration of 10 days) and chemical spray.

During the steam exposure, performance problems were noted for all of the pre-aged Okonite specimens. The fuse protecting the power supply for each of the Okonite specimens blew, indicating a short circuit in the instrumentation loop circuit. Minor leakage currents were noted for the Rockbestos specimens. No problems were noted for the AIW or the Samuel Moore specimens.

Post-LOCA visual inspection revealed that all the bonded jackets and insulation on all three preaged Okonite specimens were split open in major sections of the cables, completely exposing the copper conductor inside.

Visual inspection of the AIW control specimen found that the outer jacket was cracked and pulled away from the rest of the cable caused by dimensional swelling of the jacket. Jacket degradation was also noted on the AIW specimens preaged to 60 years. The cracking noted was confined to the outer jacket; no evidence of insulation cracking could be found visually.

For Rockbestos specimens, numerous radial and longitudinal cracks were observed in the outer jacket over the entire length of the specimens. The cracking was confined to the outer jacket; no evidence of insulation cracking could be found visually. The Samuel Moore specimens appeared to be in good physical condition.

Following the post-LOCA investigation, the test specimens were subjected to a voltage withstand test. In general, all the specimens aged to 60 years exhibited a weakening of the insulation, which was manifested in the form of high leakage currents. Some specimens were unable to hold the 2400V test voltage.

These results indicate that degradation caused by aging beyond the qualified life of the cables, based on extrapolation of the aging used in the original qualification tests, may be too severe for the insulation material to withstand and still be able to perform adequately during a LOCA.

Sandia Tests:

In 1992, Sandia National Laboratories (SNL) conducted tests on cables manufactured by three different manufacturers, including Okonite. The tests were performed to determine the minimum insulation thickness necessary for installed cable to perform its intended function should the insulation be damaged during installation, maintenance, or other activities. Therefore, the thermal and radiation aging and loss-of-coolant-accident (LOCA) testing for the cables were performed with reduced and full insulation thicknesses. The Okonite specimens tested were single-conductor,600-volt, 12 American Wire Gauge (AWG) control cables insulated with ethylenepropylene rubber (EPR) with a bonded Hypalon jacket (Okonite-Okolon). During LOCA testing, all 10 of the Okonite-Okolon cable samples failed. The other cables in this test program did not have bonded jackets and did not experience failures.

During this test program, the cables were first subjected to 130 megarads of radiation at the rate of 300 kilorads per hour for 433 hours and were then thermally aged at 158 °C (316 °F) for 336 hours. Based on the Arrhenius equation, accelerated thermal aging at this time and temperature is equivalent to a 40-year cable life at 69 °C (156 °F) for the jacket and 76 °C (169 °F) for the insulation. After thermal aging, through-wall cracks were noted on most of the Okonite-Okolon cables. However, the cracks did not prevent the cables from passing an insulation resistance (IR) test that was conducted in a dry environment.

After the aging and IR tests, the cables were subjected to a LOCA test. The test sequence was (1) 94 hours of testing to simulate the LOCA environment defined in Appendix A IEEE Std. 323-1974, and (2) 146 hours at 121 °C (250 °F) for the remainder of the test. No chemical spray was used. The cables were energized by 110-volt DC power during the test with no load. One cable with full insulation thickness failed just after the test chamber conditions became saturated at 11-1/2 hours into the test. By the fifth day of the test, all the Okonite cables had failed, as indicated by blown 1-ampere fuses. The test chamber was opened on October 24, 1992, and the cables were visually inspected. The insulation and jacket on the Okonite cables had split down the length of the cable, and bare conductor was visible.

In another Sandia test, the other Okonite cables with bonded Hypalon jackets had failed similarly. For this test, the cables were thermally aged to the equivalent of a 40-year life at 56 °C (133 °F). One out of four Okonite-Okolon cables failed during LOCA testing. Another group of Okonite cables that had been aged to a 40-year life at 50 °C (122 °F) passed this testing.

Also, cables manufactured by Samuel Moore also failed during LOCA testing. These cables were Dekoron Dekorad Type 1952, two-conductor, twisted, shielded pair, 16 AWG instrument cables covered with ethylene-propylene diene monomer (a type of EPR) insulation with a bonded Hypalon jacket and an overall jacket of Hypalon. One cable in which one conductor failed had been thermally aged to a 20-year life at 55 °C (131 °F), while the other cable in which both conductors failed had been thermally aged to a 40-year life at 56 °C (133 °F). These failures were similar to the failure of the Okonite-Okolon cable in that the insulation and bonded jacket had split open. Other samples of Samuel Moore cable survived aging and accident testing under similar conditions.

These SNL test results raised questions in 1992 with respect to the environmental qualification of Okonite cables with bonded Hypalon jackets that have not been specifically qualified for service conditions exceeding 50 °C (122 °F) for 40 years. The staff reviewed the qualification data developed by the Okonite Company and noted that Okonite 2 kV cables with 0.76 mm [30 mil] bonded Hypalon jackets and 600-volt cables with unjacketed EPR insulation were previously tested. The 600-volt cables with 0.38 mm [15 mil] bonded Hypalon jackets were qualified based on the previous 2 kV and 600- volt test results. It was believed that if the unjacketed EPR insulation passed qualification testing, then EPR insulation with a bonded jacket would also pass qualification testing. However, the Sandia test results indicated that Okonite cable with bonded Hypalon jackets may be susceptible to failure.

The qualification data reviewed by the staff for the Samuel Moore cables showed that cables with bonded Hypalon jackets had been previously tested by Isomedix, Inc. The tests documented qualification of the Dekoron Dekorad cable to a qualified life of 40 years at plant service conditions of 52 °C (126 °F) or less. The test results SNL raised questions about the qualification of Samuel Moore Dekoron Dekorad Type 1952 cables when used at higher temperatures.

Other bonded-jacket cables, qualified for up to 90 °C (194 °F) applications as claimed by various vendors, may be susceptible to the same type of failures if not specifically tested in the bonded configuration. The difference in aging rates between the jacket and the insulation may be a factor in the failure of bonded-jacket cables. Therefore, qualification testing that does not use the jacketed configuration may not be representative of actual cable performance.

Depending on the application, failure of these cables could affect the performance of safety functions in nuclear power plants. The functional integrity of the cables could be affected if the cables are used inside containment, used in continuous power circuits, routed with power cables, or routed close to hot pipes.

In another test, Sandia National Laboratories also tested cables to determine the long-term aging degradation behavior of typical instrumentation and control cables used in nuclear power plants and to determine the potential for using condition monitoring for assessing residual life. The results of this testing are described in NUREG/CR-5772, "Aging, Condition Monitoring, and Loss-of-Coolant Accident (LOCA) Tests of Class 1E Electrical Cables," Volumes 1, 2, and 3. The tests were conducted on cross-linked polyolefin/poly-ethylene, ethylene propylene rubber, and miscellaneous Class 1E cable types. The test program generally followed the guidance of IEEE Std. 323-1974 and IEEE Std. 383-1974.

The test program consisted of two phases; both phases used the same test specimens. Phase 1 consisted of simultaneous thermal and radiation aging of the cables at approximately 100 C (212 F) and 0.10 kGy per hour (10 kilorads per hour), respectively. Three different sets of cable specimens were tested in this phase: one was aged to a nominal lifetime of 20 years, a second to 40 years, and a third to 60 years. Phase 2 was a sequential accident exposure consisting of 1100 kGy (110 megarads) of high-dose-rate irradiation at the rate of 6 kGy per hour (600 kilorads per hour) followed by a simulated exposure to LOCA steam. The test profile was similar to the one given in IEEE Standard 323-1974 for "generic" qualification. The cables were energized at 110 V dc during the accident simulation. Insulation resistance was measured on line throughout the test. No chemical spray was used during the steam exposure, but a post-LOCA submergence test was performed on the cables that were aged to a nominal equivalent of 40 years.

Cable types that failed during the accident tests or that exhibited marginal insulation resistances were Rockbestos Firewall III, BIW Bostrad 7E, Okonite-Okolon, Samuel Moore Dekoron Dekorad Type 1952, Kerite 1977, Rockbestos RSS-6-104/LE Coaxial, and Champlain Kapton.

The Sandia National Laboratories test raised questions with respect to the environmental qualification (EQ) of certain cables that either failed or exhibited marginal insulation resistance values. The staff reviewed the test data and noted that cable types identified as Firewall III, Okonite, Dekorad, and Kapton failed during the simulated accident exposure, while BIW Bostrad, Rockbestos Coaxial, and Kerite exhibited marginal insulation resistances. It should be noted that the insulation resistance of the Rockbestos coaxial cables may be too low to meet specifications for use in General Atomics radiation monitor circuits, depending on the environment to which the cable will be exposed.

As part of the NRC-sponsored aging research program, the Sandia National Laboratories searched licensee event reports (LERs) to find LERs that might be related to cable aging. In NUREG/CR-5461, "Aging of Cables, Connections, and Electrical Penetrations Assemblies Used in Nuclear Power Plants," the Sandia National Laboratories concluded that although cables are highly reliable devices under normal plant operating conditions, with no evidence of significant increases in failure rate with aging, the performance experience with these components under actual accident conditions is small. The current LER data provide a very limited database for this purpose. The only significant data for cables subjected to design-basis events comes from EQ testing.

Depending on the application, failure of these cables during or following design-basis events could affect the performance of safety functions in nuclear power plants.

Conclusions on EQ Issues:

Based on the results of the testing, the following conclusions can be drawn:

Accelerated Aging Techniques:

> The data obtained suggest that the accelerated aging predictions using the Arrhenius model for thermal aging, and the equal-dose/equal-damage model for radiation aging provide adequate estimates of the degradation experienced during actual service aging. In six out of six cases, material that received accelerated aging had a lower EAB, indicating more degradation than naturally aged material of equivalent age.

Activation Energies:

> The data from these tests demonstrate that, for the two cable insulation materials tested, the activation energies used in the original qualification tests were representative of the materials being tested. While this does not confirm that accurate activation energies were used for all cables, it does provide evidence that the activation energies used in past qualification tests were reasonable.

Multiconductor Cables:

> The data obtained related to differential swelling of jacket and insulation materials caused by moisture absorption, which could occur during a LOCA, provide evidence that this phenomenon can contribute to rupture or cracking of the materials during steam exposure. If the insulation on the conductors of a multiconductor cable expands faster and to a greater degree than the outer jacket, the resulting stresses imparted on the outer jacket may be significant enough to cause it to rupture. While these results might suggest a potential common cause failure for multiconductor cables, it must be noted that, of the 15 cables pre-aged to 40 years and LOCA tested in this program, only 3 experienced performance problems that would impact their safety function. Therefore, the significance of the differential swelling phenomenon depends on the materials of construction and the cable configuration.

Bonded Jacket Cables:

> The results of this test demonstrate that the bonded jacket/insulation configuration has a potential for catastrophic failure under LOCA conditions. This catastrophic failure can occur if the composite bonded jacket/insulation is first exposed to severe aging conditions, causing it to embrittle and shrink significantly, prior to its sudden exposure to steam. The steam causes swelling stresses that can initiate failure.

Extending Qualified Life:

> The results indicate that degradation caused by aging beyond the qualified life of the cables, based on extrapolation of the aging used in the original qualification tests, may be too severe for the insulation material to withstand and still be able to perform adequately during a LOCA. For life extension purposes, the aging protocols used to establish the qualified life of the cables should be reviewed and compared to actual service environments in a plant.

Condition Monitoring Tests

As previously mentioned, two of the issues addressed by the research program relate to in situ condition monitoring (CM) of electric cables. Specifically, the issues being addressed were (1) identify CM techniques that can be used to effectively monitor the condition of cables in situ and (2) determine if CM data can be used to predict accident survivability. The approach taken to address these issues was to first identify the criteria for an "ideal" CM technique. Eleven criteria were identified. With the anticipation that no single technique would meet all of the criteria, promising CM techniques were reviewed and several were selected for further study. The following CM techniques were selected.

Visual Inspection:

In comparison to the other CM methods, which produce quantitative results, visual inspections provide a qualitative assessment of cable condition. Cable attributes that are inspected visually include (1) color, including changes from the original color and variations along the length of cable, and the degree of sheen, (2) cracks, including crack length, direction, depth, location, and number per unit area, and (3) visible surface contamination, including any foreign material on the surface. Also, the rigidity of the cable is qualitatively determined by squeezing and gently flexing it.

Elongation-at-Break:

Elongation-at-break (EAB) is a measure of a material's resistance to fracture under an applied tensile stress. It is defined as the percent increase in elongation at the time of fracture and is a well known and accepted method of measuring a polymer's condition. However, it is a destructive test that requires relatively large pieces of material. In this research program, the EAB test was used as the reference against which other CM techniques were compared.

Oxidation Induction Time:

The time at which rapid oxidation of a material occurs when held at a constant, elevated test temperature in a flowing oxygen environment is termed the oxidation induction time (OITM). As a material ages, anti-oxidants added to the material during production are gradually lost, leaving the material susceptible to oxidation. By measuring the OITM and comparing it to values when the material was new, an estimate of the condition of the material can be made.

Oxidation Induction Temperature:

As with the OITM measurements, the oxidation induction temperature (OITP) is a measure of the amount of anti-oxidants remaining in a material. The test specimen is prepared in a way identical to those for OITM. However, the OITP is the temperature of the material at which rapid oxidation occurs as the test temperature is increased at a constant rate in oxygen.

Fourier Transform Infrared Spectroscopy:

Fourier Transform Infrared (FTIR) Spectroscopy is a technique for analyzing the structure of molecules. The principle involves the measurement of absorbance or transmittance of infrared radiation by molecular structures, including those for polymers. As the radiation passes through a polymer, atoms absorb radiation and begin to vibrate. For a particular chemical bond, maximum vibration occurs for a specific wavelength of radiation. Therefore, by irradiating a specimen with a continuous spectrum of infrared radiation, and measuring the peaks (wavelengths) at which maximum absorbance or transmittance occurs, the chemical bonds that are vibrating may be identified from standard wavelengths that are available from the open literature. By identifying certain bonds that are known to form as the material degrades, the condition of the material can be characterized.

Compressive Modulus (Indenter):

Compressive modulus is a material property defined as the ratio of compressive stress to compressive strain below the proportional limit. As cable insulation and jacket materials age they tend to harden, which will cause the compressive modulus of the materials to increase. By monitoring this change in compressive modulus, an estimate of the degradation rate of the material can be made. To monitor changes in the compressive modulus, the Ogden Indenter Polymer Aging Monitor (Indenter) was used. This device presses a probe into the material being tested and measures the force required for the resulting displacement. These values are then used to calculate the compressive modulus of the material. The probe is controlled by a portable computer and appropriate software, which controls the travel of the probe to prevent damage to the cable.

Hardness:

As a comparison to the indenter, simple hardness measurements were performed on the jacket and insulation specimens and evaluated as a potential condition monitoring technique. In theory, as the cable materials harden with age, a hardness test may be useful for correlating age degradation with changes in hardness readings. Hardness measurements are similar to the indenter measurement in that a probe is pressed against the cable and the cable surface deforms. The differences between the simple hardness measurement using a Shore Durometer and the indenter are the level of sophistication of the test and the sensitivity of the instrument in taking measurements.

Dielectric Loss:

The phase angle between an applied test voltage and the total current in a circuit is known as the dielectric phase angle, and this can be measured with a signal analyzer. As the insulation on an electric cable deteriorates, it is expected that the leakage current will increase, while the capacitive current remains approximately constant. This would cause the dielectric phase angle to decrease. By monitoring the change in phase angle, the amount of deterioration can be estimated.

Insulation Resistance:

Insulation resistance measurements are commonly performed to determine the current condition of cable insulation. By applying a voltage from the conductor to ground and measuring the resulting current flow, the resistance of the insulation separating them can be measured. As insulation degrades, it is expected that the insulation resistance will decrease.

Functional Performance Test:

The main concern in monitoring cable condition is to determine whether degradation from service aging will affect the accident performance of the cable. Therefore, the performance of the cable during operation was evaluated as a CM technique to determine whether it provides information on the cable condition. To perform this evaluation, the test specimens were powered and loaded to simulate an actual circuit during accident testing. Changes in circuit current and leakage current were then monitored as the cable was exposed to simulated accident conditions.

Voltage Withstand:

As part of the current qualification procedures, cables being qualified are subjected to a submerged voltage withstand test after accident testing. Each specimen is individually submerged in tap water at room temperature while being subjected to a test voltage of 80 Vac/mil of insulation thickness for a period of 5 minutes. As the test voltage is applied, the leakage current between the conductor and electrical ground is recorded. The resulting leakage current can be used as an indicator of insulation condition.

Each of the CM techniques was performed on the test specimens at preselected intervals during the pre-aging, accident testing and post-LOCA testing, as appropriate. Note that the system functional tests and voltage withstand tests were associated only with the LOCA test sequence. The data were then evaluated to draw conclusions regarding the effectiveness of the technique.

Condition Monitoring Test Results:

One of the primary purpose of the research program for low-voltage I&C cables was to evaluate various condition monitoring methods for their effectiveness to detect degradation attributable to aging of polymers, and obtain data useful for the assessment of accident survivability of installed cable systems.

In this regard, there are two ways to categorize the condition monitoring of electric cables:

(i) Assessment of the bulk properties of insulating materials, and

(ii) Evaluation of the current condition of an entire installed cable system.

Research results indicate that meaningful information can be derived from testing samples of polymeric materials in controlled laboratory conditions. These methods include measurement of physical properties, chemical properties and electrical properties. Some of these methods can be applied with some limitations for in situ assessment of installed cable systems. The biggest limitation being the "accessibility." For example, it would not be possible to detect degradation of polymers at "hot-spots" and at localized anomalies where parts of cable systems are not readily accessible. Evaluation of localized samples of polymeric insulating materials, in situ or under controlled laboratory conditions, provides only a part of the information; it does not provide data to evaluate the condition of an entire cable system, from end to end.

From a perspective of a system-level evaluation, some electrical measurement techniques are available to detect gross defects and failures and locate failure points. However, they are ineffective for detecting incipient defects prior to failures. Some of the known electrical measurement techniques and methods include: (a) insulation resistance, (b) power factor and loss factor measurement, (c) time domain reflectometry, (d) megger test, and (e) voltage withstand. Some of the emerging electrical technologies that have potential for a system-level evaluation of cable, include: (a) dielectric spectroscopy, (b) ionized gas method, and (c) partial discharge measurement.

Based on the results of the testing, the following conclusions were drawn regarding the effectiveness of the techniques studied for monitoring cable condition:

Visual Inspection:

> While it does not provide quantitative data, it does provide useful information on the condition of the cable; that is easy and inexpensive to obtain, and that can be used to determine if further investigation of the cable condition is warranted. A significant limitation of this technique is that the cable must be visually accessible. It is suggested that visual inspection be considered for inclusion in any cable condition monitoring program.

Elongation-at-Break (EAB):

> Elongation-at-break was found to be a reliable technique for determining the condition of the polymers studied. It provides trendable data that can be directly correlated with material condition. It is useful as a reference technique, however, its destructive nature prevents it from being used as an in situ means of monitoring electric cables unless sacrificial cable specimens are available.

Oxidation Induction Time (OITM):

> OITM was found to be a promising technique for monitoring the condition of electric cables. Results show that aging degradation can be trended with this technique for both XLPE and EPR insulation. While a small sample of cable material is needed to perform this test, the relatively small amount required should be obtainable without impacting cable performance. OITM can be used as an in situ technique.

Oxidation Induction Temperature (OITP):

> While it is related to OITM, OITP was found to be less sensitive to the detection of aging degradation for the polymers studied. It can be used as an in situ technique, however, OITM is preferred at this time.

Fourier Transform Infrared Spectroscopy (FTIR):

> FTIR was found to provide inconclusive results in terms of its ability to trend aging degradation in the polymers studied. Although the results show a consistent trend with age, the technical basis for the trend remains questionable.

Indenter:

> The indenter was found to be a reliable device that provides reproducible, trendable data for monitoring the degradation of cables in situ. While it is limited to accessible sections of cables, it was found to be effective for monitoring the condition of common

cable jacket and insulation materials. Therefore, the indenter can be used as an in situ technique for monitoring, localized and accessible segments of, low-voltage electric cables.

Hardness:

This technique was evaluated since it is a simple, inexpensive technique to perform. The results indicate that, over a limited range, the hardness can be used to trend cable degradation. However, different probes must be used to accommodate the change in material hardness. Also, puncturing of the cable insulating material is a potential concern with this technique.

Dielectric Loss (DL):

This technique was found to provide useful data for trending the degradation of cable insulation. As the cables degrade, a definite change in phase angle between and applied test voltage and the circuit current can be detected at various test frequencies that can be correlated to cable condition. This technique can be used as an in situ condition monitoring technique. It is more effective when ground plane is an integral element of a cable system.

Insulation Resistance:

This technique was found to provide useful data for trending the degradation of cable insulation. As the cables degrade, a definite change in insulation resistance can be detected that can be correlated to cable condition. Using 1 minute and 10 minute readings to calculate polarization index enables the effects of temperature and humidity variations to be accounted for. This technique can be used as an In situ condition monitoring technique.

Functional Performance:

The use of functional performance data as a means of monitoring the condition of electric cables was evaluated since it is a simple, inexpensive technique to perform. While useful information can be obtained to determine if further condition monitoring is needed, this technique alone does not provide sufficient data to determine the condition of a cable. It is a "go" "no go" type of test and may not be effective in detecting degraded conditions and impending failures. Further, functional performance testing is not considered an effective method for determining, in situ, the LOCA survivability for a particular cable.

Voltage Withstand:

For a cable system to perform its intended function it must withstand the voltage in order to carry the necessary current and deliver power. Therefore, the capability of the insulating materials to withstand the circuit voltage is an indication of its dielectric performance. In order to detect defects in incipient status, applied voltages may be elevated considerably above the rated voltages of the systems; further, the equipment at

GUIDANCE FOR ACRS REVIEW OF LICENSE RENEWAL GENERIC DOCUMENT

1. Do the SRP, GALL II report, and associated regulatory guide provide adequate technical bases to support license renewal decisions?

2. Are the SRP, the GALL II report and the NEI implementation documents effectively integrated? Do they provide a consistent and understandable process? Does the SRP provide a user friendly map of how these documents come together?

3. Is guidance adequate to support effective scoping/screening of older plants? Are the lessons learned from the review of the OCONEE and Calvert Cliff Nuclear Plant license renewal applications adequately conveyed to future reviewers?

4. Does the SRP direct the staff to develop a comprehensive understanding of the technical issues and of the proposed technical solutions or direct the staff to verify the existence of aging management programs?

5. Is review of plant specific operating experience adequately emphasized by the SRP? Is guidance adequate to evaluate the effectiveness of plant programs dealing with unique types of plant specific aging degradation?

6. Have the SRP and supporting documents taken into proper consideration the issues and concerns raised by all stakeholders?

7. Are the license renewal generic issue resolutions adequately reflected in the guidance documents?

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
476th	15	15-2

Title

FUTURE ACRS ACTIVITIES

Authors

John T. Larkins

List of Documents Attached

Future ACRS Activities - 476th ACRS Meeting, October 5-7, 2000

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

From Staff Person

John T. Larkins

ANTICIPATED WORKLOAD
October 5-7, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	–	Markley Markley Weston	**Discussion of Union of Concerned Scientists Report,"Nuclear Plant Risk Studies: A Failing Grade".** NEI 00-02, "Industry PRA Peer Review Process Guidelines". Staff Views on ASME Standard for PRA .	Report Report –		P&P 10/4 (P.M.) FP 10/16-17 RF 10/18 PLR 10/19-20 RES 11/1
Bonaca	Seale	Dudley	ACRS Review of Generic Guidance Documents Associated with License Renewal [Discussion of Members' Comments]	–	PLR 10/19-20	P&P 10/4 (P.M.) DPO 10/10-14 RES 11/1
Powers	All Members	Larkins Savio El-Zeftawy	Meeting with the Commission Discussion of Industry Issues Research Report to the Commission	–	P&P 10/4 (P.M.) DPO 10/10-14 RES 11/1	FP 10/16-17 RF 10/18 PS 10/31
Shack	Wallis	Dudley	PTS Technical Basis Revaluation Project	Report		PLR 10/19-20 RES 11/1

ANTICIPATED WORKLOAD
October [illegible]-7, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Uhrig	—	Singh	Proposed Resolution of GSI-168: Qualification of Electrical Equipment [Status Report]	—	PS 10/31	PLR 10/19-20 RES 11/1

ANTICIPATED WORKLOAD
November 2-4, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	Leitch	Markley	Risk-Informed Regulation Implementation Plan	Report		P&P 10/31 (P.M.)
Bonaca	Seale	Dudley	License Renewal Documents: SRP, GALL , and Regulatory Guide	Report		P&P 10/31 (P.M.) M&M 11/16
Kress	--	El-Zeftawy/ Weston	Spent Fuel Pool Accident Risk at Decommissioning Plants	Report	SAM 11/15	TH 11/13-14 M&M 11/16
Powers	All Members	El-Zeftawy/ Duraiswamy/Shoop	Research Report to the Commission Differing Professional Opinion on Steam Generator Tube Integrity	Report Report (Tentative)	P&P 10/31 (P.M.)	

ANTICIPATED WORKLOAD
November 2-4, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Sieber	Powers	Singh	Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues	Report		SAM 11/15
Uhrig	–	Singh	ABB/CE and Siemens Digital I&C Applications (Subcommittee Report)	–	PS 10/31	

ANTICIPATED WORKLOAD
December 7-9, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG. CHAIR.	SUBC. MTG. MEMBER
Apostolakis		Markley	Proposed Revisions to 10 CFR 50.46,"Acceptance Criteria for Emergency Core Cooling Systems for Light -Water Power Reactors".	Report	—	P&P 12/6 (P.M)
		Markley	ANS Standard on External Events PRA [shift to Jan/Feb Mtg.]	Report		
Bonaca	Wallis	Boehnert	Central Issues Related to Core Power Uprate Reviews	Report	—	P&P 12/6 (P.M)
Kress	—	Boehnert	Control Room Habitability	Report	—	
Powers	—	Duraiswamy/Shoop	DPO on Steam Generator Tube Integrity	Report	P&P 12/6 (P.M)	
	—	El-Zeftawy	Research Report to the Commission [possible finalization of letter @ retreat?] (draft)	Report		
Seale		Singh	Management Directive 6.4 to address ACRS Concerns Associated with the Generic Safety Issue Process [shift to Jan/Feb mtg?]	Report	—	

ANTICIPATED WORKLOAD
December 7-9, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Shack	--	Dudley	Proposed Final Regulatory Guide DG-1053,"Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"	Report		
Sieber	Apostolakis	Weston	Draft Safety Evaluation for the South Texas Project Exemption from scope of special requirements.	Report		
Wallis		Boehnert	EPRI Report ,"Resolution of Generic Letter 96-06 Waterhammer Issues". [shift to Jan/Feb mtg?]	Report		

II. ITEMS REQUIRING COMMITTEE ACTION

1. Proposed Revision to 10 CFR Part 73, "Physical Protection of Plants and Materials" (Open) (TSK/NFD) ESTIMATED TIME: 1 ½ hours

   Purpose: Determine a Course of Action

   Review requested by the NRC staff [Michael Jamgochian, NRR]. The staff briefed the Committee on its reevaluation of power reactor physical protection regulations and its position on a definition of radiological sabotage at the May 2000 ACRS meeting. The staff is preparing a proposed revision to 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage." The staff provided the ACRS with a copy of the proposed revision on September 1, 2000.

   Dr. Kress recommended course of action: issueance of Larkinsgram proposing review after the resolution of public comments (currently 3/02).

2. Central Issues Related to Core Power Uprate Reviews (Open) (MVB/GBW/PAB/AWC) ESTIMATED TIME: 1½ hours

   Purpose: Determine a Course of Action

   ACRS Initiative. Recently, several BWR licensees have announced their intention to submit license amendment requests for plant power uprates, most comprising significant power increases (10-17% above nominal). ACRS Fellow G. Cronenberg had previously identified some significant concerns pertaining to the adequacy of the NRC staff's uprate review procedures (June 23, 2000 memorandum to the ACRS), most notably among them the lack of a Standard Review Plan Section for uprate reviews.

   During the May 2000 meeting, the Committee agreed to pursue the issue of the need for development of formal staff guidance for uprate reviews (e.g. Standard Review Plan Section) in tandem with its next power uprate review. Dr. Powers, via a May 23, 2000 E-Mail, has also raised a number of issues he believes need to be explored by the Committee during its review of this matter.

   More recently, Dr. Bonaca has drafted a proposed White Paper titled, "Potential Synergistic Effects of Industry Initiatives to Extend Plant Life, Increase Production and Reduce Regulatory Burden". In this Paper, Dr. Bonaca argues that staff approval on an individual basis of such licensing actions as power uprates, plant life extension and installation of extended burnup fuel are not accounting for the synergistic effects impacting overall plant safety margins and plant risk.

   Dr. Powers has recommended that the Planning and Procedures Subcommittee discuss this matter with the objective of the Committee engaging the staff in discussion of its concerns. NRR staff management has been informed that the Committee may request a discussion session on this issue during the December ACRS meeting.

   The Planning and Procedures Subcommittee recommends full Committee decide future course of action on this issue after staff December presentation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 29, 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 (November 2000 - February 2001) and the ACNW (November 2000 - January 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 September 26, 2000 with the OEDO, NRR, NMSS, and RES ACRS/ACNW coordinators [OEDO, G.C. Millman; NRR, M.G. Crutchley; NMSS, R.H. Turtil; RES, J.A. Mitchell and S.R. Nesmith].

A copy of the Work Items Tracking System (WITS) list for November 2000 - January 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

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DRAFT PROPOSED AGENDA FOR ACRS MEETINGS (November 2000 - February 2001)

ACRS MEETING ---- November 2 - 4, 2000				
Item #	Title/Issue	Purpose	Priority	Documents
1	Risk-Informed Regulation Implementation Plan Contact: T. King, DRAA/RES	Review and Comment	High	Draft Plan to be provided 10/15/00.
2	ABB-CE and Siemens Digital I&C Applications Contact: E. Marinos, DE/NRR	Information Briefing	High	Siemens SE has been provided to ACRS; ABB-CE SE was provided to the ACRS in late August.
3	Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues Contact: E. Weiss, DSSA/NRR	Review and Comment	Medium	RG, draft revised NFPA 805, and other relevant documents have been provided to Committee.
4	Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants Contact: T. Collins, DSSA/NRR	Review and Comment	High	Results of technical study to be provided by 10/12/00.
5	License Renewal Contact: S. Hoffman, DRIP/NRR	Review and Comment	High	Improved Renewal Guidance documents issued for public comment on 8/31/00 was discussed with ACRS on 8/30. Copies provided on 9/1/00.

ACRS MEETING --- December 7 - 9, 2000				
Item #	Title/Issue	Purpose	Priority	Documents
1	DG-1053; Dosimetry and NeutronTransport Contact: W. Jones, DET/RES	Review and Comment	High	Draft regulatory guide to be provided by 10/13/00.
2	Waterhammer Issues Contact: J. Tatum, DSSA/NRR	Review and Comment	High	EPRI interim report to be provided by 11/1/00.
3	Control Room Habitability Contact: J. Hayes, DSSA/NRR	Review and Comment	High	Draft SER to be provided by 11/3/00. Comments on revision to NEI 99-03 to be provided by 10/13/00.
4	South Texas Exemption from Scope of Special Requirements Contact: J. Nakoski, DLPM/NRR	Review and Comment	High	Draft SER to be provided by 11/3/00.
5	Part 50, Option 3 and 50.46 Contact: Mary Drouin, DRAA/RES	Review and Comment	High	Draft Commission paper to be provided by 11/15/00.
6	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 11/9/00.
7	Central Issues Related to Core Power Update Reviews Contact: T. Kim, DLPM/NRR	Information Briefing	Medium	None.

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ACRS MEETING ---- FEBRUARY 2001				
Item #	Title/Issue	Purpose	Priority	Documents
1	NEI 97-06 Steam Generator Program Guidelines Contact: E. Sullivan, DE/NRR	Review and Comment	Medium	Draft SER to be provided by mid-December.
2	Effectiveness of the ATWS Rule Contact: W. Raughley, DSARE/RES	Information Briefing	Medium	Draft ATWS report to be provided in early October.
3	GSI-152, Reprioritization of 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-RELAPS Appendix K Small-Break LOCA Code Contact: R. Caruso/R.Landry, DSSA/NRR	Review and Comment	High	SER on Code to be provided mid-December.

G Drive:PP

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No:
476th	15	15.2

Title MINUTES OF PLANNING & PROCEDURES SUBCOMMITTEE MEETING - OCTOBER 4, 2000

Authors JOHN T. LARKINS

List of Documents Attached

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From Staff Person

JOHN T. LARKINS

G:PlanPro:ppmins.476
October 6, 2000

MINUTES OF THE
PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
WEDNESDAY, OCTOBER 4, 2000

The ACRS Subcommittee on Planning and Procedures held a meeting October 4, 2000, in Room 2 B1, 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 3:45 p.m.

ATTENDEES

D. A. Powers, Chairman
M. Bonaca

ACRS STAFF

J. T. Larkins
J. E. Lyons
H. Larson
R. P. Savio
S. Duraiswamy
C. Harris
S. Meador

NRC STAFF

I. Schoenfeld, OEDO

DISCUSSION

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the October ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the October ACRS meeting are included in a separate handout. Reports and letters that would benefit from additional consideration at a future ACRS meeting was discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the October 2000 ACRS meeting be as shown in the handout.

2) Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through December 2000 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 members should provide comments on the anticipated workload. Changes will be made, as appropriate:
- The Committee should consider the Subcommittee's recommendations on items listed in Section II of the Future Activities.
- Assign lead responsibility to Mr. Leitch for reviewing proposed revision to 10 CFR Part 73, if the Committee decides to review this matter.
- In view of the anticipated heavy workload for the December ACRS meeting, the following items should be deferred to the February 2001 ACRS meeting and the Committee should seek the views of the cognizant Subcommittee Chairman:
    - Management Directive 6.4 to address ACRS concerns associated with the revised GSI process (Dr. Seale).
    - EPRI Report on Resolution of GL 96-06 related to water hammer issues (Dr. Wallis).
    - Proposed final Regulatory Guide DG-1053, Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Dr. Shack).

3) <u>Issues Associated with Core Power Uprates</u>

During its August 29th meeting, the Planning and Procedures Subcommittee discussed the central issues pertaining to core power uprates, especially those identified by the ACRS Senior Fellow, Dr. Cronenberg [Note: There has been outside interest regarding the report prepared by Dr. Cronenberg]. In addition, the Subcommittee discussed a proposed White paper prepared by Dr. Bonaca on "Potential Synergistic Effects of Industry Initiatives to Extend Plant Life, Increase Production, and Reduce Regulatory Burden" (pp. 1-4). The full Committee agreed with the Subcommittee's recommendation that the issues raised by Dr. Bonaca in his White paper be included in the ensuing ACRS report to the Commission on the NRC Safety Research Program. Also, the Subcommittee decided to continue its discussion of the issues associated with the core power uprates and related matters during its October 4, 2000 meeting. A report prepared by Mr. Boehnert is attached (pp. 5-11). Mr. Boehnert's recent discussions with the staff indicate that, if requested by the Committee, the staff is willing to brief the Committee on this matter during the December 2000 ACRS meeting. Proposed topics include:

- NRR position regarding the need for applying risk-informed decisionmaking to "significant" power uprate applications. Criteria for determining "special circumstances."
- NRR plans for revising guidance documents and developing a Standard Review Plan Section for power uprate reviews.
- Synergisms among changes in nuclear plants and their potential impact on plant safety:
  - Higher burnup fuel
  - Power uprates
  - Use of "best-estimate" or "more-realistic" analyses
  - Plant life extension
  - RES activities associated with core power uprates.

<u>RECOMMENDATION</u>

The Subcommittee recommends that the Committee hear presentations by and hold discussions with the staff on the topics proposed above during the December 7-9, 2000 meeting. Subsequent to this briefing, the Committee should decide whether further review of this matter should be performed by the Plant Operations Subcommittee or an Ad Hoc Subcommittee.

4) Mixed Oxide Fuel Fabrication Facility

The Department of Energy (DOE) plans to construct a Mixed Oxide (MOX) Fuel Fabrication Facility (FFF) on its Savannah River site to manufacture fuel from weapons program plutonium. A consortium composed of Duke Power, Cogema Fuels, and Stone & Webster (DCS) will construct and operate the facility for DOE. The NRC staff expects the application for construction authorization to be submitted in December 2000 and the operating license application in June 2002. NRC staff review of the safety design basis for the MOX FFF will occur at the construction authorization stage. The NRC staff also expects DCS to submit the license amendment for loading MOX fuel test assemblies in McGuire Unit 2 in August 2001. A license amendment for batch loading of MOX fuel in the McGuire and Catawba reactors is expected in December 2003. The Commission is interested in the ACRS review of and comment on the MOX FFF. Review of the safety issues associated with the use of MOX fuel in operating reactors is within the purview of the ACRS.

RECOMMENDATION

The Subcommittee recommends that an information briefing be scheduled for the February 2001 ACRS meeting and that the ACRS review the DCS application to construct the MOX FFF after the staff has completed its Safety Evaluation Report (SER) and provide a report to the Commission. Subsequent to the construction of this facility, the ACRS should review the operating license application along with the NRC staff's SER and provide another report to the Commission. At present the lead responsibility for reviewing this matter should be assigned to the Fire Protection Subcommittee. Subsequently, the Committee should decide whether one of the existing Subcommittees or an Ad Hoc Subcommittee should review this matter.

5) Meeting with the Nuclear Energy Institute

As part of the CY 1998 and CY 1999 ACRS self assessments, feedback was solicited from various stakeholders, including some industry groups. Some stakeholders stated that the ACRS was not sufficiently informed of the industry's concerns and needs. As a result, a meeting with representatives of the Nuclear Energy Institute (NEI) has been scheduled for the October 5-7, 2000 ACRS meeting to discuss items of mutual interest. Topics agreed to by NEI and proposed lead member assignments are provided below:

1) Risk informing 10 CFR Part 50 (Apostolakis)
2) License renewal (Bonaca)
3) Decommissioning (Kress)
4) Revised reactor oversight process (Sieber)

6) Estimation of Resources for FY 2001

Due to the anticipated high workload facing the ACRS in FY 2001, it is important to plan how to use member time most efficiently and effectively. Assuming the number of ACRS members remains constant throughout FY 2001, the maximum member time that will be available is 1,300 days.

During last month's Planning and Procedures Subcommittee meeting, we discussed the need to manage better the number of subcommittee meetings and the number of members participating in subcommittee meetings. Senior staff engineers with input from Subcommittee chairmen were asked to revise the estimate of the number of subcommittee meetings for FY 2001. The current estimate shows 36 subcommittee meetings, 10 full Committee meetings and 1 retreat, consuming a total of approximately 1155 days. The Planning and Procedures subcommittee will need to scrutinize these proposed subcommittee meetings to assess where some cuts might be made or combining of subcommittee meetings might be done. This needs to be done to make sure we do not exceed the maximum days available for members to work and also not to overburden members.

RECOMMENDATION

The Subcommittee plans to develop guidelines for use by the members in scheduling and participating in Subcommittee meetings to ensure that we do not exceed the allocated budget for FY 2001. The ACRS staff should provide a list of scheduled and proposed Subcommittee meetings for discussion by the Planning and Procedures Subcommittee during each of its meetings. The Subcommittee will use this information to prioritize the Subcommittee meetings, as warranted, to manage the budget and the members' workload.

7) Progress Made in Addressing the Commitments Resulting from the CY 1999 ACRS Self Assessment/ACRS Self Assessment for CY 2000

Results of the ACRS/ACNW self assessment for CY 1999 were documented in SECY-00-0102 and sent to the Commission on May 5, 2000. In that SECY paper, the ACRS had made several commitments (pp. 12-14). The Subcommittee discussed the actions taken to address these commitments and provided feedback on the adequacy of these actions. The Subcommittee also provided guidance for conducting the CY 2000 self assessment.

RECOMMENDATION

The Subcommittee recommends that the Committee provide feedback on the adequacy of the actions taken to address the commitments resulting from the CY 1999 self assessment. Also, the process (informal and formal feedback from stakeholders) used for the CY 1999 ACRS self assessment should be used for conducting the CY 2000 self assessment. Dr. Savio should develop a list of about eight issues for discussion during the 2001 retreat and to assess the effectiveness of the Committee in CY 2000.

8) ACRS Retreat for 2001

During the September meeting, the Committees agreed to have a retreat in January 2001 together with its visit to the San Onofre Nuclear Plant. Since the San Onofre visit did not materialize, a decision on the dates and location for the retreat should be made.

RECOMMENDATION

Because of budget constraints, the Subcommittee recommends that the Committee approve holding the retreat locally and select the dates. Also, it should propose topics for the retreat.

9) Proposed ACRS Meeting Dates for CY 2001

The proposed dates for CY 2001 ACRS meetings are included in the attached Calendar (pp. 15-26).

RECOMMENDATION

The Subcommittee recommends that the Committee approve these dates during the October or November ACRS meeting.

10) Member Issues

Dr. Powers plans to attend the 28th Water Reactor Safety meeting scheduled for October 23-25, 2000. He has been invited to serve on the Expert Panel on Challenges in the Future for Risk-Informed Regulation.

Potential synergistic effects of industry initiatives to extend plant life, increase production and reduce regulatory burden.
Mario Bonaca, ACRS, 9/12/2000, Rev 3.

The License Renewal (LR) Rule rests on the basic regulatory principle that a nuclear power plant (NPP) can continue to operate for as long as it complies with its current licensing basis (CLB), because compliance with its CLB provides assurance of adequate protection of public health and safety.

The LR implementation documents provide details on how an NPP demonstrates that aging degradation will be adequately managed so that plant structures, systems and components (SSCs) will continue to comply with their CLB requirements for as long as the plant continues to operate. Active components are excluded from LR consideration because existing regulation, such as the Maintenance Rule, already imposes requirements on timing and level of corrective action for safety significant active components.

Passive components fall into two different categories. One category includes passive components subject to periodic replacement under their CLB. These components are identified by the licensee and the staff for the purpose of reviewing existing CLB commitments dealing with age degradation and to assess their adequacy for the extended period of operation.

The other category includes long-lived passive components that are not subject to periodic replacement under their current CLB. This category includes major reactor coolant system components such as reactor coolant system piping, reactor vessel and internals, pressurizer and steam generators in pressurized water reactors (PWR), reactor coolant pump casings, emergency systems piping, secondary side major components such as steam lines, and containment. For these components aging degradation is monitored to assure that it will not exceed aging degradation limits required to support the CLB. In those cases where component operation is supported by a time limited aging analysis that does not extend beyond 40 years, the time limited aging analysis must be modified to qualify the component for the extended period of operation.

In most instances long-lived passive components are expected to operate for the extended period of operation without being replaced. This is possible because these components are designed with excess margin over the regulatory limits that support the CLB. Part of this excess margin is in fact intended to, and used for operating the plants to their currently licensed 40 years life. Extending the life of the plants beyond 40 years involves the recognition that excess margin is still available in most components after 40 years of operation and the acceptance of its use to compensate for aging degradation for the purpose of extending the life of the facility. Since regulatory limits are not exceeded, the plant continues to comply with its CLB, and this provides assurance of adequate protection.

Although regulatory limits may not be exceeded, SSCs actual margins to aging degradation limits are being reduced. At the end of 60 years life, mechanical components will be closer to their fatigue limits than at the end of 40 years, the reactor vessel will be more brittle and closer to the pressurized thermal shock (PTS) limit than at 40 years of life, and so on, and even replacement steam generators, which should be capable of reaching the end of 60 years plant life, will exhibit aging degradation from 20 additional years of service.

If a complete PRA of the plant that would describe aging effects appropriately were performed at 40 years of life, and then again at 60 years, one would expect to observe an increase in risk measures such as CDF and LERF, due to an expected higher failure probability of long lived components subjected to 20 more years of service. Higher failure probabilities would tend to affect PRA results in several ways:

- By increasing initiator frequency of accidents caused by rupture of passive components,
- By increasing the possibility of cascading failures from physical interaction of ruptured components with adjacent age-degraded components,
- By increasing the probability of failure of engineered safety systems, and
- By reducing the structural capability of the RCS and containment barriers during severe accidents.

This increase in risk measures may not be insignificant and may exceed the guidelines of Regulatory Guide 1.174 at least for plants characterized by relatively high CDF and LERF.

As stated above, the regulatory logic behind the decision to implement the LR rule without further risk consideration seems to be based on the basic concept that a plant complying with the current deterministic regulation meets the requirements for adequate protection even if its risk to the public increases with age. This concept is accepted for the first 40 years of life. The LR rule extends its acceptance beyond the first 40 years. Since the LR rule does not establish a life extension limit, there is an implication that the LR rule will allow as maximum acceptable risk from aging the risk associated with a condition where all long-lived components have aged to their regulatory limit without exceeding it. This approach would not be in conflict with the guidelines of RG 1.174 if current PRAs of operating plants already assumed aging of all components to their regulatory limits and met the subsidiary safety goals. But current PRAs have not explicitly and systematically addressed aging effects, and many plants do not meet the subsidiary safety goals of CDF and LERF. Therefore, granting a renewed license without consideration of aging risk may conflict with the guidelines of RG 1.174.

The License Renewal rule requires the implementation of aging management programs that are comprehensive and provide reasonable assurance that the increase in risk due to aging during the period of extended operation should be small. This perspective supports the acceptance of license renewal without consideration of associated risk, as an extension of the licensing philosophy supporting the first 40 years of operation.

Concerns remain about the risk implications of concurrent licensing actions proposed by licensees that may compete for the same SSC margins used to support life extension and that are likely to be evaluated without explicit consideration of aging. The exclusion of aging risk considerations from the LR rule does not mean that the aging effects experienced during the period of extended operation don't need to be considered in risk assessments of other licensing actions that may be affected by the aging of components.

Examples of such licensing actions include power up-rates and the use of high burn-up fuel currently being planned by several plants. In his June 23 report to the ACRS on this subject (Ref. 1), Dr. Cronenberg noted that "several recent operational events for uprated plants point to circumstantial evidence of compounding degradation due to aging/uprate and high-burnup/high-power effects, which have not been addressed in prior uprate reviews." The report provides several examples of pipe failures that have occurred in uprated plants. The report also states "Agency inaction for a more comprehensive uprate review process is being justified by risk arguments of minor changes in CDF for power uprates."

These power uprate requests will come in for review and approval through licensing actions under the provisions of existing deterministic rules. A study performed by Energy Research, Inc. (ERI) for the Swiss Nuclear Inspectorate in 1997 (Ref.2) assessed the risk associated with a 14.7% power increase of the Leibstadt NPP in Switzerland. Leibstadt is a BWR6 with a Mark-III containment. The study showed that the power upgrade would result in minor increases in CDF and LERF, but in a 30% increase in risk as measured by the risk metric of frequency of a release times the activity of the release. This increase is directly attributed to the increased radioactive inventory and the time acceleration of events due to the increased decay heat level at the uprated power conditions. The results of this study also showed that the metrics of RG 1.174 are not the most appropriate ones to be used to assess the risk increase due to uprate in reactor power. Even if the NRC were to perform a risk assessment of such uprates using the insight of the ERI study, approval or denial of the licensing request is likely to be based on the merits of the uprate request alone, without consideration of the additional risks associated with other licensing actions such as license renewal and of the potential synergistic effects resulting from the combined licensing actions.

Since many NPPs are planning to extend their life, and many are planning power upgrades, we may face a situation where a plant characterized by high risk (maybe not apparent because its PRA is incomplete or inadequate) could be allowed to raise its power level, and as a completely separate action go for life extension. Another concurrent separate action could include justifying continued operation for some time with degraded steam generators. The current licensing process does not allow for risk considerations to effectively enter into the decision of whether these plant actions can be supported simultaneously or even individually. PRA is the only tool having the potential capability of comprehensively exploring the synergies of such proposed plant changes. But its benefits are effectively excluded by

- Current lack of complete information on the actual risk associated with operating plants,
- Explicit exclusion of PRA considerations from LR rule,
- Weak understanding of impact of aging on plant risk (no systematic PRA study has been performed, methodology has only partially been developed)
- Lack of complete PRA models to seriously evaluate the synergistic effects of industry

initiatives to increase production, extend life and reduce regulatory burden.

- The staff needs to address the global issue being raised by the Industry's move to aggressively utilize existing plant margin above minimum regulatory limits. Piecemeal review and approval of industry requests may fail to identify important synergies that may result from the separate licensing actions. We need to understand what the NRC in general, and RES in particular are doing about this issue. Depending on the staff's initiatives in this area we may need to recommend a focused effort in our research report. Also, the metrics of RG 1.174 may need to be augmented if CDF and LERF are not sufficient to identify plant risk associated with licensing actions, as the ERI report seems to suggest.

Ref 1 - A.W.Cronenberg to ACRS members and staff, "Central Issues Related to Power Uprate Reviews", June 23, 2000.
Ref 2 - Schmocker, Khatib-Rahbar, Cazzoli, Kuritzky, "An Assessment of the Risk-Impact of Reactor Power Upgrade for a BWR-6 MARK-III Plant", PSAM-3 Meeting, Crete, Greece (1997).

PREPARED FOR INTERNAL COMMITTEE USE

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

September 25, 2000

MEMORANDUM TO: D. Powers, Chairman, Planning and Procedures Subcommittee

FROM: P. Boehnert, Senior Staff Engineer

SUBJECT: ACRS REVIEW OF ISSUES PERTAINING TO CORE POWER UPRATES

Background

During last month's meeting of the Planning and Procedures Subcommittee, there was discussion of issues pertaining to core power uprates. Specifically, Dr G. Cronenberg, ACRS Senior Fellow, and Dr. M. Bonaca, have raised a number of concerns pertaining to plans by licensees (typically of BWR plants) to apply for significant core power uprates (15 - 20% of nominal). Dr. Bonaca outlined his concerns during the above-noted meeting and said that he was preparing a White Paper on this matter (see below). The Subcommittee did not reach any definitive conclusions as to how the Committee should disposition this matter.

Issue Statement

Subsequent to last month's ACRS Meeting, I had an exchange of E-Mails with you on this matter. You noted that you intended to raise the issue of core power uprates in the context of synergisms being seen with such issues as: use of higher burnup fuel, power uprates, use of "best-estimate" codes, and plant life extension. You indicated that these synergisms can be seen as competing for plant safety margins and they will have an impact on plant risk. For my part, I suggested that the Committee consider providing formal comment to the Commission in the near future on this issue, as the lead applicant is scheduled to submit a power increase request of ~15% by November 1, 2000. Therefore, the lead-plant licensee's schedule should not be impacted in addressing the ACRS's concerns by virtue of being "first in line". You subsequently instructed that the P&P Subcommittee again discuss this issue during its October 4, 2000 Meeting.

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DRAFT PREDECISIONAL INFORMATION ATTACHED**

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New Information

NRR Guidance for Review of Power Uprates

I inquired of R. Barrett, NRR, as to the staff's policy regarding evaluation of the risk impact associated with power uprates. I reminded him that as a result of Members' inquiries during the Committee's reviews of the power uprate applications for the Monticello and Hatch plants in 1998, he informed the Committee that for all future uprate requests of a "substantial magnitude" some sort of PRA analysis would be performed by the licensee to obtain a measure of the increase in plant risk. Mr. Barrett informed me that the staff now has formulated guidance of an interim nature, subsequently approved by the Commission, as to how to deal with license amendment requests that are not risk informed. Key points noted for this guidance include:

- No specific requirements exist for a licensee to perform risk analyses in support of license amendment requests, nor is it required that licensees perform and maintain a PRA for their plants. Therefore, the staff has the burden to consider the risk impact of license amendment requests that are not risk informed (i.e., based on the deterministic regulations).

- The staff promulgated SECY-99-246 (copy attached) and Regulatory Issue Summary 2000-07 to provide information to licensees regarding interim guidance on use of risk information by the staff in its license amendment reviews that are not risk informed. The need for this guidance grew out of the Callaway steam generator Electrosleeve amendment request.

- NRR has developed an interim process for evaluating the risk impact of non-risk-informed license submittals (see Figure 1 attached). The heart of the process is to determine if "special circumstances" exist that would necessitate submittal of additional information (particularly from a risk perspective). Three Elements comprise this process, to wit:

  o Guidance for screening amendment requests to identify "special circumstances"

  o Methodology for assessing the risk implications of potentially risk-significant amendment requests

  o Guidelines for determining the acceptability of the license action which factor in risk considerations

  The staff indicated in SECY-99-246 that portions of these Elements are still under development.

- Currently, the staff screens amendment requests for risk implications but on an ad hoc basis. Plans are proposed to modify the license review process documents to provide additional guidance to aid in identifying amendment requests containing "special circumstances".

The ACRS reviewed a draft version of SECY-99-246, and provided formal comment via an October 8, 1999 report to the then-NRC Chairman (copy attached). In its report, the Committee agreed on the need for additional guidance and basically endorsed the staff's approach. The Committee did note, however, that the crucial element in the process "will be the selection of the criteria that define 'special circumstances'". The Committee also said the staff needs to be mindful of not creating a process that inhibits use of risk-based information by licensees, and that the staff needs to improve its own risk and accident analysis tools to better judge proposed risk-informed license amendments.

Dr. Bonaca's White Paper

A copy of the most-recent version of Dr. Bonaca's draft White Paper (September 12, 2000) is attached. This Paper addresses the issue of the synergistic effects of industry initiatives related to plant life extension, power uprates, and reduction of regulatory burden. This version reflects changes resulting from comments received from Dr. Powers. **Please note that this paper is labeled "Draft Predicisional".**

NRR/GE Nuclear Energy Meeting

I attended a meeting held between representatives of NRR and GE Nuclear Energy to discuss two programs proposed by GENE to facilitate core power uprates. One of these programs dealt with streamlining the so-called Extended Power Uprate Program, which will encompass power uprates of 15-20%, as noted above. A copy of my meeting summary is attached (**contains proprietary information**). GENE made a point of discussing the anticipated workload associated with the expected uprate submittals (in addition to the three submittals to be made this year, another three uprate submittals are expected in 2001 and 4-6/year starting in 2002 and beyond). GENE expressed concern with the potential for schedule delay if ACRS review of each uprate application is required. At the end of my writeup, I proposed some suggestions as to how the Committee could handle the anticipated workload.

Recommendations

Based on my preliminary discussions with NRR, the staff is expecting that the Committee will propose a discussion with representatives of the staff on this matter during the December Meeting. My recommendation is that the Committee schedule a two-hour discussion with representatives of NRR/RES management during that Meeting. Some proposed discussion items could include:

o NRR position with regard to the need for applying risk-informed decision making to "significant" power uprate applications. Criteria to be applied for determination of "special circumstances".

o NRR plans relative to revision of staff guidance documents/development of a Standard Review Plan Section addressing power uprate reviews.

o The competition for plant safety margins resulting from synergisms among proposed plant changes:

— New fuel designs/higher burnup fuel
— Power uprates
— Use of "best-estimate or "more realistic" analyses
— Plant life extension

o RES activities associated with this matter.

o ACRS workload associated with power uprate reviews

— Need for individual reviews/potential scheduler impact
— Screening of potential reviews (via risk analyses)
— Selective review procedure
— Other review approaches?

Attachments: As Stated

cc: Balance of ACRS Members
R. Savio

cc w/o attach (via E-mail):
J. Larkins
J. Lyons
H. Larson
ACRS Technical Staff & Fellows

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DRAFT PREDECISIONAL INFORMATION ATTACHED

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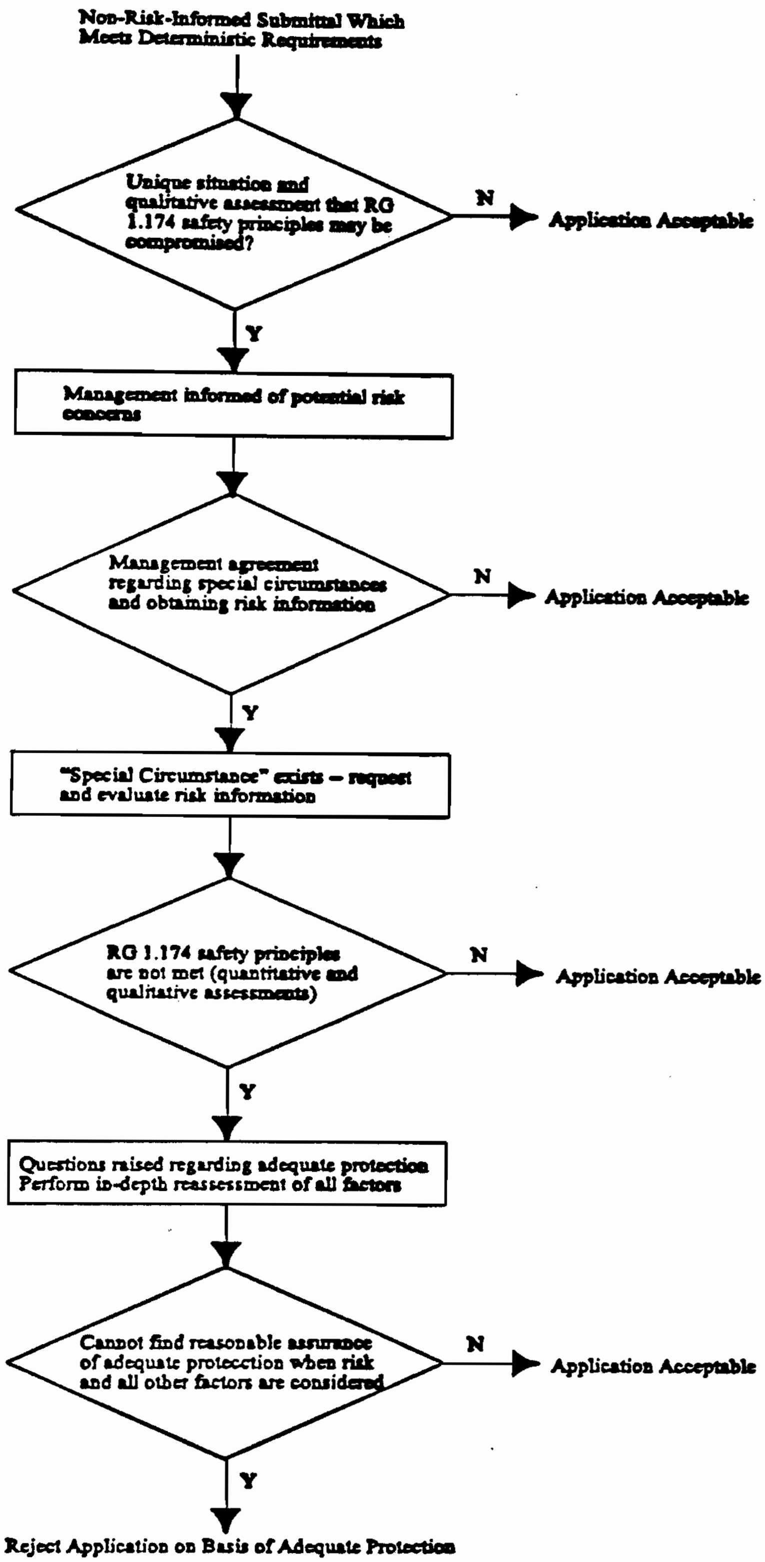
Figure 1 - Process and Logic for Considering Risk in License Application Reviews



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

September 14, 2000

MEMORANDUM TO: G. Wallis, Chairman, Thermal-Hydraulic Phenomena Subcommittee

M. Bonaca, Chairman, License Renewal Subcommittee

D. Powers, Chairman, ACRS

FROM: P. Boehnert, Senior Staff Engineer PB

SUBJECT: NRR/GE NUCLEAR ENERGY MEETING: PROPOSED LICENSING ACTIONS PERTAINING TO CORE POWER UPRATES, SEPTEMBER 11, 2000

I attended the subject meeting held between representatives of NRR and GE Nuclear Energy (GENE). The meeting was closed to the public to discuss GENE proprietary information. The purpose of this meeting was to discuss GENE's proposed licensing actions pertaining to core power uprate applications. Specifically, GENE discussed two proposed programs:

- Thermal Power Optimization (TPO) Program
- Extended Power Uprate Simplification: Constant Reactor Dome Pressure

Key points noted in the discussions included:

TPO Program - GENE has developed a so-called Thermal Power Optimization Program to allow GE plant licensees to take advantage of the recent rule change to Appendix K of the ECCS Rule. This rule change allows a small core power increase by use of more accurate flow instrumentation/methodology to address the 2% core power uncertainty requirement. GENE has submitted a licensing topical report that contains a methodology for addressing the required review items. The topical report will bound core power increases of no more than 1.5%. The TPO methodology will accommodate the choice of power optimization employed by the individual licensee (detailed analysis or improved flow instrumentation). The report will contain generic analyses and will also provide a guide to the issues/topics that

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require either confirmation of the validity of generic evaluations or stand-alone analyses, on a plant-specific basis. In response to questions from NRR, I indicated that the ACRS would likely not want to be involved in review of this matter, since the Committee reviewed and recommended issuance of the revised Appendix K Rule that is applicable to this item.

EPU Simplification - Constant Reactor Dome Pressure

GENE plans to submit a Supplement to its Extended Power Uprate (EPU) topical report that contains a streamlined methodology to guide submittal of the BWR extended power uprate (EPU) license applications. The method is based on a "no pressure increase" strategy which GE believes will narrow the NRC review focus and broaden the number of items/issues that can be addressed generically. In addition to including generic assessments, a plant-specific "shell" will be provided to guide the prospective licensee's submittal. GENE intends to submit this Supplement to NRC by the end of this year. This approach will not be applied to the EPU applications expected in the near term (Duane Arnold, Quad Cities & Dresden); however, both of these applicants are using the "no pressure increase" approach for their uprates.

GENE also discussed the anticipated workload. At this time, three additional EPU applications are expected during 2001 (Clinton, Brunswick, and a "plant-to-be-named-later"). GENE expects 4-6 additional submittals in 2002, and a similar submittal rate for 2003 and beyond. All present at the meeting agreed that this submittal rate posed a significant workload challenge to GENE and the NRC. There was some discussion of the ACRS role as well. I stated the position that the ACRS can be expected to review each EPU application. GENE expressed concern with the potential schedule delay, given Committee review.

[Note: I believe that the Committee should give thought as to how it will approach the workload associated with the EPU reviews. Some review "Options" may include: (1) review of each individual submittal, (2) a selective review procedure, perhaps dependent on the similarity of a given plant to one previously reviewed, (3) decide on need for detailed review of a given application based on a PRA evaluation of the risk impact of the power uprate, or (4) some combination of (2) & (3) above.]

cc: Balance of ACRS Members
R. Savio

cc w/o attach (via E-mail):
J. Larkins
J. Lyons
H. Larson
ACRS Technical Staff & Fellows

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Commitments from CY99 ACRS Self Assessment

1) Modify ACRS/ACNW Operating Plan in accordance with new NRC planning initiatives, draft FY2000-2005 Strategic Plan, and FY2001 Performance Plan, with incorporation of self assessment information and metrics. (Larkins/ Savio/ Gallo-- Issue to Commission by 11/30/2000)

ACTIONS: Format for a new Operating Plan is being developed as planned. Sam Duraiswamy and Mag Weston are working on a ACRS Action Plan which, as will the existing ACNW Action Plan, be incorporated into the content of the new Operating Plan.

2) Develop action plan that will identify and allocate resources for ACRS and ACNW review of selected decommissioning issues and use the Joint ACRS/ACNW Subcommittee to help coordinate work on decommissioning issues. (Outlined in split of ACRS and ACNW responsibility decommissioning paper and will be developed in ACRS/ACNW Operating Plan. Decommissioning paper will be updated as needed to account for new initiatives and schedule changes. Activities will be incorporated in Future Activities scheduling using existing processes.)) (Larkins/Savio/Larson)

ACTIONS: An action plan has been developed that identifies the decommissioning issues, schedules, Committee and Joint ACRS/ACNW Subcommittee assignments, and a general approach to the reviews. This information has been provided to the Commission as per a request from their staff. Activities are being incorporated into ACRS (and ACNW) Future Activities scheduling using existing process. Priorities (ie, resource allocation were there is the expected competition with other activities) will be broadly addressed in the ACRS Action Plan and specifically in the Planning and Procedures Subcommittee process.

3) Maintain awareness of need to preserve independence, re. early involvement in the development of NRC staff positions. (P&P Subcommittee oversight)

ACTIONS: The Planning and Procedures Subcommittee has been doing this in its monthly meetings. No issues have been identified that could not be resolved by routine Subcommittee deliberation. Stakeholder feedback will be solicited prior to the next ACRS retreat to see if there are issues that have not been addressed. This may well end up being a issue where there will continue to be different stakeholder views as to how the ACRS should conduct its business.

4) Return to a mode of operation that will afford more in-depth review of issues when warranted. (P&P Subcommittee oversight)

ACTIONS: Tabulations of ACRS activities are being developed which will be used to evaluate this issue and will be distributed the October 2000 P&P Subcommittee.

5) Look for more opportunities to increase involvement in important technical issues and minimize involvement in routine matters such as rules and regulatory guides addressing routine technical or process issues. The examples of important technical issues given in the self assessment SECY were:

a) risk-informed initiatives for improving regulation (10 CFR Part 50, pressurized thermal shock, and decommissioning)
b) future NRC research needs
c) risk-based performance indicators
d PRA quality standards
e) human performance
f) digital I&C
g) transient and accident analyses code certification
h) emerging uses of mixed-oxide and high-burnup fuels

To conserve resources ACRS would end its review efforts when technical issues have been satisfactorily resolved and staff is addressing implementation. (P&P Subcommittee prioritization and scheduling of ACRS activities)

ACTIONS: Same as Item 5–to be provided. All of the examples of important technical issues have been addressed in CY2000 ACRS activities.

6) Systematically assess how ACRS, as a Commission-level advisory committee, can add value to an issue prior to agreeing to reviewing the issue. (P&P Subcommittee oversight of proposed ACRS activities, with increased use of identified review objectives and action plans providing an assessment of resource use)

ACTIONS: An ACRS Action Plan is being developed. The P&P Subcommittee has been culling and prioritizing proposed ACRS activities in its monthly reviews. The Chairman and Vice Chairman have been meeting with and communicating with individual Commissioners to obtain their input.

7) Test and refine streamlined process for ACRS review of license renewal application. (Plant License Renewal Subcommittee)

ACTIONS: A process has been developed and discussed with the Committee and will be refined taking into account the experience gained in the ANO 1 and Hatch reviews.

8) Take actions to maintain and improve ACRS awareness of plant operations issues. (Larkins, Savio, and Plant Operations Subcommittee)

ACTIONS: The ACRS continues to have plant operating events briefings and to make a annual visit to a Region office and a operating plant. The ACRS met with a representative of UCS during the September 2000 and will meet with this individual again during the October 2000 ACRS meetings to discuss a recent UCS report on the impact of the current increased focus on the use of PRA on the safety of plant operations. The ACRS will meet with NEI representatives and discuss issues of mutual interest (risk- informing 10CFR Part 50, license renewal, decommissioning, and the revised reactor oversight process) during the October 2000 ACRS meeting. The issues of mutual interest to be discussed were selected by NEI from a longer list provided by the ACRS/ACNW office and were identified by NEI as being the four main elements of NEI's program of regulatory reform.

9) Solicit and address feedback on how annual research report can be made more useful to Commission and staff. (Safety Research Subcommittee)
ACTIONS: This was done and the feedback was used to structure the current annual ACRS review of NRC-sponsored research.

January 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
	1 New Year's Day (govt closed)	2	3	4	5	6
7 National Academy of Science	8 National Academy of Science	9 National Academy of Science	10 National Academy of Science	11 National Academy of Science	12	13
14 payperiod 3	15 Martin Luther King Jrs. Birthday (govt closed)	16 124th ACNW Meeting?	17 124th ACNW Meeting?	18 124th ACNW Meeting?	19	20
21	22	23	24	25	26	27
28 payperiod 4	29	30	31			

10/6/2000

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February 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
				1 479th	2 ACRS Meeting	3
4	5	6	7	8	9	10
11 payperiod 5 Minerals, Metals & Materials Society	12	13	14 Valentine's Day	15	16	17
18	19 Presidents Day (govt closed)	20	21	22	23	24
25 payperiod 6	26	27	28			

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10/6/2000

March 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
				1 480th ACRS Meeting	2	3
4	5	6	7	8	9	10
11 payperiod 7	12 Regulatory Information Conference	13 125th ACNW Meeting?	14	15	16	17
18	19	20	21	22	23	24
25 payperiod 8	26	27	28	29	30	31

10/6/2000

April 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
1	2	3	4	5 481st ACRS Meeting	6	7 Passover
8 payperiod 9	9	10	11	12	13	14
15 Easter	16	17 126th ACNW Meeting?	18	19	20	21
22 payperiod 10	23	24	25	26	27	28 American Physical ...
29 American Physical Society	30					

10/6/2000

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May 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
		1 American Physical Society	2	3	4	5
6 payperiod 11	7 American Nechanical Engineers	8 American Mechanical Engineers	9	10 482nd ACRS Meeting	11	12
13 Mother's Day	14	15 Nuclear Energy Institute	16	17	18	19
20 payperiod 12	21	22 127th ACNW? (new)	23	24	25	26
27	28 Memorial Day (govt closed)	29	30	31		

10/6/2000

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June 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
					1	2
3 payperiod 13	4	5	6 483rd ACRS Meeting	7	8	9
10 American Health Physics	11	12	13	14	15	16
17 payperiod 14 American Nuclear Society	18	19	20	21	22	23
24	25	26	27	28	29	30

10/6/2000

July 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
1 payperiod 15	2	3	4 Independence Day (govt closed)	5	6	7
8	9	10	11 484th ACRS Meeting	12	13	14
15 payperiod 16	16	17	18	19	20	21
22	23	24 128th ACNW? (new)	25	26	27	28
29 payperiod 17	30	31				

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10/6/2000

August 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
			1	2	3	4
5	6	7	8	9	10	11
12 payperiod 18	13	14	15	16	17	18
19	20	21	22	23	24	25
26 payperiod 19	27	28	29	30	31	

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10/6/2000

September 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
						1
2	3 Labor Day (govt closed)	4	5 485	6 th ACRS	7 Meeting	8
9 payperiod 20	10	11	12	13	14	15
16	17	18 Rosh Hashanah	19 Rosh Hashanah	20	21	22
23 payperiod 21	24	25 129th	26 ACNW	27 Yom Kippur Meeting? (new)	28	29
30						

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10/6/2000

October 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
	1	2	3	4 486th ACRS Meeting	5	6
7 payperiod 22	8 Columbus Day (govt closed)	9	10	11	12	13
14	15	16 130th ACNW Meeting	17 (Las Vegas?)	18	19	20
21 payperiod 23	22	23	24 Water Reactor Safety Conference	25	26	27
28	29	30	31			

10/6/2000

November 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
				1	2	3
4 payperiod 24	5	6	7	8 487th	9 ACRS Meeting	10
11	12 Veteran's Day (govt closed) American Institute for Chemical Engineers	13 131st	14 ACNW Meeting	15	16	17
18 payperiod 25	19	20	21	22 Thanksgiving Day (govt closed)	23	24
25	26	27	28	29	30	

10/6/2000

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December 2001

ACRS/ACNW CALENDAR FOR 2001

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
						1
2 payperiod 26	3	4	5	6 488th ACRS Meeting	7	8
9	10 Chanukah	11	12	13	14	15
16 payperiod 1	17	18	19	20	21	22
23	24	25 Christmas Day (govt closed)	26	27	28	29
30 payperiod 2	31					

10/6/2000