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

September 10, 2002

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

Dear Chairman Meserve:

SUBJECT: SUMMARY REPORT - 494th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, JULY 10-12, 2002, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 494th meeting, July 10-12, 2002, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report and letters. In addition, the Committee authorized Dr. John T. Larkins, Executive Director, ACRS, to transmit the memorandum noted below:

REPORT:

The following report was issued to Chairman Meserve, NRC, from George E. Apostolakis, Chairman, ACRS:

- Draft Final Revision 1 to Regulatory Guide 1.174 and to Chapter 19 of the Standard Review Plan, dated July 23, 2002

LETTERS:

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

- Draft Advanced Reactor Research Plan, dated July 18, 2002

- Risk Metrics and Criteria for Reevaluating the Technical Basis of the Pressurized Thermal Shock Rule, dated July 18, 2002

MEMORANDUM:

The following memorandum was issued to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

- Draft Regulatory Guide DG-1119 (Proposed Revision 1 to Regulatory Guide 1.180), "Guidelines for Evaluating Electromagnetic and Radio-frequency Interference in Safety-related Instrumentation and Control Systems," dated July 16, 2002

HIGHLIGHTS OF KEY ISSUES

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

The Committee heard presentations by and held a discussion with the NRC staff regarding the status of the staff's work to identify risk metrics and criteria that can be used in reevaluating the technical basis of the PTS rule. The status of the staff's work is provided in SECY-02-0092 (May 30, 2002).

The staff stated that reactor vessel failure frequency (RVFF) will be used as a key metric and that the criterion for RVFF will be established in a manner consistent with previous NRC consideration of core damage frequency (CDF) and large early release frequency (LERF) in other risk-informed initiatives. The staff intends that the criterion selected will establish a risk level and contribution to risk from RVFF that are consistent with the intent of the current rule.

The staff has developed three options for RVFF acceptance criteria. Pilot studies covering these options are progressing concerning the four selected plants and a draft report has been completed for one plant. Uncertainties in the individual pilot plant studies, safety margins, and the application of defense in depth need to be evaluated. Methods need to be established for extension of the insights gained from pilot plant studies to all pressurized water reactors. The staff plans to continue its interactions with stakeholders and the international community. Reevaluation of the technical basis for the PTS rule is scheduled for completion in December 2002.

Committee Action

The Committee issued a letter to William Travers, Executive Director for Operations, NRC, on this matter, dated July 18, 2002, observing that the acceptance criteria proposed by the NRC staff do not properly reflect the potential impact of an air oxidation source term on risk. The Committee plans to hold additional discussions with the staff on the PTS rule reevaluation project.

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

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning the draft final Revision 1 of Regulatory Guide 1.174 and associated changes to Chapter 19 of the Standard Review Plan (SRP), "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decision-making: General Guidance".

The staff stated that Revision 1 includes several changes, including the following:

- Risk-related information may now be requested if new, unforseen hazards or a substantially greater prospect for a known hazard emerges.

- Increases in power level, fuel burnup rates, and the use of MOX fuel may affect plant risk profiles and consequently current risk parameters, such as LERF. Therefore, a cautionary note has been added to Revision 1 to note that the staff is evaluating their impact on LERF.

The staff stated that rather than including staff endorsements of consensus industry standards and industry peer review process associated with PRA quality in Regulatory Guide 1.174, the staff is in the process of developing a separate regulatory guide to address the industry standards and peer review process. The staff plans to periodically update Regulatory Guide 1.174 to incorporate lessons learned from ongoing regulatory issues.

Committee Action

The Committee issued a report to Chairman Meserve on this matter, dated July 23, 2002, recommending that Revision 1 to Regulatory Guide 1.174 and the associated

SRP Chapter 19 not be issued until more substantive changes are made. Publications of the current revisions may send the wrong message that incomplete PRAs are acceptable for a broad range of risk-informed changes to the licensing basis.

3. Meeting with the NRC Commissioners

The ACRS members met with the NRC Commissioners on July 10, 2002, to discuss the following items of mutual interest:

- Core power uprates/license renewal activities/future ACRS activities
- Risk-informing special treatment requirements of 10 CFR Part 50
- Advanced reactors
- Pressurized thermal shock technical basis reevaluation project

In a Staff Requirements Memorandum, dated July 17, 2002, the Commission asked:

> The ACRS should consider providing a recommendation as to how license renewal guidance documentation should be updated to reflect supporting information, particularly with regard to time-limited aging analyses, that should, as a minimum, be included in license renewal applications to maximize the efficiency of the review process and minimize requests for additional information.

The Committee plans to discuss the Commission request during its September 12-14, 2002 meeting.

4. Risk-Informed Regulation Implementation Plan

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning progress on implementation of risk-informed regulation. The staff provided plans to improve the process for planning activities associated with implementation of risk-informed regulations. The staff summarized agency progress to date, including the reactor oversight process, risk-based performance indicators, risk-informed changes to 10CFR50.46 (acceptance criteria for ECCS), risk-informed changes to 10CFR73.55 (reactor security programs), etc. The staff also reported progress on current activities related to risk-informing special treatment requirements at commercial power plants.

Committee Action

This was an information briefing only. No committee action was taken.

5. Advanced Reactor Research Plan

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the draft advanced reactor research plan. The reactor designs that are being considered within the scope of the plan are the Pebble Bed Modular Reactor (PBMR), Gas Turbine-Modular Helium Reactor (GT-MHR), AP1000, and the International Reactor Innovative and Secure (IRIS). Generation IV reactor concepts have not been included in the plan, due to their preliminary stage of development. The plan, however, is expected to be a living document, and will be updated later on to accommodate Generation IV reactor concepts, and other designs such as the European Simplified Boiling Water Reactor (ESBWR), boiling water reactor (SWR-1000), and Advanced CANDU Reactor (ACR-700).

The draft plan originates from a technology-neutral perspective. It is structured to capture both technical and potential safety issues that involve uncertainties. The RES staff envisions the plan to be used in identifying key research areas, tools to address technical issues, key research outputs results and links to the regulatory process, and key milestones and resources over a 5-year period. The plan, however, does not delineate the research that will be conducted by the NRC staff. Rather, it identifies information gaps that exist at the NRC in terms of analytical tools and data.

Committee Action

The Committee issued a letter to the NRC's Executive Director for Operation, dated July 18, 2002, commending the RES staff on its effort. The Committee noted that the plan is not yet complete in the sense that it does not establish resources, schedules, and milestones. Nevertheless, the Committee believed that addressing the research needs already identified in the plan to be very important.

6. Overview of NRC Research Activities in the Seismic Area

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding past, present, and future research activities in the areas of earth sciences and earthquake engineering. The discussions included a review of funding and budgeting for the research, studies, and cooperative efforts that have been performed with other organizations to gather seismographic information. In addition the staff discussed probabilistic seismic hazard assessments and seismic margins programs.

Committee Action

This briefing was for information only. The Committee will use information obtained during this briefing to prepare input to the annual ACRS report to the Commission on the NRC Safety Research Program.

7. Development of Review Standard for Reviewing Core Power Uprate Applications

The Committee heard a presentation and held discussions with representatives of the NRC staff regarding NRR's plans for development of a Review Standard to be applied to applications for extended power uprates (EPU). Development of this Review Standard is intended to respond, in part, to concerns expressed by the ACRS as a result of the Committee's review of several EPU applications over the past year. The Standard is to provide: a clearer definition of the staff's scope of review, development of technical review criteria, process guidance, and, model, or template, safety evaluations. The staff's schedule calls for issuance of the Review Standard for interim use and public comment in December of this year. ACRS review of the Standard is expected following issuance for public comment.

Committee Action

As this briefing was for information only, no Committee action was taken at this time.

8. Application of the Probabilistic Fracture Mechanics Methodologies to Reactor Vessel Integrity Assessment

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning development of the FAVOR probabilistic fracture mechanics (PFM) computer code and its use for assessing reactor pressure vessel integrity. A brief overview description of the FAVOR code capabilities was presented, followed by a detailed description of code validation efforts stemming from participation in a number of bench-marking exercises.

Committee Action

This was an information briefing only. No committee action was taken.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee considered the response from the EDO, dated May 6, 2002, to comments and recommendations included in an ACRS report dated March 14, 2002, concerning the Arkansas Nuclear One, Unit 2, Extended Power Uprate, and the response dated May 13, 2002, to the ACRS report concerning risk-informing the special treatment requirements of 10 CFR Part 50 (Option 2).

  The Committee decided that it has continuing concerns relative to the EDO responses in the above letters. Specifically, the ACRS believes that Human Reliability Models should be reviewed, particularly as they are being utilized by the NRC. Also, uncertainty analysis should be performed as part of any risk analysis. The Committee plans to discuss these issues during future meetings.

- The Committee considered the EDO's response of June 25, 2002, to comments and recommendations included in the ACRS report dated May 8, 2002, concerning the PHEBUS-FP Program.

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

- The Committee considered the EDO's response of June 11, 2002, to comments and recommendations included in the ACRS report dated March 14, 2002, concerning Confirmatory Research Program on High-Burnup Fuel.

  The Committee plans to continue its discussions with the NRC staff regarding this issue during future meetings.

- The Committee considered the EDO's response of June 6, 2002, to comments and recommendations included in the ACRS report dated April 17, 2002, concerning the GE Nuclear Energy Licensing Topical Report NEDC-33004P, "Constant Pressure Power Uprate," Revision 1.

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

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee will continue to meet with the staff on a periodic basis to discuss progress in seismic research.

- The Committee will continue its review of the proposed resolution of GSI-185 during its September 12-14, 2002 meeting.

- The Committee plans to hold additional discussions with the NRC staff regarding the PTS rule reevaluation project during future meetings.

- The Committee has continuing concerns regarding the EDO response of May 6, 2002 to the ACRS report of March 14, 2002 relative to the ANO, Unit 2 extended power uprate and the EDO response to May 13, 2002 to the ACRS report of March 19, 2002 concerning risk-informing the special treatment requirements of 10 CFR Part 50 (Option 2). The Committee plans to continue its discussion of this matter during future meetings.

- The Committee plans to review the draft final version of DG-1119, "Guidelines for Evaluating Electromagnetic and Radiofrequency Interference in Safety-Related Instrumentation and Control Systems," after reconciliation of public comments.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from June 1, 2002 through July 9, 2002, the following Subcommittee meetings were held:

- Fire Protection - June 4, 2002

The Subcommittee reviewed (1) the proposed revision to 10 CFR 50.48 to endorse the National Fire Protection Association (NFPA) standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," and (2) the Nuclear Energy Institute (NEI) guidance document NEI-00-01, "Guidance for Post-Fire Safe-Shutdown Circuit Analysis."

- Materials and Metallurgy and Plant Operations - June 5, 2002

The Subcommittees discussed the Root Cause Report and Repair Plan associated with the Davis-Besse vessel head degradation, the status of licensees' response to Bulletin 2002-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," and data used to support findings related to the control rod drive mechanism (CRDM) penetration cracking and the reactor pressure vessel (RPV) head degradation.

- Plant Operations and Fire Protection - June 19, 2002

The Subcommittees discussed the performance of the plants in Region II (Atlanta, GA) including fire protection issues and other plant related issues.

- Thermal-Hydraulic Phenomena - June 26, 2002

The Subcommittee reviewed portions of RES's Thermal-Hydraulic Research Program. Specific topics discussed included the Phase Separation Test Program being conducted in the Air-Water Test Loop for Advanced Thermal-Hydraulic Studies (ATLATS) test facility, the status of the TRAC-M code consolidation and documentation effort, and the Reflood Test Program being conducted at Pennsylvania State University. The proposed resolution of GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs," was also reviewed.

- Future Plant Designs - July 8, 2002

The Subcommittee reviewed the proposed Advanced Reactor Research Plan and its implication on the NRC's Regulatory framework.

- Plant License Renewal - July 9, 2002

The Subcommittee reviewed the Virginia Electric and Power Company's (Dominion's) license renewal application for Surry Power Station Units 1 and 2, and North Anna Power Station Units 1 and 2, and the associated safety evaluation report with open items.

- Planning and Procedures - July 9, 2002

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

PROPOSED SCHEDULE FOR THE 495th ACRS MEETING

The Committee agreed to consider the following topics during the 495th ACRS meeting, which will be held on September 12-14, 2002:

- DOE/DOD Naval Reactors Virginia Class Nuclear Propulsion Plant Submarine Design (Closed)

- Human Reliability Analysis Research Plan
- Proposed Resolution of Generic Safety Issue-185, "Control of Recriticality Following Small-Break LOCAs in PWRs"
- Subcommittee Report Regarding D.C. Cook Switchyard Fire
- Subcommittee Report Regarding the Reactor Oversight Process
- Proposed 10 CFR 50.69, Draft Regulatory Guide DG-1121, and NEI Document NEI 00-04
- Draft Regulatory Guide DG-1120 and Standard Review Plan Section Associated with NRC Code Reviews
- Subcommittee Report on Fire Protection

Sincerely,

George E. Apostolakis
Chairman

CERTIFIED

Date Issued: 9/3/2002
Date Certified: 9/16/2002

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JULY 10-12, 2002

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APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
IV. Future Agenda and Subcommittee Activities
V. List of Documents Provided to the Committee

CERTIFIED

MINUTES OF THE 494th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
JULY 10-12, 2002
ROCKVILLE, MARYLAND

The 494th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on July 10-12, 2002. Notice of this meeting was published in the *Federal Register* on June 24, 2002 (65 FR 42589) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

A transcript of selected portions of the meeting is available in the NRC Public Document Room at the One White Flint North Building, Mail Stop 1F-15, Rockville, MD, 20852-2738. [Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc., 1323 Rhode Island Avenue, NW, Washington, DC 20005-3701, and on the ACRS/ACNW Web page at (www.NRC.gov/ACRS/ACNW).]

ATTENDEES

ACRS Members: ACRS Members: Dr. George Apostolakis (Chairman), Dr. Mario V. Bonaca (Vice Chairman), Dr. F. Peter Ford, Dr. Thomas S. Kress, Mr. Graham M. Leitch, Dr. Dana A. Powers, Dr. Victor H. Ransom, Mr. Stephen L. Rosen, Dr. William J. Shack, Mr. John D. Sieber, and Dr. Graham B. Wallis. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

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

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

II. Pressurized Thermal Shock (PTS) Reevaluation Project: Risk Acceptance Criteria (Open)

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

Dr. Thomas Kress, cognizant ACRS Member for this discussion, introduced the topic to the Committee. He stated that the purpose of this discussion was to review the status of the staff work on the development of a technical basis for risk-informed selections of PTS screening criteria to be used in a reevaluation of this current PTS rule and to provide what advice the ACRS believed was needed. The status of the NRC staff's work is provided in SECY-02-0092 (May 30, 2002).

The NRC staff presenters were Michael Mayfield, Nathan Siu, and Mark Kirk from the Office of Nuclear Regulatory Research (RES). The current PTS rule (10 CFR 50.61) provides screening criteria that are used to make a decision on the allowed degree of reactor pressure vessel (RPV) embrittlement. Regulatory Guide (RG) 1.154 contains guidelines for safety analyses that can be used to justify continued operation of the screening criteria. The staff is reevaluating the technical basis for 10 CFR 50.61 in light of the lessons learned in the use of 10 CFR 50.61 and RG 1.154, and research performed since their issuance. Studies are being performed for four pilot plants (Oconee, Palisades, Beaver Valley, and Calvert Cliffs). Insights from these studies will be used to evaluate the performance of other PWR plants. The staff plans to utilize its interactions with stakeholders and the international community in this work. Reevaluation of the technical bases for the PTS rule is scheduled for completion in December 2002.

The staff has developed three options for screening acceptance criterion for the allowed degree of reactor pressure vessels embrittlement. The staff is using reactor vessel failure frequency (RVFF) as its key metric. The criterion selected will be established in a manner consistent with NRC considerations of core damage frequency (CDF) and large early release frequency (LERF) in other risk-informed actions. ACRS members expressed concern as to the relation of criterion selections to consideration of the significant differences in the source term (i.e., air oxidation) likely to be associated with a reactor pressure failure event.

The staff has developed its three options for allowable criteria for RVFF based on existing agency work. The first (RVFF = 5×10^6/yr) is based on the pressurized thermal shock (PTS) acceptance criteria provided in Regulatory Guide 1.154. The second and third (RVFF = 1×10^5/yr and RVFF = 1×10^6/yr) are based respectively on the CDF and LERF criteria provided in Regulatory Guide 1.174 and the Option 3

framework for risk-informing 10 CFR 50. Evaluations of uncertainty and proper application of defense-in-depth are critical issues in making this selection. The staff is currently assessing the need for and the feasibility performing a scoping study on post-reactor pressure vessel failure scenarios.

Committee Action

The Committee issued a report to Chairman Meserve on this matter dated July 18, 2002, stating that the acceptance criteria proposed by the NRC staff does not properly reflect the potential impact of air oxidation source term on risk. The Committee plans to hold additional discussions with the staff on the PTS rule reevaluation project.

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

[Note: Mr. Paul Boehnert was the Designated Federal Official for this portion of the meeting. Mr. August W. Cronenberg was the cognizant staff engineer for this portion of the meeting.]

Dr. George E. Apostolakis, Chairman of the Reliability and PRA Subcommittee, stated that the Committee would discuss the staff's intention to publish Revision 1 to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and associated changes to Chapter 19 of the Standard Review Plan (SRP), "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decision-making: General Guidance." He stated that the staff was requesting a letter from the Committee for concurrence to release these revisions. Dr. Apostolakis then introduced Ms. Mary Drouin and Mr. John Lane of RES to lead off the discussion.

Ms. Drouin began the staff presentation by noting that the purpose of the briefing was to provide the ACRS with an update on the final version of Revision 1 to RG 1.174 and to request concurrence from ACRS on the release of the regulatory guide and associated changes to Chapter 19 of the SRP. She noted that the intention was to update the guide and Chapter 19 on an annual basis, and in this regard the regulatory guide can be considered a "living document." She noted that NRC's policy statement on probabilistic risk assessment (PRA) encourages the use of PRA techniques to improve safety decision making and regulatory efficiency. RG 1.174 provides guidance on the use of PRA to support licensee requests for risk-informed changes to a plant's licensing basis.

Dr. Apostolakis asked the Committee and the presenters to look at Figure 1 (Figure title: Principles of Risk-Informed Integrated Decisionmaking) on page 7 of RG 1.174. He noted that there should be a sixth box in the figure for "safety culture," which is an important attribute of risk-informed decision-making. He asked why it had not been included. Ms. Drouin responded that she agreed that it should, and probably would be included at a future time. She again noted that the intention was to update the guide on an annual basis.

ACRS member, Mr. Rosen, asked Ms. Drouin to comment on Slide-3 of her presentation regarding the statement of on-going lessons learned from Davis-Besse and the intention to periodically update the regulatory guide. Ms. Drouin responded that the staff has not come to any conclusions on Davis-Besse, but that if there were lessons regarding risk implications from that event, then such lessons would be factored into future revisions of the regulatory guide. Ms. Drouin then passed the presentation on to Mr. John Lane, also sitting at the speakers table.

Mr. Lane described to the Committee the proposed changes to the regulatory guide and Chapter 19 of the SRP. He noted that the revisions would allow the staff to request risk information for any license amendment request for a change in the licensing basis, if a new or unforseen hazard emerges associated with that change. He cited, as examples, an increase in core power level, fuel burnup rate, or use mixed-oxide fuel, and how such changes could impact the risk profile or LERF.

Dr. Kress noted that the original purpose for the proposed changes to RG 1.174 was to allow for small changes in the risk profile, but it now seems that RG 1.174 has become the paradigm or model for a host of changes to a plant's licensing, even if the risk impact is not so small. He noted that a good PRA is required to capture the impact of such a license change on risk. He stated that a clear definition of acceptable PRA quality was not evident from his review of the proposed changes to RG 1.174 and SRP Chapter 19.

Dr. Apostolakis followed by asking the staff and ACRS to turn to page 19-14 of the SRP (i.e., Page 14 of Chapter 19 of the SRP), where he quoted from the first paragraph of that page, "Although the assessment of risk implications (in light of the acceptance guidelines defined in RG 1.174) requires that all plant operating modes and initiating events be addressed, it is not necessary in risk-informed regulation that licensees submit PRAs that treat all plant operating modes and all initiating events." He stated that such statements go out of their way to accommodate poor quality PRA submittals by licensees. He said that the agency should instead be encouraging quality PRA submittals for any risk-informed request for changes to a plant license. At this point Dr.

Apostolakis indicated that he was inclined to recommend that Revision 1 to RG 1.174 not be released. He stated that this would allow the current regulatory guide to stand. He also stated that this would allow time for the staff to re-work such revisions to include statements that encourage quality PRAs in support of any risk-based requests for a change to a plant's license.

Dr. Apostolakis also pointed to page 21 of the SRP/Chapter-19 document, and noted that similar language is evident in regard to uncertainty analysis, i.e., "...this does not imply that a detailed propagation of uncertainties is always necessary, in many cases it is possible to show that a point estimate is an acceptable approximation of the mean value using qualitative arguments about the risk contributors." He stated that such language undermines the purpose of the regulatory guide and that the ACRS might want to consider a position of not supporting release of the proposed revisions to the regulatory guide.

IV. Meeting with the NRC Commissioners (Open)

The ACRS members met with the NRC Commissioners on July 10, 2002, to discuss the following items of mutual interest:

- Core power uprates/license renewal activities/future ACRS activities
- Risk-informing special treatment requirements of 10 CFR Part 50
- Advanced reactors
- Pressurized thermal shock technical basis reevaluation project

In a Staff Requirements Memorandum, dated July 17, 2002, the Commission asked:

> The ACRS should consider providing a recommendation as to how license renewal guidance documentation should be updated to reflect supporting information, particularly with regard to time-limited aging analyses, that should, as a minimum, be included in license renewal applications to maximize the efficiency of the review process and minimize requests for additional information.

The Committee plans to discuss the Commission request during its September 12-14, 2002 meeting.

V. Risk-Informed Regulation Implementation Plan (Open)

[Note: Mr. Howard J. Larson was the Designated Federal Official for this portion of the meeting. Mr. August W. Cronenberg was the cognizant staff engineer for this portion of the meeting.]

Dr. George E. Apostolakis, Chairman of the Reliability and PRA Subcommittee, stated that the committee would hear presentations by representatives of RES and the Office of Nuclear Reactor Regulation (NRR) staff concerning progress on the Risk-Informed Regulation Implementation Plan (RIRIP).

Mr. Mark Cunningham, RES, began the self-introductions, followed by Mr. Michael Johnson, Mr. William Becker and Mr. Christopher Grimes, all of NRR. Mr. Cunningham started the presentation with an overview of the recently formulated plan by RES for improving the process for implementation of risk informed regulations. He summarized the Agency progress to date, including the proposed rule-making for risk-informed changes to 10 CFR 50.46 regarding acceptance criteria for emergency core cooling system (ECCS), the development of a plan for improving coherence among risk-informed agency activities (which includes reactor as well as material and waste activities), and finally risk-information associated changes to technical specifications.

At this point, Dr. Apostolakis asked if "safety culture" considerations are to be included in the Risk-Informed Regulation Implementation Plan (RIRIP) considerations, to which Mr. Cunningham responded yes, although it was not modeled in current PRAs, but nevertheless considered in regulatory implementation. Mr. Cunningham then went on to summarize upcoming activities in the reactor safety arena including risk-informed changes to 10 CFR 50.69 (Special Treatment Requirements), risk-informing 10 CFR 50.44 and 50.46, risk-informing fire protection rules, development of a "Coherence Plan" for risk-informed activities, risk-management technical specification initiatives, and development of a RG and SRP to assess PRA adequacy.

Mr. Michael Johnson, NRR, followed Mr. Cunningham with a presentation regarding NRR plans for improving "coherence" among risk-management activities. Included in his presentation was Stu Magruder (NRR) and Mary Drouin (RES). Mr. Magruder stated that the Staff Requirements Memorandum for a "Coherence Plan" for risk-informed activities required the staff to provide a plan for moving forward with risk-informed regulation to address regulatory structure convergence with the risk-informed process. Mr. Magruder went on to state that many stakeholders believe that the present approach to risk-informing regulations is inconsistent and that the goal of the "Convergence Plan" was a common understanding of risk-informed regulatory objectives.

Dr. Apostolakis asked Mr. Magruder what was meant by "a common understanding of risk-informed regulatory objectives." Mr. Grimes, NRR, stated that the staff was working to a common set of performance standards that stakeholders and staff can buy into or accept as reasonable. Mr. Rosen interjected that his experience was that "a common

understanding of objectives" could not be accomplished by talking alone, what needs to be done is that you have to show a "benefit" to all concerned; a benefit to plant operator performance; a benefit to overall plant performance; and, a benefit to the utility in either safety or operation. He also noted that to get "buy-in" you generally need "paced implementation" so as to not overwhelm the stakeholders on new ways of doing things. Mr. Grimes stated that the staff may decide to have a workshop explaining the "Convergence Plan" and solicit input from stakeholders on the plan. He also agreed with Mr. Rosen's observations on "paced implementation," which Mr. Grimes called "moving the cheese."

Mr. William Beckner, NRR, began his presentation on "Risk-Management Technical Specifications" and discussed an overview of efforts in this regard. He went on to list eight efforts, such as technical specification action end states, missed surveillance, mode change flexibility, completion times for configuration risk management program, and non-technical specification support features. Mr. Beckner mentioned that in some cases a plant's technical specifications might require going to "cold shutdown," while in actuality "hot shutdown" might be a more risk adverse mode. Dr. Powers questioned what type of risk analysis is used to assess a risk at cold shutdown versus hot shutdown modes and to decide which is most risk-adverse. Mr. Beckner stated that it is generally not inferred from a full-blown risk analysis, but rather based on an assessment of such things as equipment availability and decay heat removal. Dr. Powers responded that this approach was not a good answer, that you could not determine the true risk from such limited information. Mr. Leitch, ACRS Member, asked if Initiative-4 regarding configuration risk management referred to plant outages; to which Mr. Beckner responded yes.

NRR and RES closed the session with an overview of additional activities related to risk-informing the regulatory process, including progress on PRA standards development and industry guidance, as well as risk-informing the review process for advanced reactors.

VI. Advanced Reactors Research Plan

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

The Committee heard presentations by representatives of the RES regarding the draft advanced reactor research plan. The reactor designs that are being considered within the scope of the plan are the Pebble Bed Modular Reactor (PBMR), Gas Turbine-Modular Helium Reactor (GT-MHR), AP1000, and the International Reactor Innovative and Secure (IRIS). Generation IV reactor concepts have not been included in the plan,

due to their preliminary stage of development. The plan, however, is expected to be a living document, and will be updated later on to accommodate Generation IV reactor concepts, and other designs such as European Simplified Boiling Water Reactor (ESBWR), Boiling Water Reactor (SWR-1000), and the Atomic Energy of Canada Limited (ACR-700).

The draft plan originates from a technology-neutral perspective. It is structured to capture both technical and potential safety issues that involve uncertainties. RES envision the plan to be used in identifying key research areas, tools to address technical issues, key research output results and links to the regulatory process, and key milestones and resources over a 5-year period. The plan, however, does not delineate the research that will be conducted by the NRC staff. Rather, it identifies information gaps that exist at the NRC in terms of analytical tools and data.

Committee Action

The Committee issued a memorandum to the Executive Director for Operations, dated July 18, 2002, commending the RES staff on its effort. The Committee noted that the plan is not yet complete in the sense that it does not establish resources, schedules, and milestones. Nevertheless, the Committee believed that addressing the research needs already identified in the plan is very important. The Committee noted several specific comments regarding the plan.

VII. Overview of NRC Research Activities in the Seismic Area (Open)

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

The Committee heard presentations by representatives of the NRC staff regarding past, present, and future research in the areas of earth sciences and earthquake engineering. The discussions included a review of funding and budgeting for the research, studies that have been performed, and cooperative efforts with other organizations to gather seismographic information. In addition the staff discussed probabilistic seismic hazard assessments and seismic margins programs.

Committee Action

This briefing was for information only. The Committee will use information obtained during this briefing to support preparation of NUREG-1635, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," Vol. 5.

VIII. Development of Review Standard for Reviewing Core Power Uprate Applications

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

Dr. G. Wallis, cognizant ACRS Member for this issue, introduced this topic to the Committee. He noted that the Maine Yankee Lessons Learned Report contained a recommendation for development of a Standard Review Plan for power uprate applications. Based on the experience gained from several EPU reviews, it has been recognized that uprate reviews might be more efficient if the staff's review criteria were more explicit. The Committee also commented to this effect. NRR has now initiated development of a Review Standard to that end. Dr. Wallis also said that no action on this matter appears necessary at this time, unless the Committee believes NRR's approach is untoward.

NRC Staff Presentation

Mr. M. Shuaibi, NRR, discussed the staff's development of a proposed Review Standard for EPU applications. Issues discussed included:

- Background
- Comments from ACRS
- Elements of a Review Standard
- Benefits of a Review Standard
- Schedule
- Concluding Remarks

NRR noted that as a result of comments from the ACRS and the results of the staff's "lessons learned" during the recent EPU reviews, it was decided that a Review Standard would benefit the staff by: establishment of a comprehensive guidance document, retention of institutional knowledge, updating existing review criteria, improving focus, consistency and completeness of review, and improving review documentation.

The Review Standard is to provide: a clearer definition of the staff's scope of review, development of technical review criteria, process guidance, and, model, or template, safety evaluations. The staff's schedule calls for issuance of the Review Standard for interim use and public comment in December of this year. ACRS review of the Standard is expected following issuance for public comment.

In response to Committee questions, NRR said that the Standard will be a living document and will be updated/revised as seen necessary. NRR said that the Standard will include input from the Resident Inspectors. Committee Members requested that NRR provide the draft Review Standard for its review. As a result of Committee discussion, NRR said that it will be looking at the role that risk analysis plays in future EPU reviews. Members also indicated that the staff's effort here looks promising.

Committee Action

As this briefing was for information only, no Committee action was taken at this time.

IX. Application of the Probabilistic Fracture Mechanics Methodologies to Reactor Vessel Integrity Assessment (Open)

[Note: Mr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting. Mr. August W. Cronenberg was the cognizant staff engineer for this portion of the meeting.]

Dr. F. Peter Ford, Chairman of the Materials and Metallurgy Subcommittee, stated that the Committee would hear presentations by the RES staff and contract personnel from Oak Ridge National Laboratory (ORNL) on activities related to the development and verification of probabilistic fracture mechanics (PFM) methods used for assessment of reactor pressure vessel (RPV) integrity. He stated that this was an information briefing only, and that no letter was requested by the staff.

Mr. Michael Mayfield, RES, and manager of the PFM efforts. He then introduced the three presenters, Dr. Mark Kirk, RES, and Drs. Robert Bass and Claud Pugh of ORNL. He stated that the presentation would center on a brief description of the PFM capabilities, with a more in-depth description of recent code validation efforts and results from participation in international bench-marking exercises.

Dr. Kirk followed with an overview of agency PFM work in support of the PTS rule (10 CFR 50.61) and more fundamental work on the linear elastic fracture mechanics (LEFM) models and their applicability to the assessment of reactor pressure vessel (RPV) failure probability assessments under PTS conditions.

Dr. Apostolakis asked what was meant by "CPI and CPF." Dr. Kirk responded that CPI means conditional probability of initiation (i.e., crack initiation) and CPF means conditional probability of failure (i.e., vessel failure).

Mr. Kirk's presentation concluded with a discussion regarding the logic structure of the Agency's PFM-LEFM code (i.e., FAVOR code). He discussed the code looping/logic structure and its relation to flaw propagation and growth, and if un-arrested vessel failure.

The next speaker, Dr. Claud Pugh introduced himself as a retired manager of the heavy-section metals department at ORNL. He stated that he had primary responsibility for management of the experimental validation efforts for PFM-LEFM codes, such as FAVOR and its forerunner OCA. He initiated his talk with a quote from a 1965 ACRS letter, which indicated the need to demonstrate that RPV failure was indeed an "incredible event". He indicated that this letter was the stimulus for initiation of experimental and analytical efforts to demonstrate RPV integrity under a wide range of accident and accident recovery conditions, including PTS conditions. He then cited an early ORNL report (ORNL-NSIC-21, Dec. 1967), which outlined a plan of research for assessment/validation of RPV integrity.

Dr. Ransom, ACRS Member, asked what was meant by "incredible". Dr. Pugh said that this could be taken as something like a RPV failure probability of 10^{-6} failures per year. Dr. Kress questioned Slide-7, "The Failure Pressures Measured During Intermediate-Scale Vessel Tests Were Predicted Well by LEFM." Mr. Pugh stated that it was a full-scale RPV of a 4-loop type, and was used by PNL (Pacific Northwest Laboratory) to characterize the flaw-size distribution in a typical reactor vessel.

The remainder of Mr. Pugh's presentation centered on a discussion of validation efforts. The next several slides showed plots of temperature versus failure pressure or temperature versus charpy energy. Dr. Wallis asked what was meant by "the lower shelf." RES stated that in a temperature versus charpy energy plot there are two distinct flat regions with a transition region in between the "lower shelf" refers to the lower/flat region of the plot. The presentation continued with various data versus OCA and FAVOR predictions, demonstrating good agreement between experimental data and predictions.

Dr. Bass, ORNL, continued the presentation with a focus on experimental validation efforts particular to PTS conditions. His presentation demonstrated that the experimental observations on crack growth and arrest behavior is well described by LEFM methods as embodied in the OCA code and the FAVOR code. He closed his presentation with the suggestion that if the Agency decides to proceed with any additional FAVOR validation efforts, there were a number of European experiments related to PTS validation that would be a good choice for comparison with FAVOR predictions.

Dr. Kirk summarized the session with remarks about the NRC research programs establishing the validity of the FAVOR methodology/code. He also stated that the supporting empirical data has made it possible to assess RPV fracture resistance under both routine operation and accident conditions using linear elastic fracture mechanics (LEFM). Such LEFM predictions of crack initiation and failure agree with the results from prototypic RPV experiments, indicating that LEFM is an appropriate methodology for use in assessments of RPV fracture resistance and integrity. He also indicated the need for continuation of a baseline PFM/LEFM capability within the agency, and cited licensee requests for removal of flux inhibitors from reactor vessels, which reduce irradiation associated RPV embrittlement. Mr. John Sieber, ACRS Member, stated that such flux inhibitors were basically fresh fuel with a high neutron-adsorption which are put at the core periphery to reduce neutron irradiation to the vessel. He also stated that one could envision a license application for a power uprate (with power flattening) that might ask for an exemption from the use of periphery fresh fuel assemblies as a flux inhibitor.

X. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations

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

- The Committee considered the response from the EDO, dated May 6, 2002, to comments and recommendations included in an ACRS report dated March 14, 2002, concerning the Arkansas Nuclear One, Unit 2, Extended Power Uprate, and the response dated May 13, 2002, to the ACRS report concerning risk-informing the special treatment requirements of 10 CFR Part 50 (Option 2).

  The Committee decided that it has continuing concerns relative to the EDO responses in the above letters. Specifically, the ACRS believes that Human Reliability Models should be reviewed, particularly as they are being utilized by the NRC. Also, uncertainty analysis should be performed as part of any risk analysis. The Committee plans to discuss these issues during future meetings.

- The Committee considered the EDO's response of June 25, 2002, to comments and recommendations included in the ACRS report dated May 8, 2002, concerning the PHEBUS-FP Program.

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

- The Committee considered the EDO's response of June 11, 2002, to comments and recommendations included in the ACRS report dated March 14, 2002, concerning Confirmatory Research Program on High-Burnup Fuel.

  The Committee plans to continue its discussions with the NRC staff regarding this issue during future meetings.

- The Committee considered the EDO's response of June 6, 2002, to comments and recommendations included in the ACRS report dated April 17, 2002, concerning the GE Nuclear Energy Licensing Topical Report NEDC-33004P, "Constant Pressure Power Uprate," Revision 1.

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

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

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on July 9, 2002. The following items were discussed:

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

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

Anticipated Workload for ACRS Members

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

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate

- Manage the members' workload for these meetings

- Plan and schedule items for ACRS discussion of topical and emerging issues

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

Quadripartite Meeting Update

The Quadripartite meeting is scheduled to be held on October 23-25, 2002, in Berlin, Germany. As confirmed at the April meeting, the following technical papers were prepared by cognizant members for discussion at the Quadripartite meeting:

- Safety Culture and Safety Management
- Risk-Informed Regulation
- Thermal-Hydraulic Analysis and Code Issues
- Stress Corrosion Cracks in Pressure Retaining Components in Nuclear Power Plants
- Risk Analysis of Spent Fuel Storage

In connection with the Quadripartite meeting trip, nine members agreed to visit a MOX facility in France prior to the meeting in Berlin, Germany.

Celebration of the 500th ACRS Meeting

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

As decided by the Committee at the June 2002 meeting, invitations were sent to Mr. Ralph Beedle, NEI, and Mr. Bert Wolf, GE, to participate in the celebration.

Workshop on Nuclear Regulatory Decisionmaking Process in Switzerland

The Swiss Federal Safety Inspectorate (HSK), in coordination with the International Atomic Energy Agency (IAEA), is organizing a Workshop on Nuclear Regulatory Decisionmaking Process to be held in Switzerland on October 13-16, 2002. As a result of the Planning and Procedures Subcommittee discussion in April, the Committee

asked the Executive Director to contact HSK to see if the workshop on Nuclear Regulatory Decisionmaking Process could be scheduled for October 16-18, 2002, instead of October 13-16. The HSK has accommodated our request. Additionally, they have included ACRS participation in two places on the agenda. One would be a presentation on ACRS Perspective on Regulatory Decisionmaking and the second is a Panel Discussion. At the May meeting, the Committee approved Drs. Bonaca, Kress, and Ransom to participate at this Workshop. In addition, the Committee stated that Dr. Kress, with assistance from Dr. Apostolakis, should prepare a paper for Committee review and presentation at the workshop, on the ACRS Perspectives on Regulatory Decisionmaking. Dr. Bonaca should represent the ACRS on the Panel discussion regarding Issues and Opportunities for the Evaluation and Improvement of Nuclear Regulatory Decisionmaking Process. Dr. Kress plans to provide a draft paper for Committee review at the September ACRS meeting.

Role and Use of PRA in the ACRS Review Process

During the June 2002 meeting, the Committee discussed the issues raised by several members regarding the role and use of PRA in the ACRS review of regulatory issues (e.g., power uprates, license renewal, etc.). The Committee decided that a "White Paper" addressing the role and use of PRA in the ACRS review process would be helpful, and suggested that the ACRS Executive Director get a specialist under contract to prepare a draft "White Paper."

EDO Responses to ACRS Reports

During the June 2002 meeting, the Committee discussed the EDO response of May 8, 2002 to the comments and recommendations included in the ACRS report, dated March 19, 2002 regarding Risk-Informing Special Treatment Requirements of 10 CFR Part 50 (Option 2). In addition, the Committee discussed the May 6, 2002 EDO response to the ACRS report of March 14, 2002 regarding Arkansas Nuclear One, Unit 2 Extended Power Uprate. In both cases, members were not satisfied with certain aspects of the EDO response. Dr. Apostolakis agreed to propose a course of action after further discussion of this matter.

Subcommittee Report

The Thermal-Hydraulic Phenomena Subcommittee held a meeting on June 26, 2002. The Subcommittee discussed: (1) selected RES thermal-hydraulic research programs; to wit: the Phase Separation Research Program being conducted at Oregon State University (OSU), the RES TRAC-M code consolidation and documentation programs, and the Rod Bundle Heat Transfer test Program; and (2) proposed resolution of GSI-185, "Control of Recriticality Following Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors."

Regarding GSI-185, the Subcommittee recommended that RES perform additional work to verify the thermal-hydraulic modeling assumptions pertaining to the "ex-vessel mixing" methodology employed to determine the extent of boron dilution seen in the vessel for the case of natural circulation flow.

Regarding the OSU Phase Separation Research Program, the Subcommittee agreed that additional work is necessary. Specifically, RES needs to verify the applicability to a full-scale plant of the correlation developed by OSU for modeling of entrainment in a horizontal pipe with an upward-oriented branch line.

Report on the Visit to Watts Bar Nuclear Plant and Region II

Members of the ACRS Subcommittees on Plant Operations and Fire Protection toured the Watts Bar Nuclear Plant on June 18 and held a meeting with Region II personnel on June 19, 2002, to discuss operational and fire protection issues. The Planning and Procedures Subcommittee stated that a report by the Chairman of the joint Subcommittees on Plant Operations and Fire Protection is suggested as information to aide the members and the ACRS staff who did not participate in the tour or the meeting.

Members Projected to Exceed the 130 Legal Day Limit

Issues have been raised regarding some members exceeding the 130 legal day limit. Because members are classified as "Special Government Employees," we are required by law to plan work so that each member stays within the 130 legal day limit. After reviewing the compensation records, it appears that 4 of the 11 members may exceed the 130 legal day limit. There may be unique situations which require a member to exceed the limit, however, the "Special Government Employee" status of the member will change after exceeding the 130 legal day limit. This will have impact on member's outside consulting/contract activities.

Member Time and Labor

As discussed at the last Planning and Procedures Subcommittee meeting, reporting compensation time by labor category is critical to the office operation. The members must use the agency's labor accounting codes when reporting their time.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 495th ACRS Meeting, September 12-14, 2002.

The 494th ACRS meeting was adjourned at 6:30 p.m. on July 12, 2002.

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

September 3, 2002

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 494th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - JULY 10-12, 2002

Enclosed are the proposed minutes of the 494th 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

September 16, 2002

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

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

SUBJECT: CERTIFIED MINUTES OF THE 494th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), JULY 10-12, 2002

I certify that based on my review of the minutes from the 494th ACRS full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

George E. Apostolakis, Chairman

September 16, 2002
Date

attorneys or law firms, state or local governments, and substate regional planning and coordination agencies which are composed of substate areas and whose governing boards are controlled by locally elected officials.

The RFP, containing the grant application, guidelines, proposal content requirements and specific selection criteria, is available at *http://www.ain.lsc.gov*. Descriptions of the New Jersey service areas are available at *http://www.ain.lsc.gov*. LSC will not fax the solicitation package to interested parties.

Issue Date: June 18, 2002.

Michael A. Genz,

Director, Office of Program Performance.

[FR Doc. 02–15836 Filed 6–21–02; 8:45 am]

BILLING CODE 7050–01–P

NATIONAL ARCHIVES AND RECORDS ADMINISTRATION

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

AGENCY: National Archives and Records Administration (NARA).

ACTION: Notice.

SUMMARY: NARA is giving public notice that the agency has submitted to OMB for approval the information collection described in this notice. The public is invited to comment on the proposed information collection pursuant to the Paperwork Reduction Act of 1995.

DATES: Written comments must be submitted to OMB at the address below on or before July 24, 2002 to be assured of consideration.

ADDRESSES: Comments should be sent to: Office of Information and Regulatory Affairs, Office of Management and Budget, Attn: Ms. J. Zieher, Desk Officer for NARA, Washington, DC 20503.

FOR FURTHER INFORMATION CONTACT: Requests for additional information or copies of the proposed information collection and supporting statement should be directed to Tamee Fechhelm at telephone number 301–837–1694 or fax number 301–837–3213.

SUPPLEMENTARY INFORMATION: Pursuant to the Paperwork Reduction Act of 1995 (Public Law 104–13), NARA invites the general public and other Federal agencies to comment on proposed information collections. NARA published a notice of proposed collection for this information collection on April 8, 2002 (67 FR 16766 and 16767). No comments were received. NARA has submitted the described information collection to OMB for approval.

In response to this notice, comments and suggestions should address one or more of the following points: (a) Whether the proposed information collection is necessary for the proper performance of the functions of NARA; (b) the accuracy of NARA's estimate of the burden of the proposed information collection; (c) ways to enhance the quality, utility, and clarity of the information to be collected; and (d) ways to minimize the burden of the collection of information on respondents, including the use of information technology. In this notice, NARA is soliciting comments concerning the following information collection:

Title: Request Pertaining to Military Records.

OMB number: 3095–0029.

Agency form number: SF 180.

Type of review: Regular.

Affected public: Veterans, their authorized representatives, state and local governments, and businesses.

Estimated number of respondents: 552,500.

Estimated time per response: 5 minutes.

Frequency of response: On occasion (when respondent wishes to request information from a military personnel record).

Estimated total annual burden hours: 46,042 hours.

Abstract: In accordance with rules issued by the Department of Defense (DOD) and Department of Transportation (DOT, US Coast Guard), the National Personnel Records Center (NPRC) of the National Archives and Records Administration (NARA) administers military service records of veterans after discharge, retirement, and death. When veterans and other authorized individuals request information from or copies of documents in military service records, they must provide in forms or in letters certain information about the veteran and the nature of the request. Federal agencies, military departments, veterans, veterans' organizations, and the general public use Standard Forms (SF) 180, Request Pertaining to Military Records, in order to obtain information from military service records stored at NPRC. The authority for this information collection is contained in 36 CFR 1228.168(b).

Dated: June 13, 2002.

L. Reynolds Cahoon,

Assistant Archivist for Human Resources and Information Services.

[FR Doc. 02–15783 Filed 6–21–02; 8:45 am]

BILLING CODE 7515–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on July 10–12, 2002, in Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, November 26, 2001 (66 FR 59034).

Wednesday, July 10, 2002

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

8:35 a.m.–10 a.m.: Pressurized Thermal Shock (PTS) Reevaluation Project: Risk Acceptance Criteria (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the risk metrics and associated criteria that can be used in reevaluating the technical basis of the PTS rule.

10:15 a.m.–11:15 a.m.: Draft Final Revision 1 to Regulatory Guide 1.174, "An Approach to Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and Associated Standard Review Plan Chapter 19 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final revision 1 to Regulatory Guide 1.174 and the associated Standard Review Plan Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance."

11:15 a.m.–12:45 p.m.: Discussion of Topics for Meeting with the NRC Commissioners (Open)—The Committee will discuss topics for meeting with the NRC Commissioners on July 10, 2002.

2 p.m.–4 p.m.: Meeting with the NRC Commissioners (Open)—The Committee will meet with the NRC Commissioners, Commissioners' Conference Room, One White Flint North, to discuss the following:

- Overview

—Core Power Uprates and License Renewals

—Future Committee Activities

- Advanced Reactors
- Risk-Informing Special Treatment Requirements of 10 CFR Part 50

• Pressurized Thermal Shock Technical Basis Reevaluation Project

4:15 p.m.–5:15 p.m.: Risk-Informed Regulation Implementation Plan (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the updated version of the Risk-Informed Regulation Implementation Plan.

5:30 p.m.–7:15 p.m.: Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Thursday, July 11, 2002

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

8:35 a.m.–10 a.m.: Advanced Reactors Research Plan (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the modifications and updates to the Advanced Reactors Research Plan.

10:15 a.m.–12 Noon: Overview of NRC Research Activities in the Seismic Area (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding ongoing and proposed research activities as well as new research needs in the seismic area.

1 p.m.–2:30 p.m.: Development of Review Standard for Reviewing Core Power Uprate Applications (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the development of a "Review Standard" for use in future reviews of Core Power uprate applications.

2:45 p.m.–6 p.m.: Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Friday, July 12, 2002

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

8:35 a.m.–10:15 a.m.: Application of the Probabilistic Fracture Mechanics Methodologies to Reactor Vessel Integrity Assessment (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding application of the probabilistic fracture mechanics methodologies (including the FAVOR computer code) to assess reactor pressure vessel integrity.

10:30 a.m.–2:30 p.m.: Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

2:45 p.m.–3:30 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

3:30 p.m.–3:45 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

3:45 p.m.–4:45 p.m.: Format and Content of the 2003 ACRS Report on the NRC Safety Research Program (Open)—The Committee will discuss the format, content, schedule, and assignments for the 2003 ACRS report to the Commission on the NRC Safety Research Program.

5 p.m.–6 p.m.: Proposed Papers for the Quadripartite Meeting (Open)—The Committee will discuss proposed technical papers on specific topics that will be presented at the Quadripartite meeting scheduled to be held on October 23–25, 2002, in Berlin, Germany.

6 p.m.–6:30 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 3, 2001 (66 FR 50462). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify the Associate Director for Technical Support named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Associate Director prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Associate Director if such rescheduling would result in major inconvenience.

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

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

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

Dated: June 18, 2002.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 02–15859 Filed 6–21–02; 8:45 am]

BILLING CODE 7590–01–P

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

June 18, 2002

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

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

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

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

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

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

3) ~~10:15~~ 10:20 - 11:15 A.M. Draft Final Revision 1 to Regulatory Guide 1.174, "An Approach to Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and Associated Standard Review Plan Chapter 19 (Open) (GEA/AWC/PAB)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final revision 1 to Regulatory Guide 1.174 and the associated Standard Review Plan Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance."

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

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

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

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

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

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

6) 4:15 - ~~5:15~~ 5:25 P.M. Risk-Informed Regulation Implementation Plan (Open) (GEA/AWC/HJL)

6.1) Remarks by the Subcommittee Chairman

6.2) Briefing by and discussions with representatives of the NRC staff regarding the updated version of the Risk-Informed Regulation Implementation Plan.

~~5:15~~ - ~~5:30~~ 5:25-5:40 P.M. *****BREAK*****

7) ~~5:30~~ - ~~7:15~~ 5:40-6:45 P.M. Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

7.1) PTS Reevaluation Project: Risk Acceptance Criteria (TSK/FPF/MWW) 5:40-6:45

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

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

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

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

9) 8:35 - ~~10:00~~ 10:10 A.M. Advanced Reactors Research Plan (Open) (TSK/MME)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding the modifications and updates to the Advanced Reactors Research Plan.

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

10) ~~10:15~~ - ~~12:00~~ 10:30-12:07 Noon Overview of NRC Research Activities in the Seismic Area (Open) (DAP/TJK/SD)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff regarding ongoing and proposed research activities as well as new research needs in the seismic area.

~~12:00~~ - ~~1:00~~ 12:07-1:10 P.M. *****LUNCH*****

11) 1:10 - ~~1:00~~ - 2:30 P.M. Development of Review Standard for Reviewing Core Power Uprate Applications (Open) (GBW/JDS/PAB)

11.1) Remarks by the Subcommittee Chairman

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

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

12) 2:45 - 4:45 ~~6:00~~ P.M. Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

4:10-4:45 12.1) PTS Reevaluation Project: Risk Acceptance Criteria (TSK/FPF/MWW)

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

2:40-3:15 12.3) Risk-Informed Regulation Implementation Plan (GEA/AWC/HJL)

3:35-4:10 12.4) Advanced Reactors Research Plan (TSK/MME)

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

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

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

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

14.1) Remarks by the Subcommittee Chairman

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

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

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

(~~12:00~~ - ~~1:00~~ P.M. LUNCH) 12:10 - 1:10 Continue discussion of proposed ACRS reports listed under Item 12.

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

16) 3:20-4:40 ~~2:45~~ - ~~3:30~~ P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL/SD)

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

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

17) 4:40 - 4:45 ~~3:30 - 3:45~~ P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GEA, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

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

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

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

20) 6:00 - 6:30 P.M. Miscellaneous (Open) (GEA/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

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

APPENDIX III: MEETING ATTENDEES

494th ACRS MEETING
JULY 10-12, 2002

NRC STAFF (July 10, 2002)
N. Siu, RES
L. Lois, NRR
M. Mayfield, RES
S. Coffin, NRR
M. Mitchell, NRR
T. Otsu, RES
J. Cai, RES
L. Kim, RES
R. Woods, RES
S. Newberry, RES
S. Malik, RES
J. Hyslop, RES
S. Dinsmore, NRR
E. Lois, RES
G. Parry, NRR
A. Singh, RES
D. Harrison, NRR
B. Hardin, RES
H. Hamzehee, RES
M. Rubin, NRR
A. Rubin, RES
C. Grimes, NRR
M. Johnson, NRR
J. Smith, NMSS
B. Leslie, NMSS
B. Dennig, NRR
W. Beckner, NRR
M. Drouin, RES
A. Velazquez, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
R. Huston, Licensing Support Services

NRC STAFF (July 11, 2002)

J. Muscara, RES
J. Flack, RES
D. Snowberger, NRR
P. Reed, RES
R. Lee, RES
D. Carlson, RES
R. Tripathi, RES
S. Arndt, RES
S. Rubin, RES
C. Greene, RES
A. Snyder, RES
E. Throm, NRR
M. Drouin, RES
M. Srinivason, RES
E. Trager, RES
P. Bennett, RES
A. Rubin, RES
A. Drozo, NRR
H. Graves, RES
A. Levin, OCM/RAM
J. Costello, RES
A. El-Bassioni, NRR
S. Ali, RES
J. Wilson, NRR
R. Kenneally, RES
A. Velazquez, NRR
S. Koenick, NRR
D. Dorman, RES
F. Eltawila, RES
N. Choksi, RES
J. Eads, NRR
J. Zwolinski, NRR
J. Zimmerman, NRR
D. Harrison, NRR
R. Jenkins, NRR
D. Spaulding, NRR
M. Shuarbi, NRR
S. Bajwa, NRR
C. Moyer, RES
R. Caruso, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

G. George, PA Consulting Group
M. Schoppman, NEI

NRC STAFF (July12, 2002)
T. Valentine, NRR
N. Chokshi, RES
S. Malik, RES
M. Marshall, NRR
S. Arndt, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
T. Dicksen, Oak Ridge National Laboratory
P. Williams, Oak Ridge National Laboratory
B. Bass, Oak Ridge National Laboratory

APPENDIX IV

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

August 19, 2002

SCHEDULE AND OUTLINE FOR DISCUSSION
495th ACRS MEETING
SEPTEMBER 12-14, 2002

THURSDAY, SEPTEMBER 12, 2002, NRC AUDITORIUM, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. Opening Statement by the Acting ACRS Chairman (Closed) (MVB/JTL)

2) 8:35 - 11:45 A.M. DOE/DOD Naval Reactors Virginia Class Nuclear Propulsion Plant
(10:00-10:20 A.M.- BREAK) Submarine Design (Closed) (JDS/PAB)

2.1) Remarks by the Subcommittee Chairman
2.2) Briefing by and discussions with representatives of the Department of Energy (DOE)/Department of Defense (DOD) Naval Reactors Organization and the NRC staff regarding the Virginia Class Nuclear Propulsion Plant Submarine Design.

[NOTE: The entire session will be closed to discuss classified information applicable to this matter.]

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

THURSDAY, SEPTEMBER 12, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

3) 1:00 - 1:05 P.M. Opening Remarks by the ACRS Chairman (Open)

3.1) Opening Statement (GEA/JTL/SD)
3.2) Items of current interest (GEA/SD)

4) 1:05 - 2:30 P.M. Human Reliability Analysis Research Plan (Open) (DAP/MME/AWC)

4.1) Remarks by the Subcommittee Chairman
4.2) Briefing by and discussions with representatives of the NRC staff regarding the Human Reliability Analysis Research Plan and related matters.

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

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

5) 2:45 - 4:15 P.M. Proposed Resolution of Generic Safety Issue-185, "Control of Recriticality Following Small-Break LOCAs in PWRs" (Open) (VHR/PAB)

5.1) Remarks by the Cognizant ACRS Member

5.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed recommendations by the Office of Nuclear Regulatory Research for resolving Generic Safety Issue-185.

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

4:15 - 4:30 P.M. ***BREAK***

6) 4:30 - 5:00 P.M. Subcommittee Report Regarding D.C. Cook Switchyard Fire (Open) (JDS/MWW)
Report by the Chairman of the ACRS Subcommittee on Plant Operations regarding the Switchyard fire event that occurred at the D. C. Cook Nuclear Power Plant on June 12, 2002.

7) 5:00 - 5:30 P.M. Subcommittee Report Regarding the Reactor Oversight Process (Open) (JDS/GEA/MWW)
Report by the Chairman of the ACRS Subcommittee on Plant Operations regarding matters discussed at the September 9, 2002 Subcommittee meeting.

8) 5:30 - 7:00 P.M. Proposed ACRS Reports (Open/Closed)
Discussion of proposed ACRS reports on:

8.1) Human Reliability Analysis Research Plan (DAP/MME/AWC)

8.2) Proposed Resolution of GSI-185 (VHR/PAB)

8.3) Virginia Class Nuclear Propulsion Plant Submarine Design (Closed) (JDS/PAB) [Room T-8E8]

FRIDAY, SEPTEMBER 13, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

10) 8:35 - 10:45 A.M. Proposed 10 CFR 50.69, Draft Regulatory Guide DG-1121, and NEI Document NEI 00-04 (Open) (GEA/AWC/HJL)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding proposed 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," DG-1121, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," and NEI 00-04, "10 CFR 50.69 SSC Categorization Guidelines."

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

11) 11:00 - 12:30 P.M. Draft Regulatory Guide DG-1120 and Standard Review Plan Section Associated with NRC Code Reviews (Open) (GBW/PAB/MWW)

11.1) Remarks by the Subcommittee Chairman

11.2) Briefing by and discussions with representatives of the NRC staff regarding draft Regulatory Guide DG-1120, "Transient and Accident Analysis Methods," and draft final Standard Review Plan Section 15.0.2, "Review of Transient and Accident Analysis Methods."

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

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

12) 1:30 - 2:00 P.M. Subcommittee Report on Fire Protection (Open) (SR/TJK/SD)
Report by the Chairman of the ACRS Subcommittee on Fire Protection regarding matters discussed during the September 11, 2002 Subcommittee meeting.

13) 2:00 - 3:00 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL/SD)

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

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

14) 3:00 - 3:15 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GEA, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

3:15 - 3:30 P.M. *****BREAK*****

15) 3:30 - 7:00 P.M. Preparation of ACRS Reports (Open/Closed)
Discussion of proposed ACRS reports on:

15.1) Human Reliability Analysis Research Plan (DAP/MME/AWC)

15.2) Proposed Resolution of GSI-185 (VHR/PAB)

15.3) Virginia Class Nuclear Propulsion Plant Submarine Design (Closed) (JDS/PAB) [Room T-8E8]

15.4) Proposed 10 CFR 50.69, DG-1121, and NEI 00-04 (GEA/AWC/HJL)

15.5) Draft Regulatory Guide DG-1120 and SRP Section Associated with NRC Code Reviews (GBW/PAB/MWW)

SATURDAY, SEPTEMBER 14, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

16)	8:30 - 12:00 Noon	Preparation of ACRS Reports (Open) Continue discussion of the proposed ACRS reports listed under Item 15.
	12:00 -1:00 P.M.	*****LUNCH*****
17)	1:00 - 1:30 P.M.	Miscellaneous (Open) (GEA/JTL) Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

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

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
494th ACRS MEETING
JULY 10-12, 2002

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

MEETING HANDOUTS

AGENDA ITEM NO.	DOCUMENTS
1	Opening Remarks by the ACRS Chairman 1. Items of Interest, dated July 10-12, 2002
2	Pressurized Thermal Shock (PTS) Reevaluation Project Risk 2. Pressurized Thermal Shock Rule (10 CFR 50.61) Re-Evaluation Risk Acceptance Criteria - Status Report presentation by RES [Viewgraphs]
3	Draft Final Revision 1 to Regulatory Guide 1.174, "An Approach to Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and Associated Standard Review Plan Chapter 19 3. Status of Regulatory Guide 1.174 presentation by M. Drouin, J. Lane, RES [Viewgraphs] 4. Safety Culture: a survey of the state-of-the-art, paper by J. Sorensen [Handout] 5. Risk-Informed Licensing for Advanced Reactors, presentation at the 6th International Conference on Probabilistic Safety Assessment and Management, San Juan, Puerto Rico, June 23-28, 2002 [Handout]
6	Risk-Informed Regulation Implementation Plan 6. Risk-Informed Regulation Implementation Plan presentation by RES and NRR [Viewgraphs]
9	Advanced Reactors Research Plan 7. Advanced Reactor Research Plan presentation by J. Flack, RES [Viewgraphs]
10	Overview of NRC Research Activities in the Seismic Area 8. Earth Sciences & Earthquake Engineering presentation by RES [Viewgraphs] 9. Bibliography, Seismotectonic Program [Handout]
11	Development of Review Standard for Reviewing Core Power Uprate Applications 10. Development of Review Standard for Extended Power Uprates presentation by NRR [Viewgraphs]
14	Application of the Probabilistic Fracture Mechanics Methodologies to Reactor Vessel Integrity Assessment 11. Probabilistic Fracture Mechanics Techniques Used in the Re-evaluation of the

Pressurized Thermal Shock Rule (10 CFR § 50.61) presentation by RES and the Oak Ridge National Laboratory

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

12. Final Draft Minutes of Planning and Procedures Subcommittee Meeting - July 9, 2002 [Handout #16]

17 Reconciliation of ACRS Comments and Recommendations

13. Reconciliation of ACRS Comments and Recommendations [Handout #16]

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	PTS Reevaluation Project: Risk Acceptance Criteria
	1. Table of Contents
	2. Proposed Agenda
	3. Status Report, dated July 10, 2002 [Internal Committee Use Only: Predecisional Material Attached]
	4. SECY-02-0092, "Status Report: Risk Metrics and Criteria for Pressurized Thermal Shock," dated May 30, 2002
	5. Letter from George E. Apostolakis to William D. Travers, "Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule," dated February 14, 2002
	6. Letter from Dana A. Powers to William D. Travers, "Pressurized Thermal Shock Technical Basis Reevaluation Project," dated October 12, 2000
3	Revision 1 to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" and Associated Standard Review Plan Chapter 19
	7. Table of Contents
	8. Proposed Schedule
	9. Status Report
	10. SECY-02-0070 Publication of Revisions 1 to Regulatory Guide 1.174 and SRP Chapter 19 and Notice of a Staff Plan for Endorsing Consensys Probabilistic Risk Assessment Standards and Industry Peer Review Programs
	11. Standard Review Plan, Chapter 19, Revision 1, Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance, April 2002
6	Risk-Informed Regulation Implementation Plan
	12. Table of Contents
	13. Proposed Schedule
	14. Status Report
	15. Draft Information Paper, Update of the Risk-Informed Regulation Implementation Plan
9	Advanced Reactors Research Plan
	16. Table of Contents
	17. Proposed Agenda
	18. Status Report
	19. Draft (Predecisional) Advanced Reactor Research Plan
10	Overview of NRC Research Activities in the Seismic Area
	20. Table of Contents
	21. Proposed Schedule
	22. Status Report
	23. NUREG-1635, "Review and Evaluation of the Nuclear Regulatory Commission

Safety Research Program," Vol. 4, Section II.8, "Civil, Structural, and Seismic Research," dated May 2001

24. Outline of Briefing at the 494th ACRS Meeting on Seismic Program: Current and Future Decision

11 Development of NRC Review Standard for Power Uprate Requests

25. Project Status Report
26. SECY-02-0106, Review of ACRS Recommendation for the Staff to Develop a Standard Review Plan for Power Uprate Reviews dated June 14, 2002

14 Application of Probabilistic Fracture Mechanics Methodologies to Reactor Vessel Integrity Assessment

27. Table of Contents
28. Proposed Schedule
29. Status Report
30. Abstract from "A Review of Large-Scale Fracture Experiments Relevant to Pressure Vessel Integrity Under Pressurized Thermal Shock Conditions"
31. Abstract from "International Comparative Assessment Study of Pressurized Thermal Shock in Reactor Pressure Vessels
32. Abstract from Fracture Analysis of Vessels - Oak Ridge, Theory and Implementation of Algorithms, Methods, and Correlations

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
494th FULL COMMITTEE MEETING

JULY 10-12, 2002

July 10, 2002
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
NATHAN SIU	NRC/RES/DRAA/PRAB
Lambros Lois	NRR/DSSA/SRXB
Michael Mayfield	RES/DET
Stephanie Coffin	NRR/DE
Matthew A. Mitchell	NRR/DE/EMCB
Tomoko Jensen-Otsu	RES/DRAA
June Cai	RES/DRAA
Lance Kim	RES/DRAA/PRAB
Roy Woods	RES/DRAA/PRAB
Scott Newberry	RES DRAA.
SHAH MALIK	RES DET
J.S Hyslop	
Stephen Dinsmore	NRR/DSSA
Erasmia Lois	RES/DRAA
Garth Parry	NRR/DSSA
Amarjit Singh	RES/PRAB
Donnie Harrison	NRR/DSSA/SPSB
Brad Hardin	RES/PRAB
HOSSEIN HAMZEHEE	RES/DERAB
Mark Rubin	NRR/DSSA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
494th FULL COMMITTEE MEETING

JULY 10-12, 2002

July 10, 2002
Today's Date

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NAME	NRC ORGANIZATION
ALAN RUBIN	NRC/RES/DRAA
Chris Grimes	NRR/DRIP/RPRP
Michael Johnson	NRR/DSSA/SPSB
Jim Smith	NMSS/RTG
Bret Leslie	NMSS/RTG
B.S Dennig	NRR/DRIP/RORP
WILLIAM BECKNER	NRR/DRIP/RORP
Mary Drouin	NRC/RRAB
Alexander Velazquez	NRC/NRR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
494th FULL COMMITTEE MEETING

JULY 10-12, 2002

JULY 10, 2002
Today's Date

ATTENDEES PLEASE SIGN IN FOR ACRS MEETING

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NAME	AFFILIATION
Roger Huston	Licensing Support Services

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
494th FULL COMMITTEE MEETING

JULY 10-12, 2002

July 11, 2002
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

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NAME	NRC ORGANIZATION
JOE MUSCARA	RES/DET/MEB
John FLACK	RES/DSARE/REAHFB
Dwight Summerlegen	NRR/DSSA/SPSB
Phil Reed	RES/DSARE/REPER WMB
RICHARD LEE	RES/DSARE/SMSAB
Don Carlson	" " /REAHFB
Raji Tripathi	RES/DSARE/REAHFB
Steven Arndt	RES/DET/REAB
Stu Rubin	RES/DSARE/REAHFB
Charles Greene	RES/DET/MEB
Amy Snyder	RES/DSARE
EDWARD THROM	NRR/DSSA/SPLB
Mary Drouin	NRC/RES/PRAB
Makuteswara Srinivasan	NRC/RES/DET/MEB
EUGENE TRAGER	NRC/RES/DSARE
Peggy Bennett	NRC/RES/DSARE
ALAN RUBIN	RES/DRAA
ANDRE DROZD	NRR/DSSA
HERMAN GRAVES	RES/ERAB
ALAN LEVIN	OCM/RAM
JF Ostell	RES/ERAB
A. El-Bassioni	NRR/SPSB
S. Ali	NRR/RES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
494th FULL COMMITTEE MEETING

JULY 10-12, 2002

July 11, 2002
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

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NAME	NRC ORGANIZATION
JERRY WILSON	NRR/NRLPO
Roger M Kennelly	RES/DET/ERAB
Alexander Velazquez	NRR/DRIP
Stephen Koenick	NRR/NRLPO
DAN DORMAN	RES/DET/ERAB
FAROUK ELTAWILA	RES/DSARE
Nilesh Chokshi	RES/DET/MEB
Johnny Eads	NRR/DLPM/LPD3
John Zwolinski	NRR/DLPM
Jalle Zimmerman	NRR/DLPM
Donnie Harrison	NRR/DSSA/SPSB
RONALDO V. JENKINS	NRR/NRLPO
Deirdre Spaulding	NRR/DLPM
Mohammed Shuaibi	NRR/DLPM
S. Singh Bajwa	NRR/DLPM/NRR
Carol Moyer	RES/DET/MEB
RALPH CARUSO	NRR/SRXB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
494th FULL COMMITTEE MEETING

JULY 10-12, 2002

JULY 11, 2002
Today's Date

ATTENDEES PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	AFFILIATION
Glenn R. George	PA Consulting Group
Mike Schoppman	NEI

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
494th FULL COMMITTEE MEETING

JULY 10-12, 2002

July 12, 2002
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

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NAME	NRC ORGANIZATION
Theresa Valentine	NRR
Nilesh Chokshi	RES/DET/MEB
SHAH MALIK	RES/DET/MEB
Michael Marshay	NRR/DE/EMCB
Steven Arndt	RES/DET/ERAB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
494th FULL COMMITTEE MEETING

JULY 10-12, 2002

JULY 12, 2002
Today's Date

ATTENDEES PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	AFFILIATION
Terry L. Dickson	Oak Ridge National Laboratory
Paul T. Williams	Oak Ridge National Laboratory
B.R. Bass	ORNL
[redacted]	

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

July 1, 2002

MEMORANDUM TO: Annette L. Vietti-Cook
Secretary of the Commission

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS MEETING WITH THE U. S. NUCLEAR REGULATORY COMMISSION, JULY 10, 2002 - SCHEDULE AND BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners between 2:00 - 4:00 p.m. on Wednesday, July 10, 2002, to discuss the items listed below. Background materials related to these items are attached.

	ESTIMATED TIME
INTRODUCTION - NRC Chairman, Dr. Richard A. Meserve	5 minutes
PRESENTATIONS - Advisory Committee on Reactor Safeguards	
1. Overview - George E. Apostolakis, Chairman, ACRS	20 minutes
• Core Power Uprates	
• License Renewal Activities	
• Future Committee Activities	
2. Advanced Reactors: A Status Report - T. S. Kress	10 minutes
3. Risk-Informing Special Treatment Requirements Of 10 CFR Part 50 - George E. Apostolakis	10 minutes
4. Pressurized Thermal Shock (PTS) Technical Basis Re-evaluation - F. Peter Ford	10 minutes
CLOSING REMARKS	5 minutes

*NOTE: Estimated times are for presentation only and do not include time for Commission Questions and Answers.

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

G. E. Apostolakis

July 10, 2002

Overview

- Core Power Uprates
- License Renewal Activities
- Future Committee Activities

Core Power Uprates

- Recommended 5 approvals:
  - Duane Arnold Energy Center (15.3%)
  - Dresden Units 2 & 3/Quad Cities Units 1 & 2 (17%/17.8%)
  - Arkansas Nuclear One Unit 2 (7.5%)
  - Clinton Power Station Unit 1 (20%)
  - Brunswick Steam Electric Plant Units 1 & 2 (14.3%)

Uprates (Cont'd)

- ACRS reviewed GE Topical Report "Constant Pressure Power Uprate" (CPPU) (4/02)
  - CPPU methodology applied to BWR uprates up to 20% nominal power
  - Committee found CPPU methodology acceptable

Uprates (Cont'd)

- Committee anticipates review of 4-5 uprate applications each in 2003 & 2004. Several other licensees are evaluating the feasibility of uprate applications.

Uprates (Cont'd)

- Committee Review Issues:
  - Lack of adequate documentation in staff safety evaluation reports – issue is being addressed via steadily improving documents
  - Need for staff guidance document on future uprate reviews–pursuant to SRM, staff is developing proposed "Review Standard"

Uprates (Cont'd)

- Core reload safety analyses – NRR performing audits to confirm appropriate use of approved methodology
- Need for staff audit calculations/detailed T/H Models - Staff to consider as part of "Review Standard" development and related activities.

License Renewal Activities

- Current Status
  - Turkey Point review complete
  - Reviews completed for at least one plant from each vendor
  - Interim letters only as needed

License Renewal (Cont'd)

- Upcoming Reviews
  - McGuire and Catawba (1st Ice Condensers)
  - Fort Calhoun (1st Using Generic Guidance Documents)
  - North Anna/Surry/Peach Bottom/ St. Lucie
- Two License Renewal Subcommittees

Future Committee Activities

- Risk-Informed Performance-Based Regulations
- Reactor Operations (including Reactor Oversight Process, Plant Operating Events)
- Safety Research (focus on Advanced Reactors)
- Reactor Fuel (High-Burnup & MOX)

Activities (Cont'd)

- Safeguards/Security
- Fire Protection
- Transient & Accident Code Reviews
- Human Factors
- Safety Culture
- Naval Reactors

Briefing Topics

- Advanced Reactors - T. S. Kress

- Risk-Informing Special Treatment
- Requirements of Part 50-
  G. E. Apostolakis

- Pressurized Thermal Shock
  Technical Basis Re-evaluation
  Project- F. P. Ford

ADVANCED REACTORS: A STATUS REPORT

T. S. Kress

July 10, 2002

ACRS Activities

- Two members participated in the NRC workshop on high temperature gas-cooled reactor safety and research issues
- Main topic at the 2001 ACRS retreat
- ACRS sponsored a workshop on future reactors
  - Identified 24 potential technical issues

ACRS Activities (Cont'd)

- Developing a task action plan to focus Committee review
- Advanced reactors; a main area in the next ACRS research report
- Completed review of policy and technical issues identified by Office of Regulatory Research

Overarching Policy Issues:

- Implementation of Commission's "expectation" that advanced reactors will provide enhanced margins of safety
- Relationship of NRC safety requirements to international safety requirements

Staff Technical Issues

- Event selection and safety classification
- Fuel performance and qualification
- Source term
- Containment versus confinement
- Emergency evacuation

Possible Impediments

- Lack of high-level risk-acceptance criteria other than Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)
- Lack of criteria for selecting design basis accidents
- The appropriate role of defense in depth

Current Activities

- Priority is AP1000
- Working to resolve potential impediments
- Planning to develop "strawman" positions on various issues

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

June 17, 2002

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

Dear Chairman Meserve:

SUBJECT: POLICY ISSUES RELATED TO ADVANCED REACTOR LICENSING

During the 493rd meeting of the Advisory Committee on Reactor Safeguards (ACRS), June 6-8, 2002, we were briefed by representatives of the NRC's Office of Nuclear Regulatory Research (RES) on issues that have potential policy implications for advanced reactor licensing, and the plans for seeking the Commission's guidance for resolving these issues. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

(1) The RES staff has identified appropriate policy issues and posed questions that must be addressed to resolve them.

(2) The existing agency positions on some of these policy issues should be reevaluated because of new perspectives on risk-informed regulation and defense in depth, as well as the new reactor designs that may be proposed.

(3) The need for greater specificity in the application of defense in depth should be made a separate overarching issue.

DISCUSSION

The issues identified by the staff fall into the following five areas:

- event selection and safety classification
- fuel performance and qualification
- source term
- containment versus confinement
- emergency evacuation

We note that in order to resolve these issues, the role of PRA and high-level risk acceptance criteria are essential in the design approval process.

The staff also identified two overarching policy issues:

(1) how to implement the Commission's "expectation" that advanced reactors will provide enhanced margins of safety

(2) what should be the relationship between the NRC's safety requirements and international safety requirements

We recommend that the need for greater specificity in the application of defense in depth should be singled out of the first overarching issue and made a separate and distinct overarching issue. With respect to the second overarching issue, we agree that it would be highly desirable to understand the bases for the international safety requirements. Nonetheless, we note that it would not be unreasonable for different countries to have different safety standards on a cost/benefit basis.

The identification and resolution of these policy issues is important to the process of licensing advanced reactors. The existing agency positions on some of these policy issues should be reevaluated because of new perspectives on risk-informed regulation and defense in depth, as well as the new reactor designs that may be proposed. Much work remains to be done, and we plan to maintain continuing interactions with the staff on possible approaches and options for resolving these policy issues.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Information Paper (Draft Predecisional) dated May 23, 2002, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Plan for Resolving Policy Issues Resulting from Technical Considerations Related to Advanced Reactor Licensing.
2. U.S. Nuclear Regulatory Commission, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," dated June 1988.

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

March 14, 2002

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

Dear Chairman Meserve:

SUBJECT: PHASE 2 PRE-APPLICATION REVIEW FOR AP1000 PASSIVE PLANT DESIGN

During the 490th meeting of the Advisory Committee on Reactor Safeguards (ACRS), March 7-9, 2002, we completed our evaluation of the Phase 2 pre-application review of the Westinghouse AP1000 passive plant design, conducted by the NRC staff. This matter was also reviewed during joint meetings of our Subcommittees on Thermal-Hydraulic Phenomena and Future Plant Designs on February 13-15, 2002, and a meeting of the Subcommittee on Thermal-Hydraulic Phenomena on March 15, 2001. During our review, we had discussions with representatives of the Westinghouse Electric Company and the NRC Staff. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The staff has made a competent and thorough review of the Phase 2 issues.

2. We agree that the proposal by Westinghouse to use Design Acceptance Criteria (DAC) for the piping design should be approved.

3. The staff's positions on the other pre-application review issues should also be approved.

4. The Office of Nuclear Regulatory Research (RES) should further investigate acceptable ranges of ratios of Pi-groups for use in scaling.

5. The ad hoc introduction of compensating processes to tune codes to the integral test data should be discouraged.

Discussion

The NRC staff and Westinghouse have agreed to a three-phased approach to the AP1000 standard plant design review. Phase 1, which was to identify the key review issues, was completed previously and resulted in the identification of four key issues:

1. Acceptability of the proposed use of DAC for particular parts of the design review.

2. Acceptability of certain exemptions that Westinghouse intends to request.

3. Applicability of the AP600 test program to the AP1000 design.

4. Applicability of the AP600 analyses codes to the AP1000 design.

The purpose of the Phase 2 review was for the staff to develop positions on these four key issues. These positions are discussed below.

Proposed Use of DAC

The Commission has determined that the level of detail in a design certification application must be sufficient to enable the Commission to judge the applicant's proposed means of ensuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design.

The staff has interpreted this policy to mean that the certification application must be complete, with two exceptions:

- items for which the technology is rapidly changing and may be significantly different at the combined operating license (COL) stage.

- items for which the level of detail cannot be provided at the time of certification review (or for which the as-procured and as-built characteristics are needed).

For these exceptions, DAC are required of the applicant. Some precedents for DAC satisfying these criteria were established with the certifications of the Advanced Boiling Water Reactor (ABWR) and System 80+ designs. For these, the staff accepted DAC for the instrumentation and control (I&C) and for the control room design, both of which were deemed to satisfy one or more of the above criteria.

In addition to these two areas for which precedents have been established, Westinghouse has proposed DAC for the AP1000 piping design.

The staff has concluded that the DAC approach should be approved for I&C and control room portions of the design based on the two criteria above and that the DAC on piping design should be approved based on the similarity of AP1000 to AP600 designs, for which the certification included sufficient piping design detail.

While we have some sympathy with this view by the staff and agree that the piping DAC should be approved, we believe the piping DAC could have been approved without invoking the similarity to the AP600 design. Our view is that, as long as sufficient detail is available to permit resolution of safety questions, the degree of detail that an applicant wishes to provide at the certification phase is a business decision. We believe the use of DAC for the piping design fits this characterization.

Exemptions

Westinghouse is requesting exemptions from the regulations in three areas:

(a.) Section 50.34 (f)(2)(iv) requires a "safety parameter display console that will display to operators a minimum set of parameters defining the safety status ... displaying a full range of important plant parameters ...., and capable of indicating when process limits are being approached or exceeded."

(b.) Section 50.62(c)(1) requires that equipment be available to ensure the automatic startup of the auxiliary feedwater system under ATWS conditions.

(c.) GDC 17 of 10CFR50 Appendix A requires two physically independent offsite power sources.

The staff agrees with the Westinghouse positions that: Item (a) will be part of the DAC for control room design; the underlying purpose of Item (b) is satisfied because AP1000 does not have (or need) an auxiliary feedwater system as the emergency core cooling system (ECCS) requirement is met by the passive residual heat removal (PRHR) system automatic initiation under ATWS; and that the underlying purpose of Item (c) is satisfied because, with the passive ECCS, AP1000 does not need offsite power to make its safety case. We also agree with these positions.

Applicability of AP600 Standard Plant Design Analysis Codes and Test Program

To address the applicability of the AP600 codes and test program, Westinghouse prepared a new AP1000 phenomena identification and ranking table (PIRT) and conducted new scaling assessments for both the codes and the tests. The AP1000 PIRT resulted in the same high- and medium-ranked phenomena as were found for the AP600, and it was noted that the AP1000 design did not entail any important new phenomena. In addition, the scaling analyses indicated that the Pi-groups identified as being important and which were to be substantially matched in the integral test program were still in the acceptable range when compared to their values for the full-scale AP1000 design. Thus, Westinghouse maintains that these results demonstrate that the AP600 test database used to validate the analysis codes is applicable to AP1000 and that the codes should be approved for use in evaluating the safety status of AP1000 design.

The staff conducted independent top-down and bottom-up scaling assessments and made audit calculations using RELAP5 for a postulated 2-inch diameter break in the cold leg and for a postulated double-ended direct vessel injection (DVI) line rupture. The staff found that, with some noted exceptions, the experimental data produced by the AP600 separate effects and integral effects test programs are appropriate for verification of the processes expected in an AP1000 plant, and the analysis codes validated for the AP600 standard plant design are applicable to the AP1000 design.

The most significant of the exceptions is that the tests are not considered sufficient to validate the entrainment model used in the NOTRUMP code for the upper plenum regions and for the hot-leg exit through the automatic depressurization system (ADS-4) depressurization valve.

Westinghouse claims that the scaling test data and analyses are sufficient to ensure that the core remains covered and that the entrainment is a self-limiting process that decreases as the core water level decreases. Westinghouse also claims that the period during which the entrainment is important in affecting the water level is so short that entrainment is not safety significant. We think such a case can be made during the certification review and, if so, additional tests would not be necessary.

Nonetheless, the staff's position has merit in that it will be necessary to better predict the entrainment behavior before judgments can be made regarding its safety significance. We believe phenomena that are ranked high or medium in importance should be properly treated in the models partly because unanticipated applications could invalidate the "non-safety-important" judgment. We remain concerned that the codes do not properly model entrainment because inapplicable maps are being used to characterize the flow regimes. The use of inapplicable maps could impact the results of the codes in unanticipated ways. Thus, we are convinced that the technical basis codes need better modeling with respect to entrainment and flow regime maps.

Other Considerations

In the scaling assessments, Westinghouse and the staff used the criterion that Pi-group ratios having values between 0.5 and 2.0 represent acceptable scaling. While this range is intuitively pleasing as an indication that the tests sufficiently match the phenomena in AP1000, we have not seen any technical justification for this criterion. Thus, we believe that RES should initiate a study with the objective of establishing a technically based approach for use in determining the significance of any general Pi-group. We think this would involve sensitivity analyses on the Pi-group in the non-dimensional scaling models. The sensitivity of the results to individual Pi-group ratios could guide the selection of acceptance ranges that might be different for different Pi-groups. Although we do not believe that this work is needed for AP1000 certification, this issue is likely to arise with certification of future reactor designs and such a study could tie down this loose end of the code, scaling, applicability, and uncertainty (CSAU) process.

There are two instances in which Westinghouse proposes to adjust its models to provide a better fit to integral data by introducing compensating processes. In one instance, the NOTRUMP code does not model the momentum flux terms in the conservation of momentum equations dealing with effects of area and density changes. This deficiency in the code impacts its ability to calculate pressurizer drainage and reactor vessel downcomer level. To compensate for this code deficiency in the AP600 certification, Westinghouse imposed a reduction in the in-containment refueling water storage tank (IRWST) level – thus reducing the driving force which would conservatively compensate for the effects that would have resulted from having the correct momentum equations. For the AP1000, instead of this same "fix," Westinghouse proposes to use an increased flow resistance penalty that would make the code calculations fit the APEX facility data for a 2-inch small-break loss-of-coolant accident (SBLOCA).

In another instance, Westinghouse concluded that the NOTRUMP PRHR model does not model the thermal plume in the IRWST. The model will over predict the outside surface heat transfer rate for the heat exchanger when the tube flow velocity exceeds 1.5 ft/sec for any

significant period of time. If this situation arises in the analyses, Westinghouse proposes to account for the non-conservative calculation by an ad hoc reduction of the predicted heat exchanger performance.

These temporary fixes should provide conservative results to support the certification of AP1000 design. Nevertheless, we view both of these as instances of purposeful introduction of compensating errors in the codes rather than improving the models. We consider it bad practice to allow these errors to persist in the codes and believe that the actual physics should be properly represented in the long term.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated February 4, 2002, transmitting draft SECY Paper, undated, Subject: Use of Design Acceptance Criteria and Exemptions for the AP1000 Standard Plant Design (Predecisional), and draft SECY Paper, undated, Subject: Applicability of AP600 Standard Plant Design Analysis Codes and Test Program to the AP1000 Standard Plant Design (Predecisional).
2. Memorandum dated June 21, 2000, from John T. Larkins, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: AP1000 Pre-Application Review.

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

September 14, 2000

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

Dear Chairman Meserve:

SUBJECT: PRE-APPLICATION REVIEW OF THE AP1000 STANDARD PLANT DESIGN - PHASE I

During the 475th meeting of the Advisory Committee on Reactor Safeguards, August 29–September 1, 2000, we discussed the results of the staff's pre-application (Phase I) review of the Westinghouse Electric Company's proposed AP1000 Standard Plant Design. During this meeting, we had the benefit of discussions with representatives of the staff and of the documents referenced. A list of our issues that need to be addressed during the AP1000 pre-application review was sent to the NRC Executive Director for Operations on June 21, 2000.

Background

Westinghouse plans to seek certification of a 1000 MWe nuclear plant similar to the certified AP600 design, and seeks NRC feedback on the scope and cost for review and certification of the AP1000 design. The NRC and Westinghouse have agreed to a three-phase review approach. Phase I is to: identify the review assumptions and issues that need to be evaluated; identify the information necessary to evaluate the assumptions and issues; estimate the resources required to perform the Phase II review; and provide a schedule for the certification review.

In a letter dated May 31, 2000, Westinghouse identified five "fundamental assumptions" for evaluation by the staff during Phase II review:

1. The AP1000 Design Certification Application will reference sections of the AP600 Design Control Document that do not change for AP1000.

2. The AP1000 Design Certification Application will not require additional tests to be performed by the applicant.

3. The AP1000 Design Certification Application can utilize the AP600 analysis codes with limited modifications.

4. The AP1000 Design Certification Application can utilize the AP600 probabilistic risk assessment (PRA) supplemented with a sensitivity study to meet the requirements for a plant-specific PRA.

5. The AP1000 Design Certification Application can defer selected design activities to the Combined License (COL) applicant.

In its Phase I assessment, the staff addressed these assumptions and provided Westinghouse with expectations on information that must be provided to the staff to assess the validity of these assumptions.

Recommendations

1. The PRA should include uncertainty distributions on core damage frequency, conditional containment failure probability (CCFP), and large, early release frequency (LERF).

2. The seismic analysis should not be left solely to the COL applicant and should be included in the PRA using a representative site.

3. The applicant's results from the codes NOTRUMP, WCOBRA/TRAC, LOFTRAN, and WGOTHIC for the design basis accidents should be accompanied by uncertainty assessments.

4. The staff should obtain and exercise the above codes to assist its independent evaluation and validation of these codes.

Discussion

The staff has done a commendable job of determining the information it will need to assess the five assumptions proposed by Westinghouse, and we generally agree with the staff's initial positions on these assumptions. We are concerned, however, that the staff may not be requesting sufficient information to conduct the certification review without undue reliance on judgment. Because the applicant does not plan to perform additional tests, certification of the AP1000 will be more dependent on the results of analyses than was the case for the AP600.

In a Staff Requirements Memorandum of July 21, 1993, the Commission approved the use of a CCFP goal of 0.1 along with a containment performance goal for advanced light-water reactor designs. Westinghouse, for points of reference in development of the AP600 PRA, used a LERF goal of 10^{-6} per year as well as the CCFP goal of 0.1.

The AP600 PRA reported an overall LERF of about 10^{-8} per year and a CCFP of about 0.1. While this low value of LERF was comforting, it was based on new systems and components [passive emergency core cooling system (ECCS) combined with active systems, reactor vessel external flooding, etc.] for which there was little experience. Thus, the CCFP and LERF results for the AP1000 are likely to be subject to much greater uncertainty than that associated with current operating plant PRA results. With "reasonable" variation of parameters, the staff estimated that the AP600 CCFP could have easily been 0.5 at a reasonable confidence level. The design changes along with the increased plant size and power rating of the magnitude

proposed will negatively impact both the LERF and the CCFP as well as increase the uncertainties associated with these acceptance parameters.

Increasing the height of the containment and the quantity of water in the tank on top may well increase the vulnerability of the AP1000 containment to seismic events. Both selections of site characteristics and seismicity are challenges to the conduct of a PRA for the AP1000 that includes external event initiators. It is most important that artificial uncertainty not be injected into the PRA results by including bounding ranges of site characteristics and seismicities. A representative site and representative seismicity for the recommended PRA would be satisfactory.

We are concerned that the AP1000 defense in depth associated with a CCFP goal of 0.1 might be unduly compromised by the increase in plant size and the uncertainties could be much greater than those for the AP600. If the staff is to properly assess the AP1000 design with respect to acceptance values of risk metrics and its compliance with the defense-in-depth philosophy, the PRA will need to include an uncertainty analysis. Without such a PRA, we will be faced with insufficient information on which to base our judgment on the defense-in-depth acceptability of the AP1000 containment.

Our second concern relates to the deterministic part of the design certification. The acceptability of the AP600 for certification with respect to the design basis deterministic aspects was partially based on the use of computer codes with validation based on data from separate effects and integral tests.

The AP600 certification was also partially approved on the basis that the scaled integral experiments demonstrated the robustness of the AP600 ECCS for keeping the core covered over the entire period of the design basis accident sequences. It is likely that this level of comfort will be eroded for the AP1000 because of scaling issues that could make the integral tests no longer directly applicable to the full-scale design. Thus, for the AP1000 there will be much greater reliance on the code results. The concern involves, then, the use of codes that have not been validated for the AP1000 conditions to determine margins.

In past licensing reviews, the staff has been content to use a process in which conservative analyses were used to demonstrate that acceptance criteria (e.g., peak clad temperature) could be met. This process could be used because extensive experience and experimental data were available to substantiate the judgment that the analyses were indeed conservative. Extensive experience and data are not available for passive plants. For the AP600, correctly scaled experiments were performed that demonstrated the robustness of the emergency core cooling. If the scaling of these experiments proves to be less satisfactory for the AP1000, greater reliance on thermal-hydraulic codes will be required.

The use of the predictive codes NOTRUMP, WCOBRA/TRAC, LOFTRAN, and WGOTHIC has been approved only for the AP600, and the validity of these codes for application to the AP1000 must be determined. The available experimental data relevant to passive flow conditions may not be sufficient to validate the use of these codes for the AP1000 geometry and conditions. The applicant intends to conduct a detailed scaling analysis to demonstrate the sufficiency of these experimental data for the AP1000.

If the scaling analysis is less than satisfactory, it will be necessary to determine the uncertainties of the predictions of the codes NOTRUMP, WCOBRA/TRAC, LOFTRAN, and WGOTHIC in a technically defensible manner. This could even necessitate additional, properly scaled experiments to provide confidence that the calculated figures of merit are conservative.

In any case, it will be necessary to assess the uncertainty and validation analysis of the codes provided by Westinghouse. The staff should acquire and exercise these codes so that it can independently evaluate the sensitivity of their predictions to assumptions, model idealizations, and choices of parameters in the correlations.

Sincerely,

Dana A. Powers
Chairman

References:

1. Memorandum dated July 27, 2000, from Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, to W. E. Cummins, Westinghouse Electric Company, Subject: AP1000 Pre-Application Review - Phase One.
2. Memorandum dated May 31, 2000, from M. M. Corletti, Westinghouse Electric Company, to Document Control Desk, U. S. Nuclear Regulatory Commission, Subject: AP1000 Pre-Application Review Items.
3. Memorandum dated June 21, 2000, from John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, to William D. Travers, Executive Director for Operations, NRC, Subject: AP1000 Pre-Application Review.
4. Memorandum dated July 21, 1993, from Samuel J. Chilk, Secretary of the Commission, for James M. Taylor, Executive Director for Operations, NRC, Subject: SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.
5. U. S. Nuclear Regulatory Commission, Final Safety Evaluation Report Related to Certification of the AP600 Design, Vol. 2, September 1998.

Risk-Informing Special Treatment Requirements of 10 CFR Part 50

G. E. Apostolakis

July 10, 2002

Risk-Informed Categorization Scheme

	Safety Related	Non-Safety Related
Safety Significant	RISC-1	RISC-2
Not Safety Significant	RISC-3	RISC-4

Previous Comments October 12, 1999 Report

- Terminology of "safety-related" Systems Structures and Components (SSCs) should be preserved
- Significance of importance measures and their limitations
- Recommended guidance to the expert panel

March 19, 2002 Report

Reviewed NEI 00-04/Rev. B

Option 2 Implementation Guideline

<u>Recommendations</u>

- **The criteria used by Integrated Decision-making Panel (IDP) should be explicit and include risk metrics that supplement CDF and LERF (late containment failure; inadvertent radionuclide release)**

March Report (Cont'd)

- A more complete set of risk metrics may allow the elimination of special treatment requirements for class RISC-3.
  - Difficulty in treatment of RISC-3 because risk concerns cannot be completely addressed by CDF and LERF
  - Materials degradation should be considered by IDP

March Report (Cont'd)

- Guidance to IDP should include:

  - Whether SSC acts as barrier to fission product release during severe accidents
  - Whether the SSC is relied upon in Emergency Operating Procedures or Severe Accident Mitigation guidelines

March Report (Cont'd)

- Whether failure of SSC results in an inadvertent radionuclide release

March Report (Cont'd)

- Treatment of uncertainties in PRA results should be made consistent with the current capabilities of PRA software and data.
- When simplified methods are used, comparison with more rigorous analyses should be available to demonstrate the adequacy of these methods

March Report (Cont'd)

- Use of risk information in regulations is still viewed with skepticism by some groups
- Rigor would contribute to building confidence
- Substituting "sensitivity" analysis for uncertainty analysis does not contribute to confidence building

March Report (Cont'd)

- Assessing the impact on CDF and LERF of changing the failure rates by factors ranging from 2 to 5 (in lieu of the South Texas Project factor of 10) needs better justification

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

March 19, 2002

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED RULEMAKING AND ASSOCIATED GUIDANCE FOR RISK-INFORMING THE SPECIAL TREATMENT REQUIREMENTS OF 10 CFR PART 50 (OPTION 2)

During the 490th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 2002, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss the proposed rulemaking and associated guidance for risk-informing the special treatment requirements of 10 CFR Part 50 (Option 2). We discussed the staff's draft rule language for 10 CFR 50.69 and proposed industry guidance in NEI 00-04, Revision B, "Option 2 Implementation Guideline." Our Subcommittee on Reliability and Probabilistic Risk Assessment discussed these matters during meetings on December 4, 2001, and February 22, 2002. We also had the benefit of the documents referenced. This report focuses primarily on the proposed industry guidance in NEI 00-04, Revision B.

Conclusion and Recommendations

1. The criteria used by the Integrated Decision-making Panel (IDP) for categorizing structures, systems, and components (SSCs) should be made explicit and should include consideration of risk metrics that supplement core damage frequency (CDF) and large early release frequency (LERF), such as late containment failure and inadvertent release of radioactive material.

2. Categorization of SSCs performed with a more complete set of risk metrics may allow the elimination of additional treatment requirements for components in the risk-informed safety class 3 (RISC-3) category (safety related, low safety significant).

3. The rigor in the treatment of uncertainties in probabilistic risk assessment (PRA) results should be made consistent with the current capabilities of PRA software and data. When simplified methods are used, comparison with more rigorous analyses should be available to demonstrate the adequacy of these methods.

Discussion

The overall categorization process described in NEI 00-04, Revision B, relies heavily on the judgments of the IDP. The Panel's decision concerning the assignment of an SSC to a risk-informed safety class is based on a variety of qualitative and quantitative inputs. The quantitative inputs are produced by a PRA, if available. A large majority of SSCs are categorized without the benefit of quantitative inputs from a PRA. Two major elements of the categorization process are the risk-informed decision criteria and the processes used by the IDP in making judgments.

In our report dated October 12, 1999, we commented extensively on the decision-making process and the need for guidance and training in conducting expert-panel sessions. Our comments on the processes described in the then-proposed Appendix T to 10 CFR Part 50 remain valid and are a continuing concern. This report focuses on additional issues that warrant attention in the revision of NEI 00-04 to support the proposed 10 CFR 50.69 rulemaking.

The traditional criteria for evaluating risk significance use the metrics CDF and LERF. The initial screening of SSCs for which PRA results are available is carried out by using importance measures that are based on these two metrics. We believe that the probability of late containment failure should be added to CDF and LERF to provide a more complete characterization of risk.

In categorizing SSCs for which PRA results are unavailable, qualitative considerations serve as the primary basis for decisionmaking. Even when PRA results are available, the risk-informed approach requires that the IDP consider qualitative inputs based on defense in depth and safety margins, as articulated by the principles in Regulatory Guide 1.174. NEI 00-04, Revision B, provides very little guidance to assist the Panel in making these qualitative assessments. Explicit criteria should be developed for the qualitative categorization of SSCs and the decision-making process needs to be scrutable with results that can be documented. Guidance to accomplish this should be included in NEI 00-04.

The qualitative considerations used by the IDP should include defense in depth and the traditional graded approach in which relatively frequent events are intended to not fail any of the barriers to the release of radioactivity, but relatively infrequent events are allowed some fuel damage provided that the resulting release is limited by the requirements of 10 CFR Part 100. Specific guidance to the IDP could include requirements for the Panel to determine whether (1) the SSC supports a system that acts as a barrier to fission product release during severe accidents; (2) the SSC is relied upon in the emergency operating procedures or the severe accident management guidelines; and (3) failure of the SSC will result in the inadvertent release of radioactive material even in the absence of severe accident conditions.

If any of the above conditions are true, the IDP should consider including such SSCs in RISC-1 (safety related, safety significant) or RISC-2 (non-safety related, safety significant) category. The IDP could justify its conclusions in the risk categorization by demonstrating that one of the following conditions are met:

- Relaxing the requirements will have minimal impact on the failure rate increase.
- Showing that adequate data are available to demonstrate that failure modes that prevent the SSC from fulfilling its function are unlikely to occur.
- Such failure modes can be detected in a timely manner.

The choice of appropriate treatment for RISC-3 has been a difficult issue for staff and industry. We believe that much of this difficulty has arisen because the staff recognizes that risk concerns cannot be completely addressed by CDF and LERF and is, therefore, reluctant to relax some special treatment requirements. By explicitly addressing all risk concerns in the categorization process, as discussed above, it may be easier to obtain agreement that components assigned to RISC-3 do not require any treatment beyond "commercial practice."

We note that materials degradation is not directly assessed in NEI 00-04, Revision B. We believe that aging phenomena and the management of degradation must be considered in the IDP deliberations concerning affected SSCs and passive system components.

The use of risk information in regulatory decisionmaking is relatively new. Some within the NRC, the industry, and the public view this evolution with skepticism. The NRC Strategic Plan has established increasing public confidence as a performance goal. The use of rigorous methods to produce risk information is essential to achieving this goal.[1] In many instances, simplified methods can yield satisfactory results. It should be demonstrated, however, that these simplified methods yield results that are consistent with those provided by more rigorous methods and that their limitations are well understood.

In our reports dated October 12, 1999 and February 11, 2000, we commented extensively on the limitations of importance measures. The requirement to use sensitivity studies to determine ΔCDF and ΔLERF provides evidence that NEI 00-04, Revision B, recognizes the major limitation of importance measures, namely, their inability to determine the change in risk associated with a group of components. We

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[1]In his speech to the Regulatory Information Conference on March 5, 2002, Commissioner Diaz stated: "This is the year 2002, almost 30 years after WASH-1400, and it is time that all licensees have a quality Level 2 PRA so they can effectively utilize our regulatory processes."

believe that the IDP would benefit from an explicit identification and discussion of this and other limitations that have been identified in the literature (References 8 and 9).

NEI 00-04, Revision B, shies away from providing guidance or encouragement for licensees to perform uncertainty analyses and relies heavily on sensitivity studies that are substitutes for uncertainty analyses. Modern PRA tools make it relatively routine to perform a genuine uncertainty analysis, i.e., one that propagates the uncertainties in failure rates, and such analysis should be performed where possible.

The argument has been made that using mean values for the failure rates in performing the PRA and the screening is "good enough." We agree that, in the majority of cases, this argument may be true provided that mean values are indeed used, although relatively few investigations are available in the literature (References 8 and 11) to substantiate this claim. We object to the practice of taking arbitrary "point" values of the parameters and declaring them as mean values. Such practices do not contribute to the credibility of the categorization process.

One of the most significant limitations of importance measures is that they measure the impact of individual SSCs on risk, and, consequently, they cannot be used directly to estimate changes in risk for a group of SSCs. This limitation is recognized in NEI 00-04, Revision B, and additional sensitivity studies are suggested to attempt to assess the impact of changing treatment requirements on a group of components. In NEI 00-04, Revision B, it is suggested that the failure rates of RISC-3 SSCs be increased by factors ranging from 2 to 5 to evaluate changes in CDF and LERF. The current justification for this choice of values is weak, and a better justification is needed, especially since these factors are smaller than the factor of 10 used in the South Texas Project multiple exemption request. A distinction between parameter and model uncertainties would be very useful in this case.

We look forward to reviewing the draft final rule language and associated guidance as more progress is made.

Sincerely,

George E. Apostolakis
Chairman

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PRESSURIZED THERMAL SHOCK (PTS) TECHNICAL BASIS RE-EVALUATION

F. Peter Ford

July 10, 2002

PTS Re-evaluation

- **Need For Re-evaluation:**
  - **Less frequent/better Operator performance**
  - **Tougher reactor vessel**
  - **Smaller cracks**
  - **Original criterion overly conservative**

PTS Re-evaluation (Cont'd)

- Integrated Approach
  - Probabilistic Risk Assessment (PRA)
  - Thermal Hydraulics (T-H)
  - Probabilistic Fracture Mechanics (PFM)

PTS Re-evaluation (Cont'd)

- Application of integrated analytical process
  - Oconee Unit 1
  - Beaver Valley Unit 1
  - Palisades
  - Calvert Cliffs Unit 1

PTS Re-evaluation (Cont'd)

- Current process versus 1980's analysis
  - Latest PRA/human reliability data
  - More refined binning
  - Operator action/Acts of commission
  - External events
  - More T-H sequences modeled

PTS Re-evaluation (Cont'd)

- Current versus 1980's analysis (continued)
  - Conservative bias in toughness model removed
  - Spatial variation influence
  - Smaller embedded flaws
  - Non-conservatisms removed

PTS Re-evaluation (Cont'd)

- Observations
  - Primary system LOCAs dominant
  - Realistic operator action
  - Main steamline/steam generator tube rupture no longer dominant
  - Safety relief valve closure time

PTS Re-evaluation (Cont'd)

- Ongoing work
  - Complete internals events analysis
  - External Events
  - Containment Integrity
  - Source Terms

PTS Re-evaluation (Cont'd)

- ACRS Conclusions
  - Extensive/technically sound project
  - Preliminary results of Oconee reactor pressure vessel (RPV) analysis indicate that the current PTS screening criterion may be overly conservative.

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

February 14, 2002

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: REEVALUATION OF THE TECHNICAL BASIS FOR THE PRESSURIZED THERMAL SHOCK RULE

During the 489th meeting of the Advisory Committee on Reactor Safeguards, February 7-8, 2002, we reviewed the methodology and initial results of the Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project. Our Subcommittee on Materials and Metallurgy also reviewed this matter on January 15-16, 2002. During our reviews, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The PTS Reevaluation Project is extensive and appears to be technically sound.

2. The preliminary results of the analysis of the Oconee Unit 1 reactor pressure vessel indicate that when the current PTS screening criterion is reached, the frequency of throughwall cracking of the vessel would be approximately two orders of magnitude below the acceptance criteria for vessel failure given in Regulatory Guide (RG) 1.154. If the ongoing work demonstrates that such results are characteristic of the fleet of pressurized water reactors (PWRs), then the current PTS screening criterion may be overly conservative.

3. When the factors that have large impacts on the failure frequency of the reactor vessel have been identified, they should be scrutinized appropriately.

BACKGROUND

The PTS Rule, 10 CFR 50.61, was established as an adequate protection rule in 1985 in response to a longstanding design-basis issue concerning the integrity of irradiation embrittled PWR pressure vessels during scenarios in which there is a thermal transient in conjunction with the maintenance of system pressure. The rule specifies numerical values of an end-of-life material toughness parameter (RT_{PTS}). Licensees are required to demonstrate that the material

toughness (RT_{NDT}) in their pressure vessels is less than the PTS screening criterion, which depends on the orientation of the crack. The analyses that defined the screening criterion included a number of assumptions that may make the criterion overly conservative. The staff is now reevaluating the degree of conservatism in the technical basis for the screening criterion in the Rule and the associated RG 1.154 acceptance criteria.

Elements of the reevaluation include: (1) a probabilistic risk assessment (PRA) to identify the event sequences that could lead to PTS and then estimate their frequencies; (2) thermal-hydraulic calculations of the pressure, temperature, and heat transfer coefficient in the coolant adjacent to the pressure vessel wall following the various event sequences; and (3) probabilistic fracture mechanics (PFM) estimates of the probabilities of initiating, propagating, and arresting a crack in the pressure vessel for the sets of plant operational and thermal-hydraulic conditions identified in the previous elements. The PFM estimates are calculated using the Fracture Analysis of Vessels - Oak Ridge (FAVOR) code, which is based on earlier Oak Ridge National Laboratory codes; these, in turn, had their foundation in fracture experiments on prototypical pressure vessels started in the 1970s. The current version of the FAVOR code (v01.0) incorporates the probabilistic aspects of the inputs, such as, PRA analysis of operational scenarios and thermal hydraulic, material, and stress conditions, with the output being a calculated distribution of the frequency of throughwall cracking of the vessel. The PTS Reevaluation Project involves the application of this integrated analytical process to four PWRs that reflect a range of designs: Oconee Unit 1, Beaver Valley Unit 1, Palisades, and Calvert Cliffs Unit 1.

In this letter, we comment on the technical progress to date. We do not comment on issues such as external events, containment integrity, and source terms, which are pertinent to potential changes to the throughwall cracking frequency criteria given in RG 1.154 or the PTS screening criterion. These topics will be examined in the future.

DISCUSSION

The PTS Reevaluation Project involves integration of tasks involving PRA, thermal-hydraulics, and PFM including an integrated, quantitative treatment of uncertainty. Overall, the analytical logic and the approach to the physical reality of the technical basis appear to be sound.

The staff has committed to provide us with additional information concerning: how the dynamic events associated with a main steamline break will affect the assumed responses of the operators and the plant; the variance narrowing associated with histogram sampling; and the sensitivity of results to changes in reactor operating power and fuel burnup.

An important aspect of this reevaluation is providing explicit credit for mitigative actions by the operators. The Oconee Unit 1 analysis indicates that some of these actions may have a large impact on the vessel failure frequency. The probabilities of operator failure are evaluated by assessing the relevant performance shaping factors and employing expert judgment. Due to the potential significance of these actions, detailed scrutiny of these probability estimates, including sensitivity studies, alternative human reliability analysis models, and independent peer reviews, should be performed.

There appear to be other factors, such as the spatial and size distribution of flaws, that have a significant impact on the results but have a relatively weak empirical basis. Like the modeling of human error probabilities, these factors should also receive appropriate scrutiny. Prior to completing this Project, it is important to document the validation bases of the relevant codes and databases. We look forward to reviewing further progress.

Sincerely,

George E. Apostolakis
Chairman

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ITEMS OF INTEREST

494th ACRS MEETING

JULY 10-12, 2002

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
494TH MEETING
JULY 10-12, 2002

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Maximizing Nuclear Safety Information: The Contribution of Performance Indicators

Remarks of Richard A. Meserve
Chairman, United States Nuclear Regulatory Commission

before the

NEA/WANO International Forum on Nuclear Regulator/Licensee Interface Issues
Paris, France
19 June 2002

It is my pleasure to join you today to address this session on measuring and communicating safety performance. I will use the occasion primarily to discuss the role of performance indicators in helping to assess the safety of operating reactors and my views on the sharing of such information with the public.

WANO, of course, has been one of the pioneers in the area of performance indicators. Since 1991, when it published its first performance indicator report, and 1993, when reporting began for all reactor designs, the concept has become ever more firmly established as a component of fostering improved nuclear power plant performance. Such indicators are now widely used as the benchmarks.

The U.S. Nuclear Regulatory Commission, as many of you know, introduced a new Reactor Oversight Program or ROP in April 2000, which took the place of the Systematic Assessment of Licensee Performance program, or SALP, with its "Watch List" for problem performers. The SALP was developed when there was relatively little operational experience with nuclear power plants. A governing presumption was that plants were safe if they were in compliance with NRC regulations. As a result, the focus of the SALP process was often on compliance, regardless of the safety implications of a failure to comply. SALP was also criticized for being overly subjective and unpredictable. The process was also largely retrospective, with the result that problems might be cited that had long been corrected, while emergent issues could be overlooked.

In response to these criticisms and in concert with the decision to move toward a more risk-informed regulatory philosophy, the NRC developed the ROP. Since it dramatically differed from the SALP, the ROP was tested at nine nuclear plants before being applied to all licensees. It provides for quarterly evaluation of the performance of every operating station, based on both performance indicators and inspection findings. It monitors plant performance in three broad areas: reactor safety, radiation dose to workers and the public, and security against threats. These broad areas reflect the perspective that reactor safety is maintained by avoiding accidents and reducing their consequences if they occur.

The ROP focuses on seven specific cornerstones: initiating events, mitigation systems, barrier integrity, emergency preparedness, public radiation safety, occupational radiation safety, and security. It is the premise of the ROP that if the licensee performs acceptably in these cornerstones, then reactor safety is maintained.

The ROP uses a number of performance indicators (PIs) to illuminate licensee performance in the areas defined by the cornerstones. For example, the PIs for the initiating event cornerstone include the number of unplanned scrams and losses of the normal reactor cooling system during unplanned scrams. The mitigating systems cornerstone is evaluated in part based on the incidences of safety system failures or unavailability. PIs such as the level of radioactivity in the reactor coolant system and reactor coolant system leakage measure, in part, the integrity of the barrier to the release of radioactivity. Licensees generate data on these performance indicators for submission to the NRC, and the NRC then verifies the data as part of its baseline inspection program. The PIs are compared against established thresholds which are related to their effect on safety.

I discovered at the second review meeting of the Nuclear Safety Convention that there was a belief among some of the participants that the U.S. relies predominantly on performance indicators. This is incorrect. The PIs provide insights into plant performance for selected areas. The NRC's inspection program provides a greater depth and breadth of information for consideration in assessing performance. The inspection program focuses on areas that are not evaluated by PIs or for which PIs cannot tell the whole story. Essentially, the more fully an indicator measures an area, the less extensive is the scope of inspection. It is worth stressing that the PIs are intended to supplement, not to replace the inspection program.

For example, under the initiating event cornerstone, the adequacy of flood protection measures must be evaluated. Due to the rare but possibly risk significant nature of flooding events, no performance indicator was judged to be suitable for monitoring licensee performance in this area. Therefore inspection activities alone verify that compensatory measures are documented, and that equipment is available, is routinely tested, and is fully capable of performing its intended functions.

The availability and reliability of plant equipment is evaluated based on PIs as well as inspection of the implementation of the maintenance rule. The inspection effort ensures that there is a proper balance between availability and reliability when considering the removal of equipment from service for preventive maintenance. The PI is objective, whereas the inspection effort relies on judgment. But both are a measure of the likelihood that accident mitigation systems will perform successfully when needed. The PIs and the inspection efforts are complementary.

The PIs have helped to ameliorate the limitations of the previous inspection program. Most PIs use data that licensees already compile for other performance assessment programs, so their generation is not burdensome. Indicators are thus an efficient means of gathering information about the safety of a plant. PIs are also objective measures of safety and are consistently evaluated across the industry. The PIs themselves and their thresholds for actions based in the PIs are risk-informed, which helps ensure we are spending time on issues of safety significance. However, we do recognize their limitations.

For example, PIs might result in unintended consequences. The NRC and the industry developed and tested replacement PIs for the "unplanned scrams" and related PIs. Some in the industry were concerned that operators would be reluctant to scram a plant manually when necessary in order to avoid an adverse PI, thus degrading plant safety rather than being a measure of it. Although we have seen nothing to indicate that this is occurring, NRC and the industry have sought replacements (thus far unsuccessfully).

Although the PIs and their thresholds were selected using risk insights, their impact on safety is evaluated in isolation. The ROP focuses on the change in core damage frequency that results from changes in a single, isolated parameter, assuming that all other factors that can affect CDF remain constant. Yet based on our understanding of risk, we realize that synergies exist and that a realistic assessment of the change in CDF cannot be related to the change in a single PI. In short, PIs cannot provide a complete picture of risk. By contrast, the risk significance of inspection findings are evaluated using risk assessment models that incorporate the known state of the plant at the time of the fault and therefore the models can provide more insight in cases of multiple faults.

Thus, although the use of PIs and risk assessment tools helps make the ROP more objective, we realize that the assurance of nuclear safety is not reducible to a set of numbers on a grid. Rather, we need a combination of the objective and the subjective. As a result, our evaluation of safety considers both quantitative and qualitative measures. While coolant system leakage can be expressed in figures, maintenance of a proper safety culture cannot.

The ROP views these abstract aspects of licensee performance, including safety conscious work environment, human performance and the effectiveness of problem identification and corrective action programs as "cross-cutting" because they impact more than one cornerstone. Since they are not easily quantifiable, they are evaluated through the inspection program rather than with PIs.

My perspective is the PIs have a role to play in evaluating plant safety. Used properly, they can contribute significantly to our understanding; but they cannot by themselves provide complete understanding. To paraphrase a proverb, "they make good servants but poor masters." Thus, even as we continue to refine and improve our performance indicators, we must remain conscious of their limitations, and not let favorable findings produce complacency either in operators or regulators.

The Davis-Besse plant has recently illustrated the need for caution when assessing safety using both performance indicators and inspection programs. According to the PIs, the Davis-Besse plant was not a cause for concern. But, as you know, the plant turned out to have an undetected safety problem of long standing: corrosion of low alloy steel in the vessel head, resulting from boric acid leakage. I am sure you are familiar with this incident, which is still under review by the NRC.

At this point, it is too early to say whether a more refined set of performance indicators would have made us aware of the corrosion at Davis-Besse sooner, or whether greater attention to the results of the existing indicators would have identified the problem. We will be looking at that issue with great care. Whatever the answer may be, I certainly view it as a lesson to both licensees and regulators of the need to maintain a vigilant and questioning attitude toward plant safety that must not be constrained.

In the meantime, we are continuing our efforts to improve the performance indicators. We are about to begin a pilot program using modified reliability and unavailability indicators for mitigating systems in place of the current PIs, which rely on overly conservative estimates of fault exposure times. We are also developing new PIs for measuring containment integrity and for the elements that comprise the physical security cornerstone.

The NRC also relies on industry-level PIs to confirm that the nuclear industry as a whole is maintaining the safety of operating power plants. The key output of this program is an assessment of statistically significant adverse industry trends in safety performance– a measure of both industry performance and the effectiveness of the NRC's regulatory program. The data show that performance in the U.S. has significantly improved. For example, the average number of automatic scrams has declined by approximately a factor of 3 in the past decade. It is important that the improvement in safety performance is highly correlated with improvement in economic performance: we observe a corresponding increase in the average capacity factor from approximately 65% just 10 years ago to over 90% today. The objective measures of industry performance have helped to increase the public confidence in nuclear power, allowing generating companies to consider power uprates, license extension, and even new construction.

I was asked to comment on openness with regard to performance indicators. Let me first deal with one simple issue -- transparency between the regulator and its licensees. The benefits of performance indicators can only be realized if all parties have a common understanding of how these measures are used and what regulatory actions will result from them. Such an understanding can be reached only if there is an open and frank dialogue, based on trust, and clearly defined responsibilities. Licensees need to be fully forthcoming with regard to data, but regulators have their own duty to make their regulatory requirements clear, understandable, and predictable, as well as consistent in their application. If the situation is working as it should, neither licensees nor regulators should have cause to be surprised by the other.

Openness with the public is a more complex issue. One of the reasons that the ROP has been so widely accepted is the increased access of the public to timely information about plant performance on the NRC web site. The critics of the nuclear industry have joined our licensees in endorsing the ROP as an improvement on the former system largely because of the public availability of detailed and current information.

I must candidly acknowledge that there are some issues associated with public availability. In order to illuminate the data for the benefit of the public and to facilitate the sorting of plants for NRC decision-making, we have established color thresholds to characterize performance. Colors of green, white, yellow and red signify increasing values of risk significance and hence degradation of

safety. Although in reality there may be an insignificant difference between a plant that is at the bottom of the green band and one that is at the top of the white band, the consequences of the binning can be significant, particularly in terms of public perception. This is perhaps the inevitable consequence of the need to provide bands within which to characterize plants. But openness exacerbates the consequences.

Nonetheless, I am strongly committed to the continued public accessibility of such information. We operate on the principle that the public has a right to know how decisions affecting health and safety are made. There is a companion principle: that the more people know about the facts on which those decisions are based, the more confidence they are likely to have in the soundness and integrity of the outcome. Although there may be some who will use the openness to skew the information, benefits of openness outweigh the drawbacks.

Accordingly, I conclude that the answer to the concern about possible public misunderstanding of performance indicators is greater public education, not denial of the information. Indeed, one of the consequences of the September 11 attacks is the growth in public interest in the safety of nuclear plants. The provision of accurate information can help enable this discussion to be illuminated by the facts, rather than by fears. Performance indicators are part of the backdrop against which the discussion of nuclear safety can productively take place.

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THE DYNAMICS OF NUCLEAR POWER TECHNOLOGY AND REGULATION

Remarks of Commissioner Nils J. Diaz
U.S. Nuclear Regulatory Commission

at the

ANS 2002 Annual Meeting

Hollywood, Florida
June 12, 2002

Activities in and of a free society are either regulated or unregulated. Activities in totalitarian societies are all regulated. It is, therefore, fundamental to a free society to not only have both regulated and unregulated activities, but to ensure that regulated activities are beneficial to society, and that they change, as needed by society.

Regulation is a tool of society to frame what society needs, in an orderly, equitable and fair manner. I believe that the role of regulation is to provide a meaningful and useful framework for the protection of rights and the discharge of responsibilities in the areas of health, safety and the environment. Regulation is to be done only for the people, with their best interests as their essential objective; it is to be done for the common good, with full consideration of the national interest.

Regulation does not make you safe; the safe execution of the regulated activity does.

Good regulation provides for the proper exercise of democratic and free market processes to enhance the common good. It is established to provide a framework that allows for the conduct of individual, industrial, commercial, financial, and other activities. Although all regulations restrict, regulation should not deter beneficial activities, but frame them and guide them. Regulation must be dynamic and keep pace with the technology it regulates. Thus, the minimal amount of regulation that achieves what society needs is best for our society.

Science, engineering and technology are mostly "free" until they become part of a regulated activity. I am not going to discuss these interfaces per se, so let me jump directly to nuclear technology and regulation. It is more than obvious that regulation, or overregulation, had a deleterious effect on the development of nuclear technology. It is not so obvious that a static nuclear technology had the same effect on regulation. This impasse, probably could not have been helped prior to the present, eventually, but not without pain, the technology and the regulations wiggled its way to the presently high performance and high level of safety of existing nuclear power plants. But they both stagnated over the most rapid pace of technological improvements in the history of mankind. A bit of history would help to emphasize these points:

- the core of nuclear reactor technology is about 40 years old
- the core of nuclear reactor regulation is about 30 years old

- the technology is defined by a docketed design basis, which lasts the plant lifetime, amended slightly by 50.59 changes and a bit more by license amendments

For example, the key reactor safety criteria and regulations, like 10 CFR 50 Appendices A and B, ECCS criteria, etc. are 30 years old and have stood the test of time.

Surprisingly -or perhaps not surprisingly- the industry performance gains from 1985 to 1996 were achieved without technological or regulatory breakthroughs, but by steady, systematic improvements. The overall performance gains, including improved economics, then enabled the industry to make major commitments for stabilization and prosperity, like license renewal, power uprates, and technological improvements; but all of them still bounded by the traditional design basis and accident criteria, and all they entail. The safety performance then enabled major regulatory improvements, like the revised 50.59, the revised Maintenance Rule, Reg. Guide 1.174, and the Reactor Oversight Process. I might add that there is one proven technological fact whose significance probably has not been well understood or well utilized: leak-before-break, but that is the topic of a future speech.

The S curves of nuclear power plant performance have turned for the better and are now approaching asymptotes. For example, capacity factors are in the 90 percent range (see attached figure), and safety indicators are approaching limits of performance. The SYSTEM has learned. The only way to get out of asymptotic behavior, i.e., to improve performance, is to change either the equations or the constants in the equations. No small fiddling with parameters will affect an asymptotic curve. What this nation needs now is a new system of equations to improve the safety and overall performance of nuclear power, to better serve the people in improving energy independence, the economy and the environment. We are expecting new reactors and we cannot afford to wait another 20 years to have "learned systems." Thus the interaction between the technology and the regulations must advance hand-in-hand, that is, in an in-phase manner.

There are a few lessons in the last 30 years that should not be lost to those seeking to reduce to practice what has been learned. One is very apparent to me: nuclear technology and its regulatory framework must be in-phase, compatible and predictable.

It is obvious that the development and sustainability of nuclear power requires careful attention to political, economical, technological and regulatory factors, so that society can benefit the most. Since the "engineering" of all of these is beyond a regulator's scope of activities, I am going to concentrate on one point: the need to have in-phase, compatible and predictable technology and regulation. Let me up the ante: the need is to achieve and maintain state-of-the-art technology and regulation, with a built-in capability to upgrade both by quantifiable discrete steps, without significant lag by the regulator, so the next improved state-of-the-art technology and regulatory framework levels can be reached effectively and efficiently.

Why is there a need to have a built-in capability to upgrade technology and regulations in discrete steps? Competition over long periods of time coupled with the need for top notch safety performance! One fact has unfolded recently in the US to add to the stage: most existing nuclear power plants in the US are expected to operate for 60 years, an eternity in the on-going technological revolution. And new nuclear power plants might be designed and constructed for even longer periods of time.

There are many other reasons, some quite technical. For example, the Large Break LOCA is no longer useful as the dominant accident sequence, and neither conventional defense-in-depth nor the design basis have allowed for significant technological and regulatory innovation.

Does it make sense to operate in 2002 with technological and regulatory constraints 30 or 40 years old? Of course not; no matter how conservative you are - I am particularly conservative myself. It is not good regulatory policy - nor is it good business - to ignore the new goods or not to discard the not so good old ones.

I say it will make even more sense to think of new deployable nuclear technologies and their regulatory framework in non-rigid design basis terms, but as time-dependent upgradable systems --- both hardware, software and management systems --- that are safer, better, more reliable and more economical for the country and its people. I believe that there is a need for dynamically, strongly coupled technological and regulatory systems, that can stand the test of time because they change with time, and they are developed in-phase, using similar wavelengths. Some might

question the need for independence. I maintain that the independence of a regulator is exercised at decision-making time and suffers not from a proactive regulatory development that is technology-based nor from strong interaction with the industry and other stakeholders.

My friends, that is why I advocate risk-informed and performance-based regulation for nuclear power . There is really no alternative. A risk-informed, and performance-based regime is more quantifiable and more amenable to change as scientific knowledge, engineering, technological and regulatory know-how increase. By defining integral safety envelopes we allow the technology and the regulation to achieve better performance as the systems learn and improve. It is time to think and eventually implement regulatory policies that are as dynamic as the country needs, policies that do not hamper or delay, but serve the people, based on **reasonable** assurance of protection of public health and safety. The key is that **reasonable** is not a stagnant criterion but one that is dynamic and quantifiable. And therein lies the challenge, to solve the coupled technological and regulatory equations simultaneously, while maintaining independent regulatory decision- making conducive to reasonable protection.

We have experienced what happens when regulation is imposed after the fact on a technology being deployed. It was not possible to do it any other way thirty years ago. But it is now possible to jointly develop nuclear technology and its regulatory framework. There is relevant and extremely valuable experience that has been gained from the Advanced Reactors certification program. This program allows for the resolution of substantive technological and regulatory issues during pre-application and application process. It produced better reactors with minimal patchwork requirements. This experience is the right stepping stone for a new way of doing things.

And, I strongly believe that a new way is needed. The current state of regulation may be acceptable for plants operating today; however, a totally new and complete risk-informed and performance-based regulatory regime is needed now to address the possible deployment of new reactors in the USA, if we are to achieve comparable levels of safety and performance, or better, without waiting 20 years to get to that asymptotic portion of the curve. I applaud the Department of Energy initiative to work in partnership with the NRC and industry to develop a requisite and innovative regulatory framework, serving safety and reliability. But it is time to be bold and ask what more can we do for our country ... what more can we do for our country; to allow technological and regulatory innovation to be inserted, as needed, at the beginning, the middle or the end of the process, whether designing, building or operating. The tools exist, they are not perfect but they are quite good.

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HIGH-LEVEL WASTE: CHALLENGES FOR THE NRC

Remarks of Chairman Richard A. Meserve

before the Annual Information and Planning Conference

Office of Inspector General

June 12, 2002

It's a pleasure to address your annual information and planning conference. I am pleased to see that the theme for this year's conference is NRC's Role in Protecting Public Health and Safety in the High-Level Radioactive Waste Arena. You have recognized an important area of NRC activity.

You will be hearing from the staff panel concerning a broad range of issues associated with this topic. I will focus on two areas I believe are the most significant: the review of the potential application to build a high-level waste repository at Yucca Mountain, and the transportation of high-level waste.

I believe that the Agency is approaching one of the most formidable challenges in its history. I am referring to the possibility -- contingent on Congressional action -- that the NRC will receive an application from the Department of Energy for a permit to construct a permanent repository for high-level waste at Yucca Mountain.

It is not an exaggeration to say that no single NRC decision or set of decisions since the response to Three Mile Island is likely to be scrutinized as closely, from a technical, legal, and public confidence standpoint, as those concerning this one-of-a-kind facility at Yucca Mountain. And that is regardless of how the decision comes out.

Let me review briefly where we stand today. As you are probably aware, the Secretary of Energy made a formal recommendation in favor of the Yucca Mountain site; and the President endorsed the DOE Secretary's recommendation. However, the Governor of Nevada has given notice, as the law allows, of the State's disapproval. That has transferred the issue to the Congress. Under the law, the State's disapproval of the Presidential recommendation stands unless, within 90 days of continuous session, Congress votes to disapprove it. The issue has already come to a vote in the House, where the President's determination was supported by an almost three-to-one margin. In the Senate, the Energy and Natural Resources Committee approved the President's recommendation on June 5 by a closer margin. It now goes to the full Senate for a vote which is likely to be close. If Congress approves the President's action, then DOE has indicated that it intends to submit an application to NRC to construct the Yucca Mountain facility in December of 2004. The law then gives NRC up to four years to decide whether to grant the license, including the completion of the administrative proceeding.

Although we have provided comments on the adequacy of site characterization in terms of a possible license application to construct the facility, the Commission has taken no position on the

suitability of the site. Rather, our role, under the Nuclear Waste Policy Act, is to exercise our independent judgment as an expert technical agency and decide the issues on the record developed in the administrative proceeding. This will require schedule discipline, technical excellence, and procedural fairness consistently throughout the review process.

The NRC has for several years been making preparations for the eventuality of an application for a high level waste repository. Last November, we issued Part 63, the regulations setting out the technical requirements a repository must meet in order to be licensed by NRC.[1] These regulations establish performance objectives based on the dose to the reasonably maximally exposed individual, as calculated using reasonable assumptions. Also, as required under EPA's Yucca Mountain standard (40 C.F.R. Part 197), Part 63 also contains requirements for the protection of groundwater. These requirements are somewhat unique to the HLW program in that, in general, our regulations concerning waste disposal are focused on individual protection rather than on the protection of a resource.

I would not for a moment prejudge whether DOE will be able to satisfy us that it has met the demands of the regulations. If it does so, however, I am confident that public health and safety and the environment will be protected, now and in the future.

Although these regulations are risk-informed and performance-based, major challenges exist in demonstrating compliance with the requirements. The system contains both natural and engineered barriers and the system of barriers must function effectively for 10,000 years -- longer than recorded human history. As you can understand, this is unlike any licensing proceeding the agency has faced in the past. To guide the agency's review of a possible license application, the staff recently issued, as a draft for public comment, the Yucca Mountain Review Plan. This Plan is designed to ensure the quality and uniformity of our licensing reviews. I think it is worth describing the plan briefly, to give you a better idea of the scope of the task that will be facing the NRC if an application is submitted.

First there is an acceptance review, which is a preliminary screening of the application to see whether it contains enough information to establish compliance with the regulations. This is not a judgment on the technical adequacy of the application. Rather, it is an evaluation to determine whether the information submitted, if found to be valid, would be sufficient to support granting a license -- in other words, whether the application is ready for the NRC staff to begin its detailed technical review. Even at this stage of the review, there are unique hurdles. Many of you may have heard about the 293 agreements in which DOE committed to provide additional information on technical issues associated with the high level waste repository. At the time of acceptance review, the staff will have to ensure that the information provided under these agreements results in a full and complete license application. All this must be accomplished within 90 days of receiving the application.

The regulations in 10 CFR Part 63 also provide for a preclosure safety analysis. This analysis is designed to ensure that operational exposure limits to workers and the public are not exceeded. This will involve examining the site, the design, potential hazards and their consequences, and the probabilities and uncertainties associated with those hazards. The review will focus on the applicant's ability to demonstrate that the design, construction, and operation of the facility will meet the performance objectives. The staff proposes to allocate resources according the safety significance of the various systems, structures, and components concerned. Many of the staff's review methods currently used in the licensing of spent fuel handling facilities can be applied to the licensing review of repository pre-closure facilities because of the similarity in functions.

Probably the most complex aspects of the review will be the postclosure period of performance, because it involves estimations of repository performance over thousands of years. Our regulations require DOE to conduct a postclosure performance assessment to demonstrate compliance with performance objectives. This means a systematic analysis of the expected performance of the repository as well as consideration of the probability and consequences of external events, such as volcanos and climate changes, that could affect the facility. Moreover, DOE must demonstrate that both engineered and natural barriers contribute to the isolation of the waste -- DOE cannot rely only on the engineered barriers to meet the dose limits. The postclosure performance objectives also require an assessment of how the facility would perform under conditions of human intrusion. A specific scenario involving drilling into the repository has been adopted based on requirements in the EPA standard.

There is more to the review plan, but I think I have described enough to give you a feel for the magnitude of the challenge, and its complexity. Another challenge DOE faces which I have not focused on today, but which is of equal importance to the production of an acceptable license application, is having an adequate Quality Assurance program. Although DOE has signaled its intent to qualify all data, software and models used in a license application fully, quality management continues to be a challenging program area for DOE.

As the previous discussion indicates, we are operating on the assumption that if an application to build a repository at Yucca Mountain is submitted, the administrative proceeding will be massive -- perhaps as vast and complex as any the Federal Government has ever seen. That in and of itself presents a significant challenge: ensuring that all parties and the decisionmakers have timely access to filings and exhibits.

The framework for making the documents available is something we addressed some time ago by creating the Licensing Support Network (LSN). To save time and money that would otherwise be spent duplicating and mailing copies of documents, the LSN serves as a central location where parties and potential parties can submit and obtain the documents they need electronically.

Before turning you over to the rest of your program, I would like to discuss some of the issues concerning the transportation of high-level waste. Press reports on the repository program have noted that some opponents have expressed concerns about the security of the transportation of spent fuel. Federal regulation of spent fuel transportation is shared by the U.S. Department of Transportation (DOT) and the NRC. DOT regulates the transport of all hazardous materials, including spent fuel, and has established regulations for shippers and carriers regarding, among other things, radiological controls, hazard communication, and training. For our part, NRC establishes design standards for the casks used to transport licensed spent fuel, and reviews and certifies cask designs prior to their use. We also conduct an inspection and enforcement program, and review and approve physical security plans for spent fuel shipments.

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

In conclusion, I have tried today to outline some of the issues likely to confront the agency in the event that an application to construct Yucca Mountain is filed. Although the Office of Nuclear Materials Safety and Safeguards (NMSS) will bear most of the burden of this task, should it come to pass, the entire agency, including the Office of Inspector General, is likely to feel the stress. Indeed, given the intense public and political controversy that has surrounded this project, I would expect that OIG will find itself engaged in a variety of projects related to Yucca Mountain in the years ahead if Congressional approval is granted. I appreciate your foresight in preparing yourselves through this conference.

The importance to this country and to this agency of the tasks ahead is beyond dispute. As in the past, we will be counting on OIG to provide informed, thoughtful, and independent assessments of NRC programs. The agency's challenge will be OIG's challenge as well.

Thank you.

[1]Recently the State of Nevada filed a lawsuit challenging our regulations. Nevada v. NRC (D.C. Cir.). We are working with the Department of Justice in defending our regulations.

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NUCLEAR POWER IN THE 2002 NATIONAL ENERGY ARENA

Remarks of Commissioner Nils J. Diaz
U.S. Nuclear Regulatory Commission

at the

The Southeastern Association of Regulatory Utility Commissioners (SEARUC)

Miami Beach, Florida
June 4, 2002

Good morning, ladies and gentlemen. I am pleased to be able to speak today to the Southeastern Association of Regulatory Utility Commissioners (SEARUC) on the topic of nuclear power in the 2002 national energy arena. I would first like to thank, my friend, the President of SEARUC, Braulio L. Baez of the Florida Public Service Commission, for inviting me to speak before your group. I see a number of familiar faces in the audience, but I would particularly like to acknowledge Commissioner Bubba McDonald of the Georgia Public Service Commission and Chairman of the NARUC Subcommittee on Nuclear Issues/Waste Disposal. Commissioner McDonald has appeared before the U.S. Nuclear Regulatory Commission (NRC) on issues of interest to both the NRC and NARUC. The NRC appreciates Commissioner McDonald's leadership, and yu'all contributions to the welfare of our nation.

NRC ACTIVITIES IN RESPONSE TO THE EVENTS OF SEPTEMBER 11

I cannot begin to discuss nuclear power in the 2002 national energy arena without discussing briefly the NRC's response to the shocking events of September 11, 2001. Protection of public health and safety and the common defense and security is NRC's business and they were both attacked on September 11th. For nuclear power plants, when the threat is terrorism and sabotage, security is a subset of safety. Prior to September 11th and even more so today, security is very important; however, it should not overwhelm the safe operation and regulation of nuclear power plants. Security of nuclear power plants must be established in an integral manner with all the safety objectives and all their safety features, internal and external to the plant, and be consistent with the overall requirements of national security. Our national security begins and ends with the principles and practices of our democratic society, and with every component of our society that assures our freedom and the pursuit of happiness. Security does not depend on any one component, but rather on multiple layers of physical structures, systems and components, as well as other protective measures. Achieving a proper balance between them is the present challenge.

By the way, I believe energy security is a key component of national security. The safe and reliable operation of nuclear power plants is vital to our energy security and, therefore, to the well-being of our people. Thus, it is our responsibility to bolster nuclear facilities' defenses. To ensure that adequate levels of protection are in place, the NRC has issued Orders to all 104 commercial nuclear power plants to implement interim compensatory security measures for the current high-level

threat environment. Some of the requirements formalize a series of security measures that NRC licensees had taken in response to advisories issued by the NRC in the aftermath of the September 11th terrorist attacks. Additional security enhancements, which have emerged from the ongoing comprehensive security review, are also spelled out in the Orders. The requirements will remain in effect until such time as the Commission determines that the level of threat has diminished, or that other security changes are needed following the comprehensive re-evaluation of current safeguards and security programs. The Commission views these compensatory measures as prudent, interim measures to address the current high-level threat environment in a consistent manner throughout the nuclear reactor community. The specific actions taken are understandably sensitive, but they generally include requirements for increased patrols, augmented security forces and capabilities, additional security posts, installation of additional physical barriers, vehicle checks at greater standoff distances, enhanced coordination with law enforcement and military authorities, and more restrictive site access controls for all personnel.

The NRC just established an Office of Nuclear Security and Incident Response to consolidate and streamline selected NRC security, safeguards, and incident response responsibilities and resources. Let me conclude my remarks regarding security by stating that we are providing prudent and necessary security measures and that our multiple layers of defense are adequate to protect public health and safety.

INDUSTRY RESTRUCTURING

The NRC has been engaged in a comprehensive effort to address the implications of electric utility rate deregulation for the adequate protection of public health and safety. On September 22, 1998, the NRC published in the *Federal Register* a final rule on financial assurance requirements for decommissioning nuclear power reactors. Among other things, the rule (1) broadens allowable funding assurance mechanisms to include, for example, non-bypassable wires charges to recover decommissioning costs that many States are imposing as part of their deregulation initiatives; (2) requires licensees to report biennially on the status of their decommissioning funds; and (3) allows a 2% credit for decommissioning fund earnings if a Public Utility Commission has not allowed some other rate. In addition, the NRC has continued to monitor rate deregulation developments in the States and uses its regulatory and oversight programs to review and monitor operational experience to ensure that plants continue to operate safely. More recently, the NRC issued a proposed rule on the terms and conditions of decommissioning trust agreements that the NRC believes are necessary to protect these funds. The NRC expects to issue a final rule in this area later this year.

The NRC also has developed or is developing guidance, including Standard Review Plans (SRPs), in several program areas in response to rate deregulation, including: (1) financial qualifications and decommissioning funding assurance; (2) foreign ownership, control, and domination; (3) non-owner operators; and (4) technical qualifications. I have been saying for years, and the data support it, that reliable and economical nuclear power plants are correlated with safe operation. Many plants have increased capacity factors and reduced O&M costs to the point where they appear to be well positioned to compete in any electricity marketplace. In most cases, these plants have demonstrated excellent safety performance, as evidenced by performance indicators and NRC oversight program findings. These plants should continue to be excellent safety performers as deregulation evolves in the United States, as long as their current safety focus is maintained.

LICENSE TRANSFERS AND INDUSTRY CONSOLIDATION

In 1998, the license transfer process was identified as an area for improvement. The goal of this effort was to enhance the predictability, timeliness, and efficiency of the process for transferring power reactor licenses, while maintaining a framework to ensure adequate protection of public health and safety. One of the NRC's significant accomplishments toward this end was to issue a final rule in December 1998 that streamlined the hearing process for license transfers. Among other things, this rule established a more informal, speedier hearing process and incorporated a categorical exclusion to eliminate the need for case-specific Environmental Assessments and No Significant Hazards determinations.

Over the past several years, the NRC has reviewed over 100 license transfer applications. For the first time, the NRC reviewed and approved applications for the sales of entire nuclear units from one owner to another, unrelated owner - the NRC approved the sale of Three Mile Island, Unit 1 (TMI-1), on April 12, 1999, and the sale of the Pilgrim Station on April 29, 1999. Several other

plants have been sold or are in the process of being sold to new owners since then, including Seabrook to Florida Power &Light. Although there has been some reduction in the rate of new license transfer applications as a result of the California experience and the events of September 11th the NRC expects to continue to receive new transfer applications for the foreseeable future.

The NRC focuses its reviews of technical qualifications in license transfer applications on determining whether the proposed transferee has the technical expertise to continue to run the plant safely. For indirect transfers, where the licensee itself remains the same, technical qualifications are generally not an issue in the NRC's review.

In a related issue, and as I stated in my votes on the issue of industry consolidation, the agency needs to have firmly established plans to effectively carry out its mission in this changed environment of consolidation. If we need to either clarify or develop new regulations, we should do so mindful of the many possible restructuring options available to the industry. I will continue to work to ensure that our regulations enhance protection of pubic health and safety, and do not unnecessarily hinder deregulation.

REGULATORY EFFICIENCY

Because power reactor licensees are faced with an increased pressure to reduce or contain costs, often due to rate deregulation, they in turn strongly desire the NRC to act to decrease both direct and indirect costs imposed on them as a result of NRC's actions. The primary example of direct costs is license fees. For the past several years, the U.S. Congress has required the NRC to recover the costs of its regulatory programs through fees levied on its licensees. These fees are calculated both from actual effort expended by the NRC in performing its regulatory duties with respect to specific licensees and from apportionment of overhead and other general costs among all licensees.

Indirect costs include those cost impacts on licensees that arise from the regulatory actions that the NRC takes as part of its mission. The NRC has initiated several actions to reduce unnecessary regulatory burden, including, for example, a major initiative to improve our regulatory system through the application of risk information to risk-informed regulation and another major initiative to improve the NRC's reactor inspection and oversight program.

These initiatives make good regulatory sense by helping us focus on what is truly important to safety. As rate deregulation proceeds, I expect that the NRC will be continually challenged to improve its regulatory efficiency. We will need to continually strive to minimize the direct and indirect cost impacts of our actions, while maintaining our focus on our mission to ensure adequate protection of public health and safety.

LICENSE RENEWAL

Regulatory progress is also evident in the license renewal area. I believe that the NRC has established a license renewal process that can be completed in a reasonable period of time with clear requirements to assure safe plant operation for an additional 20 years of plant life. Plant extensions add predictability to the energy supply pool. To date, eight licenses, including the three Oconee units in South Carolina, Arkansas Nuclear 1, Unit 1 in Arkansas and Hatch 1 & 2 in Georgia, have been renewed. Fifteen license renewals are being processed, including Turkey Point 3 & 4 and St. Lucie
1 & 2 in Florida, the Surry and North Anna plants in Virginia, and McGuire 1 & 2 and Catawba in North Carolina. Over twenty other applications, many for multiple reactors, are expected in the next few years, including plants in Alabama, Arkansas, North Carolina, South Carolina and Tennessee. The NRC is completing license renewal approvals approximately 24 months after receiving the applications. I expect that virtually all of the fleet with a good safety record and maintenance will apply to renew their licenses; efficiencies are being achieved in the timely processing of the applications. In summary, the NRC has established an efficient and effective process to conduct the safety and environmental reviews of license renewal applications.

POWER UPRATES

While license renewal is important for the long-term stability and economics of electricity generation in the United States, power uprates for existing facilities result in a more immediate increase of electricity to meet the needs of our nation today, without compromising safety. NRC regulates the maximum power level at which a commercial nuclear power plant may operate. NRC

uses this power level along with other data in many of the licensing analyses that demonstrate the safety of the plant. This power level is included in the license and technical specifications for the plant. NRC controls any change to a license or technical specification, and the licensee may only change these documents after NRC approves the licensee's application for change.

The NRC has completed over 70 power uprate reviews for approximately 9800 MWt or an equivalent of three nuclear power plant units. The staff estimates that licensees will submit 35 additional power uprate requests in the next five years resulting in about 1590 MWe of added capacity. Upgrades up to 20% increases in full power are under consideration, and one has been granted. We have generally completed these reviews in a manner that does not unnecessarily delay implementation.

NEW NUCLEAR POWER PLANTS

The expectations for new nuclear power plant orders were enhanced last year when the President and the Vice President of the United States presented a National Energy Policy for the United States. The policy is designed to help bring together business, government, local communities and citizens to promote dependable, affordable and environmentally sound energy for the future. In this report, the President supports the expansion of nuclear energy in the United States as a major component of the national energy policy. Notably, the report states that the NRC has made great strides to provide greater regulatory certainty while maintaining high safety standards.

The President's proposal specifically encourages the NRC to ensure that safety and environmental protection are high priorities as we prepare to evaluate and expedite applications for licensing of new advanced-technology nuclear reactors.

Although the events of September 11th may have hurt the economy, there appears to be positive momentum for additional nuclear generating capacity. TVA recently approved a plan to restart the Browns Ferry Plant, Unit 1, which is still licensed to operate. Also, the Watts Barr Nuclear Plant, Unit 2, and the Bellefonte Units remain in a deferred status with construction permits. Several potential applicants for early site permits have been identified, including Exelon Generation, Dominion Generation, and Entergy Operations, with applications expected in the near term. Westinghouse has applied for design certification of the AP1000, General Atomics has submitted a pre-application licensing plan for the Gas Turbine-Modular Helium Reactor, and General Electric has requested a pre-application review of the 4000 MWt European Simplified Boiling Water Reactor. Although there is no certainty that new plant construction is in our near future, the NRC is preparing to carry out our responsibilities in this area.

DISPOSAL OF SPENT FUEL AND HIGH LEVEL RADIOACTIVE WASTE

The Commission believes that a permanent geologic repository can provide the appropriate means for the United States to manage spent nuclear fuel and other high-level radioactive waste in a safe manner. We also believe that public health and safety, the environment, and the common defense and security can be protected by deep underground disposal of these wastes. The Commission takes no position on whether such a repository should be located at Yucca Mountain, Nevada. Our views on that question must be shaped by the results of the Congressionally mandated licensing process.

Based on our technical reviews and pre-licensing interactions, we believe that sufficient information can be available at the time of a license application. The U.S. Department of Energy (DOE) and NRC have reached and documented numerous agreements regarding additional information that will be needed for a licensing review. Approximately two thirds of these agreements call for DOE to document the bases for assumptions or conclusions. The remainder oblige DOE to perform specific tests or analyses, to document prior tests or studies, or to provide other information. As DOE completes the actions necessary to fulfill these agreements, NRC will review the results promptly and notify DOE of our findings. Based on these agreements, it appears that DOE can assemble the information necessary for an application that NRC can accept for review.

One issue that will certainly play an important role on the resolution of the high-level radioactive waste disposition is transportation. Federal regulation of spent fuel transportation safety is shared by the U.S. Department of Transportation (DOT) and the NRC. DOT regulates the transport of all hazardous materials, including spent fuel, and has established regulations for shippers and carriers regarding radiological controls, hazard communication, training, and other aspects. For its part,

NRC establishes design standards for the casks used to transport licensed spent fuel, and reviews and certifies cask designs prior to their use. Further, cask design, fabrication, use and maintenance activities must be conducted under an NRC-approved Quality Assurance program.

The NRC believes the safety protection provided by the current transportation regulatory system is well established. Nonetheless, we continually examine the transportation safety program, and the events of September 11, 2001, have added to our concerns. In FY 2000, NRC reevaluated its generic assessment of spent fuel transportation risks to account for the fuel, cask and shipment characteristics likely to be encountered in future repository shipping campaigns. As a part of its evaluation, the NRC staff is analyzing appropriate national transportation accidents, such as the 2001 train accident in Baltimore, Maryland, to determine if lessons learned from such events should be included in our transportation requirements or analyses. Finally, NRC is sponsoring a study to update its evaluation of cask response to acts of sabotage. These studies should be available by the time a license application for an underground repository is received.

The Commission believes that deep geologic disposal is appropriate for high-level radioactive wastes and spent nuclear fuel. Our role is to put in place a licensing system that will ensure adequate protection of public health and safety and the environment and to efficiently review and evaluate any license application submitted to ensure its compliance with regulatory requirements. And, as I hope you can glean from the actions I described above, there are many challenges facing us; and many opportunities. Most of these could impact your duties and responsibilities. I'm sure my fellow panelists can expand on these thoughts.

EA-02-014 - Millstone (Dominion Nuclear Connecticut, Inc.)

June 25, 2002

EA-02-014

Mr. J. Alan Price, Vice President
Nuclear Technical Services-Millstone
c/o Mr. D. A. Smith, Manager-
Regulatory Affairs
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road, Waterford, CT 06385

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY - $288,000 (NRC Special Inspection Report No. 50-245/01-013)

Dear Mr. Price:

This letter refers to the NRC special team inspection conducted at the Millstone Power Station, Unit 1, between October 9 - 18, 2001, and continued in the NRC Region I office until December 21, 2001, to review the results of your investigation of the loss of two irradiated fuel rods at the facility. The findings of the on-site portion of the inspection were presented to you on October 18, 2001. Subsequently, all of the findings were presented to you in an exit meeting open to public observation on January 15, 2002. The inspection report, which was sent to you on February 27, 2002, described two apparent violations identified during the inspection. The violations involved the failure to adequately account for special nuclear material (SNM) contained within two irradiated fuel rods, and report to the NRC, in a timely manner, the missing licensed material.

The NRC letter that transmitted the inspection report provided you an opportunity to either request a predecisional enforcement conference to discuss the apparent violations or explain your position in a written response. In a telephone conversation on February 28, 2002, Mr. David Smith of your staff informed Dr. Ronald Bellamy, NRC Region I, that Dominion Nuclear Connecticut, Inc. (DNC) declined a predecisional enforcement conference, but intended to provide a written response to address the violations. Your response was provided in a letter dated March 28, 2002, wherein you did not contest the apparent violations, which occurred while that plant was being operated by Northeast Nuclear Energy Company (NNECO). Based on the information developed during the inspection, and the information that you provided in your March 28 response, the NRC has determined that violations of NRC requirements occurred, as described above. These violations are cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding them are described herein, as well as in detail in the subject inspection report.

NNECO initially informed the NRC of the circumstances surrounding the two lost fuel rods on November 16, 2000. At the time, NNECO indicated that engineers, while performing a records reconciliation and verification of the Millstone 1 spent fuel pool inventory in June 2000, identified that the two fuel rods were not in the locations reflected in the SNM records. Although the engineers, at first, considered the discrepancy to be a record keeping problem, a subsequent search of the Unit 1 spent fuel pool did not locate the rods. As a result, the NRC was informed of this finding, and an extensive investigation was initiated to determine the possible location of the two rods. Based on that investigation, NNECO concluded that most likely in the Fall of 1979, the two rods were cut up during spent fuel pool processing activities because they were likely

mistaken for local power range monitors (reactor hardware) that were similar in size and shape. The cut-up rods were then likely sent to a low level radioactive waste facility along with irradiated reactor hardware sometime between March 1985 to December 1992.

The NRC agrees with the broad conclusions in your March 28 response letter regarding the location of the missing fuel. Specifically, as noted in our inspection report: (1) there is no evidence to support the possibility of theft or diversion of the missing fuel rods, and (2) the missing fuel rods are most likely located in a licensed low level radioactive waste facility, and because of the radiological controls in place at these facilities, realistically, there is no current threat to public health. The NRC concluded that it is highly unlikely the fuel rods, in their entirety, remain in the Millstone 1 spent fuel pool. The NRC also concluded that presently, there are adequate controls to account for all SNM at the Millstone Station, except for the missing fuel rods.

Notwithstanding the fact that there was no realistic threat, past or present to the public health, the loss of highly radioactive fuel rods is unprecedented and is a very significant violation. Therefore, this violation, which is described in Section I of the enclosed Notice, is categorized in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600 at Severity Level II. Although the accountability of the two fuel rods was likely lost in 1979 or 1980, and the cut-up rods were likely sent offsite sometime between March 1985 to December 1992, the violation, which involved a failure to: (1) keep adequate records, (2) establish adequate procedures for control and accounting of SNM, and (3) conduct adequate physical inventories of SNM, continued until November 16, 2000.

In accordance with the Enforcement Policy, a base civil penalty in the amount of $96,000 is considered for a Severity Level II violation. Because the violation has been classified at Severity Level II, the NRC considered whether credit was warranted for *Identification and Corrective Action* in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. With respect to *Identification,* you had prior opportunities to identify the location of the fuel rods and institute corrective actions. The first opportunity occurred in 1979 when inventory cards were generated for the fuel rods. However, these records were not maintained as required by your procedures and were not brought into your inventory process. A second opportunity occurred in 1980 when the spent fuel pool map was revised, which inadvertently removed the location of the fuel rods. As a result of both missed opportunities to adequately account for the fuel rods, your annual physical inventories did not identify the fuel rods. It was only when you did a comprehensive inventory in 2000, including a physical verification of SNM contents versus SNM data for all records, that you identified the two fuel rods were missing. Therefore, credit for identification is not warranted.

The NRC determined that credit for *Corrective Action* is warranted. Your investigation of the missing fuel rods was thorough and complete, the root cause analysis was comprehensive, and your physical inspection process of the Millstone 1 spent fuel pool was thorough and comprehensive, as noted in our February 27 letter that transmitted the inspection report. In addition, your response dated March 28, 2002, indicated that with the exception of the two missing fuel rods, you have accounted for all fuel at Millstone Unit 1. You also summarized the various corrective actions that have been taken, which include, but are not limited to: (1) appointing a dedicated manager responsible for onsite physical fuel management activities, including the spent fuel pool, (2) enhancing procedures to strengthen the SNM and accountability program, (3) upgrading procedures to require detailed waste characterization and verification of irradiated components being placed in disposal containers, (4) emphasizing your performance and accountability expectations to contractors working at the facility, and (5) accomplishing continuous process improvement within the corrective action program by maintaining a management-directed low threshold for condition report initiation.

Based on the above, application of the normal civil penalty assessment process would result in a $96,000 civil penalty. However, notwithstanding the normal assessment process, the NRC can exercise discretion to escalate the base civil penalty given the circumstances of the particular violation. The NRC has decided to triple the base amount of the civil penalty given the unprecedented nature of the loss of highly radioactive fuel from a nuclear power reactor and to further emphasize the importance of adequate accounting of irradiated fuel at nuclear power reactors. Accordingly, I have been authorized, after consultation with the Director, Office of Enforcement, and the Deputy Executive Director for Reactor Operations, to issue the enclosed Notice of Violation and Proposed Imposition of Civil Penalty (Notice) in the amount of $288,000. The NRC recognizes the fuel rods were likely disposed of a number of years ago, and in this case,

there was no realistic impact, past or present on public health. Had it not been for this and your thorough and comprehensive corrective actions, the civil penalty would have been higher.

The second violation, involving the failure to report the missing fuel rods to the NRC in a timely manner, is also described in the enclosed Notice and is classified at Severity Level IV. In our October 31, 2001 letter, we explained that the NRC Office of Investigations (OI) conducted an investigation into whether there was any deliberate effort to delay reporting this information to the NRC, and OI did not identify any willfulness associated with the late report. However, the NRC did not issue a non-cited violation (NCV) in this case because you did not meet the NCV criteria specified in Section VI.A.1 of the Enforcement Policy since corrective actions to address recurrence were not specified. Although you wrote a condition report (CR 02-02376) on March 4, 2002, to broadly incorporate the OI report into your corrective action program, the CR only discussed processes currently in place to prevent reporting violations and stated that no further corrective actions were required at this time. Further, no specific corrective actions were discussed in your response dated March 28, 2002.

While the events described herein occurred when the plant was being operated by NNECO, DNC is the current licensee of Millstone and, as acknowledged in your March 28 response letter, you recognize your responsibility and accountability for safe operation of the facility and as custodian of its nuclear fuel. You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

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

Sincerely,

/RA/ Hubert J. Miller

Hubert J. Miller
Regional Administrator

Docket No. 50-245
License No. DPR-21

Enclosure: Notice of Violation and Proposed Imposition of Civil Penalty

cc w/encl:
D. A. Christian, Senior Vice President - Nuclear Operations and Chief Nuclear Officer
W. R. Matthews, Vice President and Senior Nuclear Executive - Millstone
J. A. Price, Site Vice President - Millstone
S. E. Scace, Director, Nuclear Engineering
G. D. Hicks, Director, Nuclear Station Safety and Licensing
C. J. Schwarz, Director, Nuclear Station Operations and Maintenance
P. J. Parulis, Manager, Nuclear Oversight
D. A. Smith, Manager, Licensing
L. M. Cuoco, Senior Nuclear Counsel
N. Burton, Esquire
V. Juliano, Waterford Library
S. Comley, We The People
J. Buckingham, Department of Public Utility Control
E. Wilds, Director, State of Connecticut SLO Designee
First Selectmen, Town of Waterford
D. Katz, Citizens Awareness Network (CAN)
R. Bassilakis, CAN
J. M. Block, Attorney, CAN

J. Besade, Fish Unlimited
G. Winslow, Citizens Regulatory Commission (CRC)
J. Markowicz, Co-Chair, NEAC
E. Woollacott, Co-Chair, NEAC
R. Shadis, New England Coalition Staff

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION AND CIVIL PENALTY

Dominion Nuclear Connecticut, Inc.
Millstone Power Station, Unit 1

Docket No. 50-245
License No. DPR-21
EA No. 02-014

During an NRC inspection conducted between October 9, 2001 and December 21, 2001, for which a public exit meeting was held on January 15, 2002, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the NRC proposes to impose a civil penalty pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282 and 10 CFR 2.205. The particular violations and associated civil penalty are set forth below:

I VIOLATION ASSESSED A CIVIL PENALTY

10 CFR 70.51(b), (c) and (d) require, in part, that each licensee shall keep records showing the receipt, inventory (including location), disposal, acquisition, and transfer of all special nuclear material (SNM) in its possession regardless of origin or method of acquisition; each licensee who is authorized to possess at any one time SNM in a quantity exceeding one effective kilogram of SNM shall establish, maintain, and follow written material control and accounting procedures that are sufficient to enable the licensee to account for all SNM in his possession under license; and each licensee who is authorized to possess at any one time and location SNM in a quantity totaling more than 350 grams of contained uranium-235, uranium-233, or plutonium, or any combination thereof, shall conduct a physical inventory of all SNM in its possession under license at intervals not to exceed 12 months.

Contrary to the above, the licensee, who was authorized to possess SNM in excess of the quantities stated above, failed to:

1. keep adequate records showing the inventory (including location), disposal and transfer of the SNM in irradiated fuel rods BKO136 and BPO406, from the Fall of 1979 until November 16, 2000. Specifically, inventory records were in error (after the two fuel rods in the spent fuel pool most likely were mistakenly cut up, in the Fall of 1979), in that the records were not revised to reflect the change in the spent fuel pool. It was not until November 16, 2000, that the licensee documented in Condition Report M1-00-0548 that the location of the two fuel rods was not properly reflected in SNM records and their location could not be determined.

2. establish, maintain, and follow adequate written material control and accounting procedures sufficient to account for all SNM in his possession. Specifically, Operating Procedure 1001, "Fuel Inventory and Control," January 1972, and Reactor Engineering Procedure 1001, "SNM Inventory and Control," required preparation of a Material Transfer Form for any movement of SNM, and required maintaining database card file records with current location information. In several instances between May 4, 1974 to March 13, 1979, and several times after the Fall of 1979, personnel failed to follow these procedures when SNM was moved without preparing a Material Transfer Form. In addition, although inventory cards were generated in May 1979 describing the two fuel rods and their location, the cards were not maintained after that time even though the rods were moved several times. These records were in error until November 16, 2000, when the licensee generated Condition Report M1-00-0548; and

3. conduct adequate physical inventories of all SNM in its possession under license. Specifically, due to errors in SNM records, physical inventories conducted since 1980 did not identify that the two fuel rods were missing, until September 12, 2000, when the licensee determined that the two fuel rods were not in the locations as specified in the SNM records.

This violation has been categorized at Severity Level II (Supplement VI).
Civil Penalty - $288,000.

II VIOLATION NOT ASSESSED A CIVIL PENALTY

10 CFR 20.2201(a)(1)(ii) requires the licensee to report by telephone within 30 days after the occurrence of any lost, stolen, or missing licensed material becomes known to the licensee, all licensed material in a quantity greater than 10 times the quantity specified in Appendix C to Part 20.

Contrary to the above, the licensee failed to notify the NRC by telephone within 30 days of the occurrence of missing licensed material in a quantity greater than 10 times the quantity specified in Appendix C to Part 20. Specifically, on September 12, 2000, following an unsuccessful search of the locations specified in the records (fuel assembly and certain areas of the spent fuel pool) for two spent fuel rods containing licensed material in a quantity greater than 10 times the quantity specified in Appendix C to Part 20, the licensee had sufficient information at the management level to conclude that two fuel rods were missing. Nevertheless, the NRC did not receive any notification that the fuel rods were missing until informally notified via telephone on November 16, 2000, followed by a formal notification via telephone to the NRC Operations Center on December 14, 2000. This notification was in excess of the 30 day notification requirement.

This violation has been categorized at Severity Level IV (Supplement VI).

Pursuant to the provisions of 10 CFR 2.201, DNC (Licensee) is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the date of this Notice of Violation and Proposed Imposition of Civil Penalty (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each alleged violation: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, and if denied, the reasons why, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps that will be taken to avoid further violations, and (5) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an Order or a Demand for Information may be issued as why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown.

Within the same time as provided for the response required above under 10 CFR 2.201, the Licensee may pay the civil penalty proposed above in accordance with NUREG/BR-0254 and by submitting to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, a statement indicating when and by what method payment was made, or may protest imposition of the civil penalty in whole or in part, by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the Licensee fail to answer within 30 days of the date of this Notice of Violation and Proposed Imposition of Civil Penalty, an order imposing the civil penalty will be issued. Should the Licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violations listed in this Notice, in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty in whole or in part, such answer may request remission or mitigation of the penalty.

In requesting mitigation of the proposed penalty, the factors addressed in Section VI.C.2 of the Enforcement Policy should be addressed. Any written answer in accordance with 10 CFR 2.205

should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the Licensee is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing a civil penalty.

Upon failure to pay any civil penalty due which subsequently has been determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282c.

The response noted above (Reply to Notice of Violation, statement as to payment of civil penalty, and Answer to a Notice of Violation) should be addressed to: Mr. Frank Congel, Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region I.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

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

Dated this 25th day of June 2002

EA-02-041 - Beaver Valley (FirstEnergy Nuclear Operating Company)

June 24, 2002

EA-02-041

Mr. L. W. Myers
Senior Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, Pennsylvania 15077

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION AT THE BEAVER VALLEY POWER STATION (NRC Inspection Report 50-334/02-03, 50-412/02-03)

Dear Mr. Myers:

The purpose of this letter is to provide you with the final results of our significance determination of the preliminary Yellow finding identified in the subject inspection report that was discussed during an inspection exit meeting via telephone on March 15, 2002, with yourself, and other members of your staff. The inspection finding was assessed using the Emergency Preparedness Significance Determination Process (EP SDP), and was preliminarily characterized as Yellow, a finding with substantial importance to safety that will result in additional NRC inspection and potentially other NRC action.

As noted in the NRC letter dated April 12, 2002, forwarding the inspection report, this preliminary Yellow finding involved personal home alerting devices (PHADs), which comprise a small portion of the public alert and notification system (ANS) for the Beaver Valley Power Station (BVPS). The remainder of the ANS is comprised of pole-mounted sirens. Specifically, the PHADs were not being adequately tested and maintained to fulfill their design function of alerting members of the public in the BVPS Emergency Planning Zone (EPZ) who may not hear the pole-mounted sirens. As a result, you apparently did not meet a risk significant planning standard (RSPS) set forth in 10 CFR 50.47(b)(5), which requires that means have been established to provide early notification to the populace within the plume exposure EPZ.

At your request, a Regulatory Conference was held on May 15, 2002, at the Region I Office in King of Prussia, PA, with you and members of your staff, to further discuss your views on this issue. A copy of the handout you provided at the conference has been entered in the NRC's document system (ADAMS) and is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html under accession number ML021500424. During the conference, you did not agree with the NRC's preliminary Yellow assessment of this issue, and you described your assessment of the significance of the finding. You contended that essentially 100% of the ANS was functional and the impact on public health and safety was very low. Specifically, you indicated, based on subsequent siren tests in areas near PHAD locations, that the sirens would in fact provide the means for early notification in 75 percent of the areas where PHADs were located. As a result, you contended that the test results demonstrated that less than 1 percent of the population was potentially affected by the PHAD deficiencies, rather than the approximate 3 percent of the population served by the PHADs.

After considering the information developed during the inspection and the information you provided at the conference, the NRC has concluded that the inspection finding is more appropriately characterized as White, an issue with low to moderate safety significance, which may require additional NRC inspections. The EP SDP recognizes that a finding placed in context through the SDP can result in a color that exceeds the actual impact on public health and safety. You did not meet the requirements of the risk significant planning standard (RSPS) set forth in 10 CFR 50.47(b)(5) because a majority of the PHADs were degraded or removed, which constituted a degradation of the means for early notification of the public. However, in the event of a radiological emergency, the majority of the public would be notified directly by the ANS, which includes the sirens and the functional PHADs. The populace not covered by the sirens and functional PHADs would likely be informed via "informal alerting" by such means as television, radio, or "word-of-mouth." We concluded that the condition of the ANS due to degraded and removed PHADs did not have a substantial impact on the EP Cornerstone Performance Expectation, and therefore, the finding does not rise to the level of substantial safety significance (Yellow) and is more appropriately characterized as low to moderate safety significance (White).

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

The NRC has also determined that a violation of 10 CFR 50.47(b)(5) occurred, as cited in the attached Notice of Violation (Notice). The circumstances surrounding the violation are also described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

During the conference, your staff discussed your root cause evaluation and corrective actions to address the violation. Specifically, your evaluation found that a formal testing and maintenance program was not implemented to ensure satisfactory performance of the PHADs. As a part of your corrective actions, you: (1) implemented interim measures by September 1, 2001, to assure that the affected population was notified in the event of an emergency at the BVPS; (2) trained your Emergency Preparedness Section on the design basis document; (3) installed and tested additional sirens to eliminate the need for PHADs; and (4) submitted a revised design report that removed the PHADs from the BVPS ANS. In addition, you stated that you will perform a self-assessment of this area in September 2002.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket as summarized herein, in NRC Inspection Report 50-334/02-03, 50-412/02-03 dated April 12, 2002, and in your slides (ADAMS accession number ML021500424) used during the Regulatory Conference held in the Region I office on May 15, 2002. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

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

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

Sincerely,

/RA/ James T. Wiggins Acting For

Hubert J. Miller
Regional Administrator

Docket Nos. 50-334, 50-412
License Nos. DPR-66, NPF-73

Enclosure: Notice of Violation

cc w/encl :
L. W. Pearce, Plant General Manager
R. Fast, Director, Plant Maintenance
F. von Ahn, Director, Plant Engineering
R. Donnellon, Director, Maintenance
M. Pearson, Director, Services and Projects
J. Lash, Personnel Development
L. Freeland, Manager, Nuclear Regulatory Affairs & Corrective Actions
M. Clancy, Mayor, Shippingport, PA
Commonwealth of Pennsylvania
State of Ohio
State of West Virginia
P. Cote, Acting Regional Director, FEMA Region III

NOTICE OF VIOLATION

First Energy	Docket No. 50-334; 50-412
Beaver Valley Units 1 and 2	License No. DPR-66; NPF-73
	EA No. 02-041

During an inspection conducted between September 1, 2001, and March 15, 2002, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR 50.54(q) specifies that a licensee authorized to possess and operate a nuclear power reactor shall follow and maintain in effect emergency plans which meet the standards in 10 CFR 50.47(b).
>
> 10 CFR 50.47(b)(5) requires, in part, that the licensee establish a means to provide early notification to the populace within the plume exposure pathway Emergency Planning Zone (EPZ).
>
> The Beaver Valley Emergency Plan, Appendix F, Revision 12, states, in part, that the siren notification system consists of two types of sirens: (1) Large, pole-mounted sirens; and (2) Personal Home Alerting Devices (PHADs).
>
> Contrary to the above, prior to September 1, 2001, the licensee could not provide early notification to the entire populace within the plume exposure pathway EPZ because a majority of the PHADs were degraded or removed.

This violation is associated with a WHITE significance determination process finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in NRC Inspection Report 50-334/02-03, 50-412/02-03 dated April 12, 2002, and in your slides (ADAMS accession number ML021500424) used during the Regulatory Conference held in the Region I office on May 15, 2002. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

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

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

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

Dated this 24th day of June, 2002

EA-02-107 - Columbia Generating Station (Energy Northwest)

June 24, 2002

EA-02-107

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968; MD 1023
Richland, Washington 99352-0968

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC Inspection Report No. 50-397/02-05)

Dear Mr. Parrish:

The purpose of this letter is to provide you with the final results of our significance determination for the preliminary White finding identified in the subject inspection report. The inspection finding was assessed using the significance determination process and was preliminarily characterized as White, a finding with low to moderate increased importance to safety, which may require additional NRC inspections. This White finding involved degradation of 16 safety-related and 6 important-to-safety breakers that were replaced during the refueling outage ending in June 2001.

In a telephone conversation with Mr. William B. Jones of my staff on June 3, 2002, Ms. Christina Perino of your staff indicated that Energy Northwest did not contest the characterization of the risk significance of this finding and that you declined your opportunity to discuss this issue in a regulatory conference.

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

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

The NRC has also determined that the failure to verify the design adequacy of the breakers properly and the failure to identify promptly and correct the breaker malfunction when it initially occurred on June 29, 2001, and November 19, 2001, violated the requirements of 10 CFR Part 50, Appendix B, Criterion III and Criterion XVI, as cited in the enclosed Notice of Violation. The circumstances surrounding the violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice of Violation when preparing your response. Because plant performance for this issue has been determined to be in the regulatory response band, we will use the NRC Action Matrix to determine the most appropriate NRC response for this event. We will notify you by separate correspondence of that determination.

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

Sincerely,

/RA/

Ellis W. Merschoff
Regional Administrator

Docket: 50-397
License: NPF-21

Enclosure: Notice of Violation

cc w/Enclosure:

Chair
Energy Facility Site Evaluation Council
P.O. Box 43172
Olympia, Washington 98504-3172

Rodney L. Webring (Mail Drop PE08)
Vice President, Operations Support/PIO
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Greg O. Smith (Mail Drop 927M)
Vice President, Generation
Energy Northwest
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Lynn Albin
Washington State Department of Health
P.O. Box 47827
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NOTICE OF VIOLATION

Energy Northwest
Columbia Generating Station

Docket No. 50-397
License No. NPF-21
EA-02-107

During an NRC inspection which concluded on May 2, 2002, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR Part 50, Appendix B, Criterion III, Design Control, states, in part, that measures shall be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of structures, systems and components to which Appendix B applies.
>
> 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Actions, states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. For significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude recurrence.
>
> Contrary to the above, in June 2001, the licensee completed Design Modification 99-0140-0, "Breaker Replacement," and the design control measures established by the licensee were not adequate to assure the suitability of the replacement breakers. Specifically, the licensee failed to incorporate vendor information regarding maintenance of mechanism-operated cell (MOC) switches in these breakers, resulting in breaker failures that affected the safety-related functions of plant systems. For example, on June 29, 2001, the Division II standby service water MOC switch failed to reposition during breaker closure, rendering the standby water service water train inoperable. In addition, despite failures of this type occurring on June 29, 2001, and November 19, 2001, the licensee failed to identify the cause of the condition and take corrective actions to preclude recurrence of this significant condition adverse to quality. Consequently, on February 13, 2002, a similar failure occurred involving the MOC switch associated with the Division II emergency diesel generator.
>
> This violation is associated with a White SDP finding.

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

should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

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

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 24th of June 2002

EA-02-090 - Point Beach (Nuclear Management Company, Inc.)

June 13, 2002

EA-02-090

Mr. M. Warner
Site Vice President
Kewaunee and Point Beach Nuclear Plants
Nuclear Management Company, Inc.
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT NOS. 50-266/02-03; 50-301/02-03 AND 50-266/02-05; 50-301/02-05) (POINT BEACH NUCLEAR PLANT, UNIT 2)

Dear Mr. Warner:

The purpose of this letter is to provide you with the final results of our significance determination of the finding identified in the subject inspection reports. As discussed in the letter from the NRC to you, dated May 14, 2002, the inspection finding was assessed using the significance determination process and was preliminarily characterized as White, an issue with low to moderate increased importance to safety, which may require additional NRC inspections. This White finding involved the self-revealing failure of safety injection system pump 2P-15B due to nitrogen gas binding.

In a telephone conversation with Mr. Roger Lanksbury of NRC, Region III, on May 24, 2002, you indicated that Nuclear Management Company agreed with the preliminary characterization of the risk significance of this finding and the apparent violation associated with this issue. You also indicated that there was no additional information you wished to present and, therefore, a Regulatory Conference was not needed.

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

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

The NRC has also determined the failure to take prompt corrective actions to preclude repetition after Point Beach personnel concluded that the safety injection system was susceptible to gas binding and when decreasing trends in the Unit 2 A safety injection accumulator level were identified is a violation of Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B, which requires, in part, that conditions adverse to quality be promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition. The violation is cited in the attached Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in the subject inspection reports. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement because it is

associated with a White finding.

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

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

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

Sincerely,

/RA/

Geoffrey E. Grant, Director
Division of Reactor Projects

Docket Nos. 50-266; 50-301
License Nos. DPR-24; DPR-27

Enclosure: Notice of Violation

cc w/encl:
R. Grigg, President and Chief Operating Officer, WEPCo
R. Anderson, Executive Vice President and Chief Nuclear Officer
T. Webb, Licensing Manager
D. Weaver, Nuclear Asset Manager
T. Taylor, Plant Manager
A. Cayia, Site Director
J. O'Neill, Jr., Shaw, Pittman, Potts & Trowbridge
K. Duveneck, Town Chairman, Town of Two Creeks
D. Graham, Director, Bureau of Field Operations
A. Bie, Chairperson, Wisconsin Public Service Commission
S. Jenkins, Electric Division, Wisconsin Public Service Commission
State Liaison Officer

NOTICE OF VIOLATION

Nuclear Management Company, Inc.
Point Beach Nuclear Plant, Unit 2

Docket No. 50-301
License No. DPR-27
EA-02-090

During NRC inspections conducted from January 22 through March 31, 2002, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B, requires, in part, that conditions adverse to quality be promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition.

Contrary to the above, between April 2000 and February 20, 2002, the licensee failed to promptly identify and correct a significant condition adverse to quality regarding leakage from the 2T-34A safety injection accumulator. Nitrogen that leaked from the accumulator caused gas binding and subsequent failure of the 2P-15B safety injection pump on February 20, 2002. Specifically:

a. In April 2000, the licensee completed a review of Information Notice 88-023, Supplement 5, "Potential for Gas Binding of High-Pressure Safety Injection Pumps During a Loss-of-Coolant-Accident," and identified that the Point Beach safety injection systems were susceptible to gas binding in the event of leakage from the safety injection accumulators through multiple check valves and/or motor-operated valves. However, corrective actions were not promptly taken.

b. On February 12, 2001, (Condition Report 01-0454), and on January 15, 2002, (Action Request 1862), licensed control room operators identified decreasing trends in 2T-34A safety injection accumulator level. However, the cause of the condition was not determined and corrective actions were not taken to preclude repetition.

This violation is associated with a White SDP finding.

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

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

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

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

Dated this 13th day of June 2002.

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

April 16, 2002

EA-00-022
EA-01-310

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

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

Dear Mr. Scarola:

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

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

After the regulatory conference, the NRC revised its risk assessment after considering the information discussed at the conference, and an additional review of the factors and assumptions used in the NRC's initial determination of the increase in risk. The NRC's letter of March 18, 2002, forwarded the NRC's revised risk estimate and the basis for the factors that support the estimate, and offered CP&L the opportunity to provide its perspectives on the updated information. CP&L subsequently informed the NRC by telephone that the risk information presented at the regulatory

conference was sufficient to characterize the risk properly, and that it did not intend to provide any additional technical information on the matter at this time.

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

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

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

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

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

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

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

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence is already adequately addressed on the docket in this letter and in the information presented by CP&L at the regulatory and predecisional enforcement conferences. Therefore, you are not required to respond to the

provisions of 10 CFR 2.201 unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

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

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

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

Sincerely,

/RA/

Luis A. Reyes
Regional Administrator

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

Enclosure: Notice of Violation

cc w/encls:

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

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

NOTICE OF VIOLATION

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

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

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

> 10 CFR 50.48 requires that all operating nuclear power plants have a fire protection program that satisfies Criterion 3 of Appendix A to 10 CFR 50.

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

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

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

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

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

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

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

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC web site at http://www.nrc.gov/reading-rm/ADAMS.html (the Public Electronic Reading Room). Therefore, to

the extent possible, the response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

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

Dated this 16th day of April 2002

PRELIMINARY AGENDA

Nuclear Safety Research Conference (NSRC)
October 28-30, 2002
Marriott at Metro Center
(775 12th St. NW, Washington DC)

Monday, October 28, 2002

8:30-8:45am	**Opening Remarks**	
8:45-9:30am	**Keynote Speaker**	
9:30 - 11:30am	**Panel Discussion - Advanced Reactors** Objective: This panel will discuss the regulatory research needed to support the licensing of advanced reactor designs and focus on the kind of research that is needed to resolve technical/policy issues. Potential panel members to include: Representatives from DOE, Industry, University, public interest groups, and NRC.	
11:30 - 1:00pm	**LUNCH**	
1:00 - 5:00pm	**Degradation of Reactor Coolant Pressure Boundary Materials** Objective: To describe the results of research addressing the response of reactor coolant pressure boundary materials to active degradation mechanisms and mitigation/repair methodologies. Potential topics for presentation include: - Discussion of PWSCC, Alloy 600, and welds	**Advanced Reactors Session** Objective: To present specific work or planned work on gas cooled reactors and present DOE's envision for Generation IV reactors. Potential topics for presentation include: - Test matrix for TRISO fuel - Applications of graphite technology to advanced reactors - Generation IV roadmap - GRSAC code as a regulatory tool

Tuesday, October 29, 2002		
8:30 - 9:30am	**Plenary - NRC Chairman Speech**	
9:45 - 11:45am	**Fuels Session** Objective: To describe recent work on the technical basis for embrittlement criteria and evaluation models that are applicable at high burnup for loss-of-coolant accident (LOCA) analysis and for establishing optional performance-based criteria in 10 CFR 50.45 Potential topics for presentation include: - Testing of high burnup BWR fuel under simulated LOCA conditions - Measurement of oxidation kinetics of cladding from high burnup fuel - Testing for ductility in cladding subjected to LOCA conditions - Analysis of high burnup fuel behavior under LOCA conditions	**Formal Decision Methods and Nuclear Safety Research** Objective: To describe research activities for developing the technical basis and enhancing the transparency and objectivity of decision-making in the regulatory environment. Potential topics for presentation include: - The formal decision making framework - Probabilistic risk analysis: a component of formal decision-making - Research activities in the field of regulatory decision-making - Stakeholder input to the decision-making process
11:45 - 1:15pm	**LUNCH**	
1:15 - 5:00pm	**Dry Cask Storage and Transportation of Spent Nuclear Fuel Session** Objective: To communicate recent accomplishments and future plans to assess key safety issues related to spent fuel transportation and storage in dry casks. Potential topics for presentation include: - Determination of the performance of dry casks during transportation - Risk Assessment of on-site dry casks - Assessment of structural integrity of on-site storage casks	**Fuels Session** Objective: To describe recent work on the technical basis for fuel enthalpy criteria that are applicable at high burnup for reactivity-initiated accident (RIA) analysis and for modifying Regulatory Guide 1.77. Potential topics for presentation include: - Testing of high burnup fuel under simulated RIA conditions - Measurement of cladding-to-coolant heat during transients - Measurement of cladding mechanical properties applicable to RIA conditions - Analysis of high-burnup fuel behavior under RIA conditions

Wednesday, October 30, 2002		
8:30-9:15am	**Plenary - Commissioner Speech**	
9:30-11:30am	**Panel Discussion - Risk Informed Initiatives** Objective: This panel will communicate recent improvements on how NRC uses risk information in regulatory decision making and how work in the Office of Nuclear Regulatory Research supports such uses. Potential panel members include: Representatives of industry, EPRI, international organizations, and NRC.	
11:30 - 1:00pm	**LUNCH**	
1:00 - 3:00pm	**Clearance Session (Session A)** Objective: To discuss the status on the development of the technical basis for control of slightly contaminated materials. Potential topics for presentations include: - Surveys of volumetric contamination and difficult geometries - Revision of NUREG-1640 (individual doses) - Follow-on work to address collective doses	**PRA Session (Methods/Analysis, and Operational Experience)** Objective: To communicate recent advances in risk analysis methods as well as recent advances in using operational data in agency regulatory activities. Potential topics for presentation include: - Fire risk analysis - Human reliability analysis - Assessment of uncertainties - PRA standards - Standardized plant analysis risk (SPAR) models - Risk-based performance indicators
3:15pm	**CLOSING REMARKS**	

United States
Nuclear Regulatory Commission

Pressurized Thermal Shock Rule (10CFR50.61) Re-Evaluation
Risk Acceptance Criteria – Status Report

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Office of Nuclear Regulatory Research

Presented to
Advisory Committee on Reactor Safeguards
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Presentation Overview

- **Project overview and status**
- **PTS acceptance criteria – approach and issues**
- **Next steps**

Project Overview and Status

Current Rule

- **Focused on allowed degree of reactor pressure vessel (RPV) embrittlement**
- **Multi-level structure**
  - **Compare deterministically computed RPV embrittlement (RT_{PTS}) against screening criteria**
  - **If necessary, employ reasonably practicable flux reduction measures**
  - **If necessary, perform safety analysis (RG 1.154) to justify continued operation**
- **Use of risk information:**
  - **Risk implications of screening criteria explored as part of original technical basis**
  - **RG 1.154 acceptance criteria: TWCF of 5×10^{-6}/yr**

Project Objective

Reevaluate the technical basis for 10 CFR 50.61:

- **lessons learned in the application of 10 CFR 50.61 and RG 1.154, and**
- **research results developed since 1983**

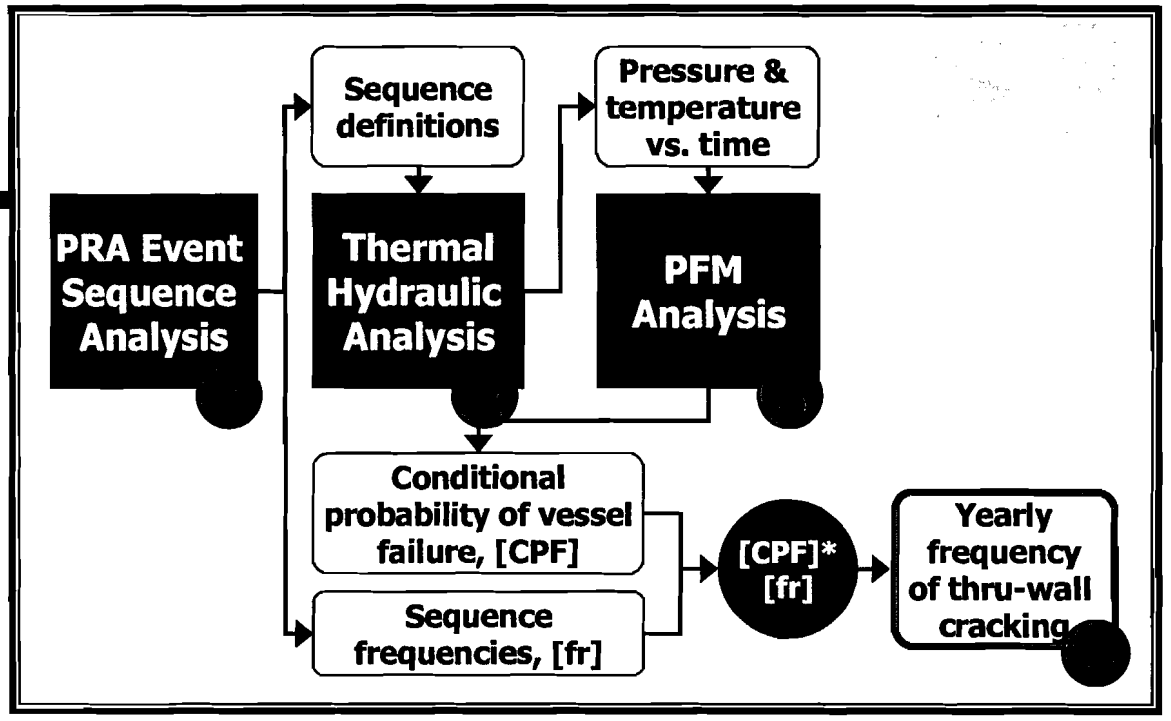
Is there a technical basis for modifying the rule?

Project Activities

- **Evaluation of frequency of PTS-induced RPV failures at pilot plants**
  - **Quantitative estimates including uncertainties**
  - **Key contributors to frequencies and uncertainties**
- **Extension to non-pilot plants**
- **Identification and evaluation of potential PTS risk acceptance metrics and criteria**

Project Status

- Approach developed to assess the PTS risk
- Involves inputs from and models developed in three different technical areas
  - Probabilistic Risk Assessment (PRA)
  - Thermal Hydraulics (TH)
  - Probabilistic Fracture Mechanics (PFM)



Plant	PRA	TH	PFM	TWCF
Oconee	draft	draft	draft	draft
Palisades	Licensee revising	1st cut	1st cut	1st cut
Beaver	draft	1st cut	1st cut	1st cut
Calvert	Licensee revising	1st cut	--	--

PTS Acceptance Criteria

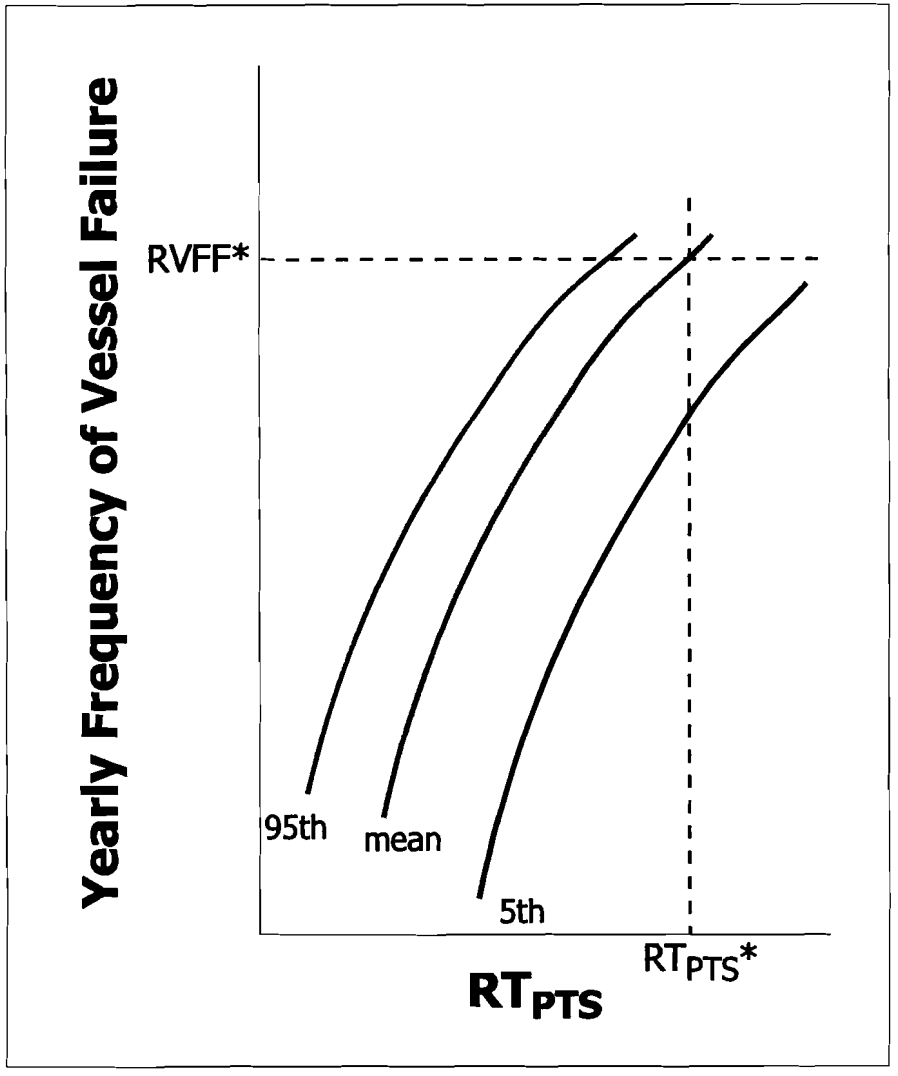
Acceptance Criteria Task

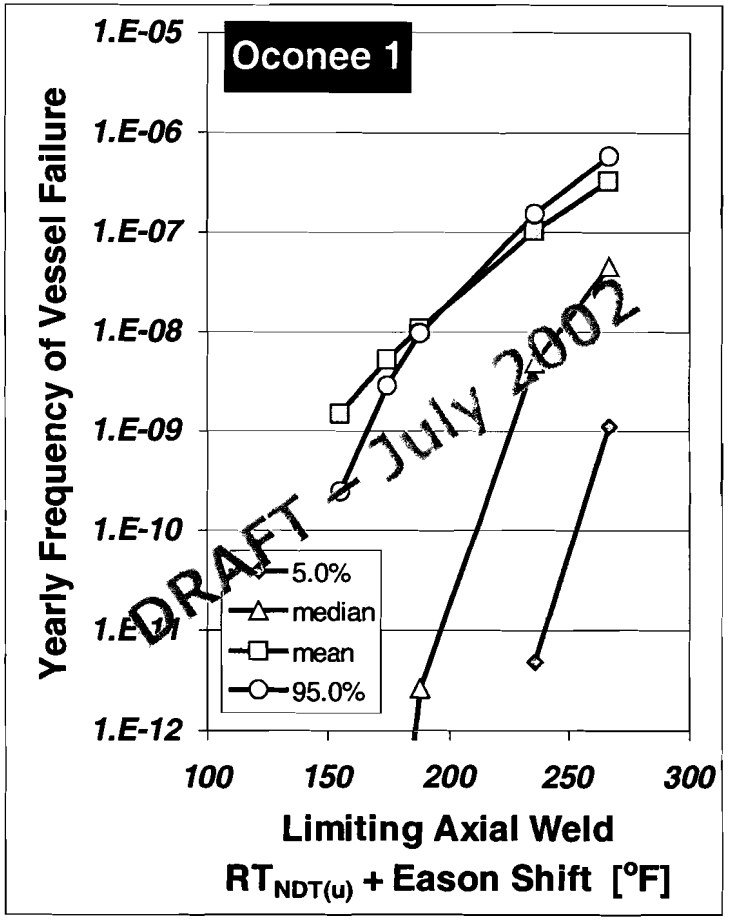
- **Objective: provide technical basis for risk-informed selection of PTS screening criteria**
- **Status report provided in SECY-02-0092 (May 30, 2002)**

Acceptance Criteria Task Assumptions

- **Key issue for rule and possible rule revisions: allowed degree of RPV embrittlement**
- **Allowed degree of embrittlement will be established in a risk-informed manner**
- **Reactor vessel failure frequency (RVFF) will be used as a key metric**
- **The criterion for RVFF will be established in a manner consistent with agency considerations of CDF and LERF in other risk-informed initiatives**

Conceptual Model for Developing PTS Acceptance Criteria





Risk Acceptance Criteria
Principles in Developing Options

- **Consistency with intent of original rule**
  - **Low risk level**
  - **Low relative contribution**
- **Consistency with recent risk-informed initiatives**
  - **Risk metrics**
  - **Risk criteria**

Risk Acceptance Criteria
Qualitative and Quantitative Options

- **Definition of RVFF**
  - **RVFF = f(PTS-induced RPV failure)**
  - **RVFF = f(PTS-induced crack initiation)**
- **RVFF acceptance limits**
  - **RVFF* = 5×10^{-6}/ry**
  - **RVFF* = 1×10^{-5}/ry**
  - **RVFF* = 1×10^{-6}/ry**

Option Evaluation Issues

- **Uncertainties in pilot plant studies**
  - **Driving sources of uncertainty**
  - **Implications for additional work on gaps in knowledge regarding containment performance and source terms**
- **Uncertainties in extensions of pilot plant studies**
  - **External events**
  - **Non-pilot plants**
- **Application of defense-in-depth and safety margins considerations**

Next Steps

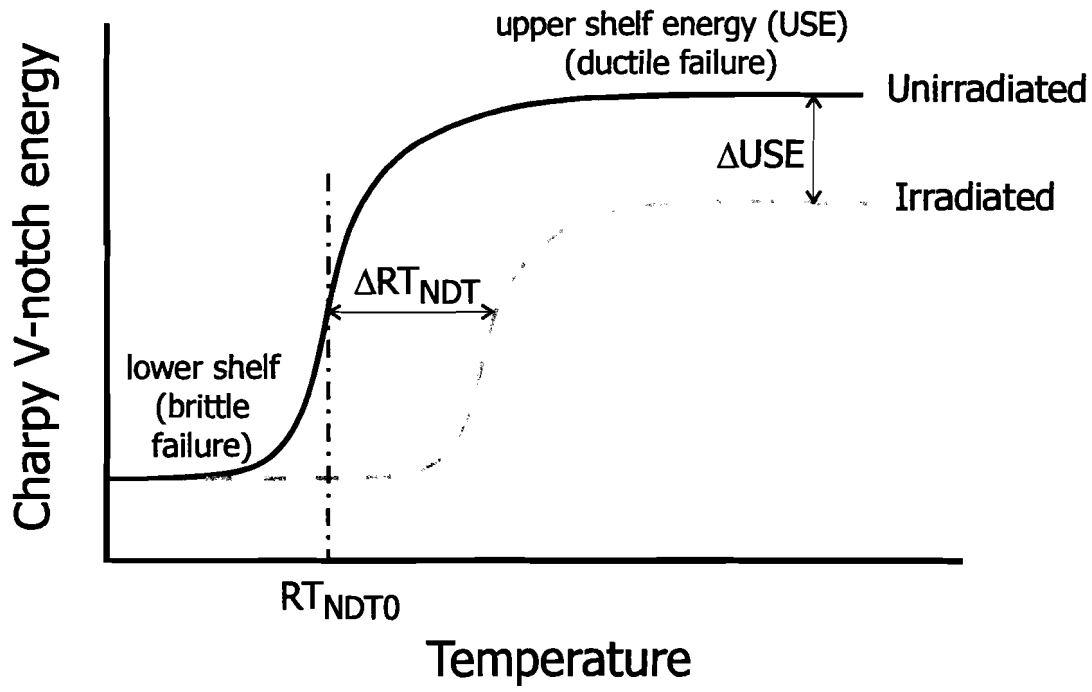
- **Complete pilot studies**
- **Address external events and extension to broader population**
- **Assess need for and feasibility of scoping study on post-RPV failure scenarios**
- **Continue interactions with international community**

Summary

- Reevaluation of PTS rule technical basis (including recommendations) scheduled for completion in December 2002
- Reevaluation addresses
  - PTS risk at selected pilot plants (quantitative estimates including uncertainties + drivers of results)
  - Extension to non-pilot plants
  - Potential PTS risk acceptance criteria
- Risk acceptance criteria options have been identified and will be evaluated
- Defense-in-depth and safety margins considerations may play a leading role in recommendations

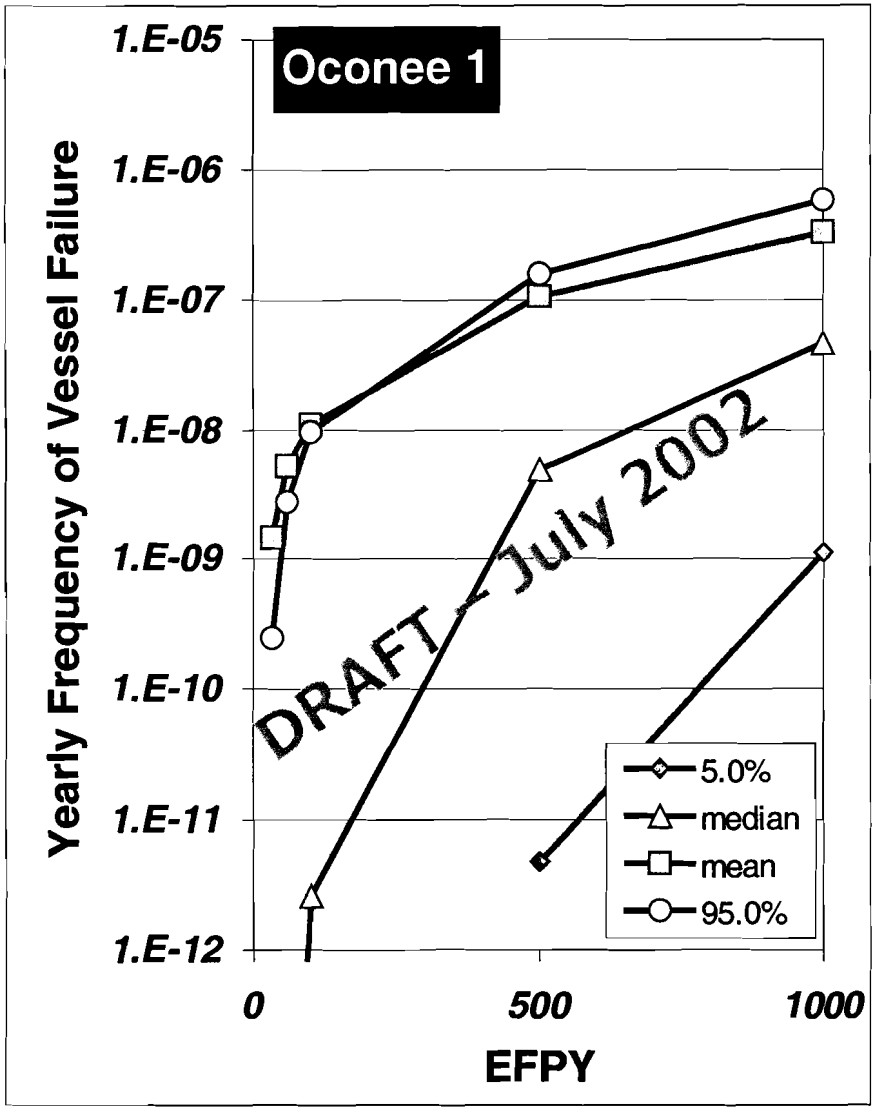
Backup Slides

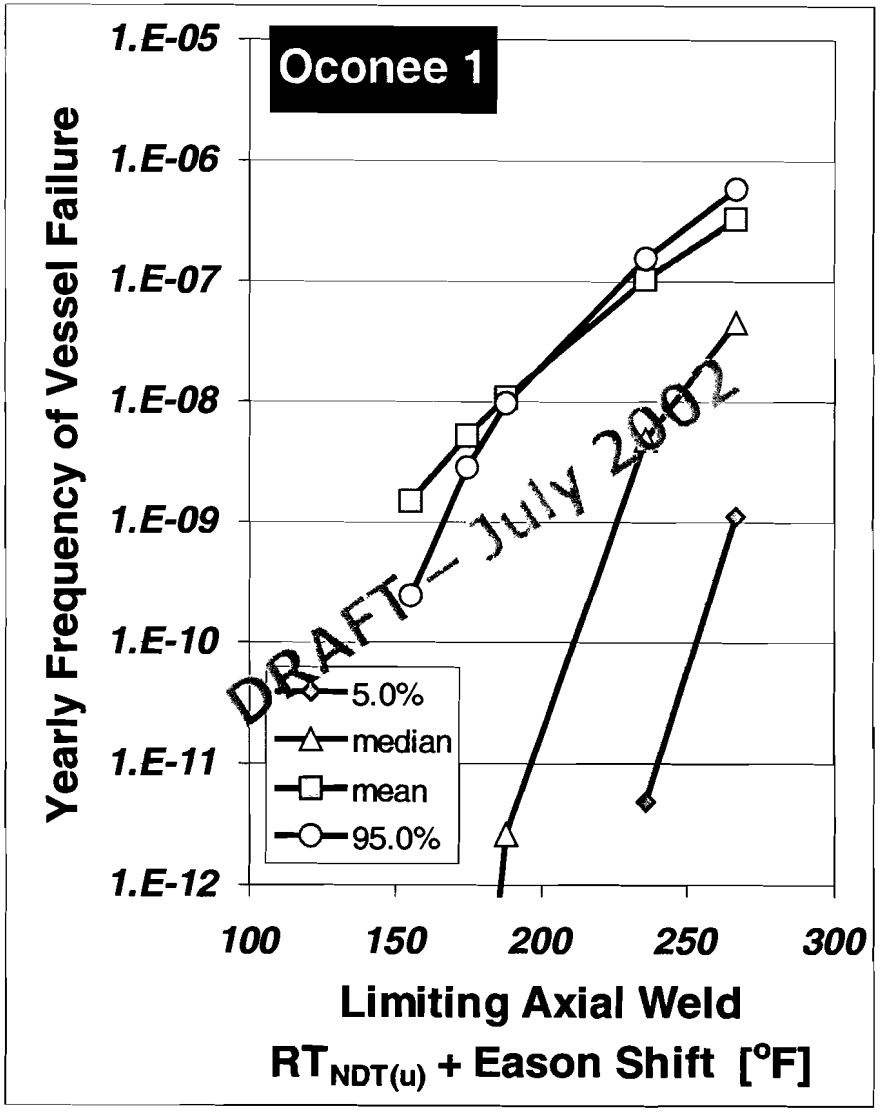
Irradiation and Embrittlement



Oconee 1

(Current Draft Results, July '02)





Status of Regulatory Guide 1.174

Advisory Committee on Reactor Safeguards

Mary Drouin
John Lane
ACRS Full Committee Meeting
July 10, 2002

PURPOSE OF PRESENTATION

- Provide status on final version of Revision 1 to Regulatory Guide 1.174
- Request ACRS concurrence in publication of Rev. 1

BACKGROUND

- Regulatory Guide 1.174 is intended to be periodically updated
  - Lessons learned from ongoing issues, such as those at Davis-Besse will be considered in future revisions

- Issued draft Revision 1 to Regulatory Guide 1.174 and SRP Chapter 19 in June 2001 with 90-day comment period

- Staff presentation to ACRS (2-5-02) on final changes to Revision 1
  - no issues raised by ACRS
  - staff recommendations unchanged

INITIAL PROPOSED CHANGES

1. Risk-related information may now be requested if new, unforeseen hazards or a substantially greater prospect for a known hazard emerges
   –Reporting responsibilities not explicitly stated or defined
   –ACRS, CRGR and Commission approval received

2. Increases in power level, fuel burnup rates and the use of mixed oxide fuel may effect plant risk profiles, and consequently current risk parameters, such as LERF, may need to be re-evaluated

3. Identification and description of scope and minimal technical attributes comprising a PRA based upon SECY-00-0162, Attachment 1

4. Example provided of the use of risk insights in the decision making process

COMMENT REVIEW

Comments received from:

Individuals

Nuclear industry – utilities, industry groups

Comment Summary:

1. Risk-related information–no comments received
2. Increases in power level, fuel burnup uprates, and use of mixed-oxide fuel
   - No basis or justification since power level increase requests already exceed 3800 MWt
   - Additional staff guidance should be provided as to whether burnup rates pertain to core average, bundle average or peak rod exposure
3. Additions from SECY-00-0162
   - Inadequate explanation as to their purpose and use
   - Inappropriate document to include the information
4. Risk-informed example-no comments

STAFF RECOMMENDATION FOR REVISION 1 CHANGES

- Issue Revision 1 to Reg Guide 1.174 with:
  - Risk-related information–no change from draft (Attachment 1)
  - Increase in power level, fuel burnup rates and use of mixed oxide fuel–add cautionary note that the staff is evaluating their impact on LERF (Attachment 2)
- Rather than include information from SECY-00-0162 and staff endorsements of consensus standards and industry peer review process in Reg Guide 1.174, incorporate in separate regulatory guide
  - under development
- Risk insights examples–add additional ones to new regulatory guide

SUMMARY

- Staff requests ACRS letter recommending publication of Revision 1 to Regulatory Guide 1.174
  - Revision 1 adds appropriate guidance regarding staff requests for risk related information for new, unforeseen hazards or when a substantially greater prospect for a known hazard emerges
  - Revision 1 appropriately notes that the staff is evaluating the impact on LERF of increases in power level, fuel burnup rates and the use of mixed-oxide fuel
- Periodic updates to the reg guide will address on-going lessons learned

ATTACHMENT 1

Licensee-initiated LB changes that are consistent with currently approved staff positions (e.g., regulatory guides, standard review plans, branch technical positions, or the Standard Technical Specifications) are normally evaluated by the staff using traditional engineering analyses. A licensee generally would not be expected to submit risk information in support of the proposed change.

Licensee-initiated LB change requests that go beyond current staff positions may be evaluated by the staff using traditional engineering analyses as well as the risk-informed approach set forth in this regulatory guide. A licensee may be requested to submit supplemental risk information if such information is not submitted by the licensee. If risk information on the proposed LB change is not provided to the staff, the staff will review the information provided by the licensee to determine whether the application can be approved. Based on the information provided, using traditional methods, the NRC staff will either approve or reject the application.

However, licensees should be aware that special circumstances may arise in which new information reveals an unforeseen hazard or a substantially greater potential for a known hazard to occur, such as the identification of an issue related to the requested LB change that may substantially increase risk. In such circumstances, the NRC has the statutory authority to require licensee action above and beyond existing regulations and may request an analysis of the change in risk related to the requested LB change to demonstrate that the level of protection necessary to avoid undue risk to public health and safety (i.e., "adequate protection") would be maintained upon approval of the requested LB change.

This regulatory guide describes an acceptable method for the licensee and NRC staff to use in assessing the nature and impact of LB changes when the licensee chooses to support or is requested by the staff to support the changes with risk information. The NRC staff would review these LB changes by considering engineering issues and applying risk insights. Licensees who submit risk information (whether on their own initiative or at the request of the staff) should address each of the principles of risk-informed regulation discussed in this regulatory guide. Licensees should identify how their chosen approaches and methods (whether quantitative or qualitative, deterministic or probabilistic), data, and criteria for considering risk are appropriate for the decision to be made.

Additional guidance is provided to the NRC staff (in Appendix D to Chapter 19 of the Standard Review Plan, Ref. 3) regarding the circumstances and process under which NRC staff reviewers would request and use risk information in the review of non-risk-informed license amendment requests.

The guidance provided in this regulatory guide does not preclude other approaches for requesting changes to the LB. Rather, this regulatory guide is intended to improve consistency in regulatory decisions in areas in which the results of risk analyses are used to help justify regulatory action. As such, the principles, process, and approach discussed herein also provide useful guidance for the application of risk information to a broader set of activities than plant-specific changes to a plant's LB (i.e., generic activities), and licensees are encouraged to use this guidance in that regard.

ATTACHMENT 2

Current LERF guidelines are based on assumptions of reactor power level, fuel burnup rates, and extent of the use of mixed oxide fuel. The staff is undertaking an evaluation of the impact, if any, of increases in these parameter on LERF.

The technical review that relates to the risk evaluation will address the scope, level of detail, and technical acceptability of the analysis, including consideration of uncertainties as discussed in the next section. Aspects covered by the management review are discussed in Section 2.2.6, Integrated Decisionmaking, and include factors that are not amenable to PRA evaluation.

2.2.5 Comparison of PRA Results with the Acceptance Guidelines

This section provides guidance on comparing the results of the PRA with the acceptance guidelines described in Section 2.2.4. In the context of integrated decisionmaking, the acceptance guidelines should not be interpreted as being overly prescriptive. They are intended to provide an indication, in numerical terms, of what is considered acceptable. As such, the numerical values associated with defining the regions in Figures 3 and 4 of this regulatory guide are approximate values that provide an indication of the changes that are generally acceptable. Furthermore, the state-of-knowledge, or epistemic, uncertainties associated with PRA calculations preclude a definitive decision with respect to which region the application belongs in based purely on the numerical results.

The intent of comparing the PRA results with the acceptance guidelines is to demonstrate with reasonable assurance that Principle 4, discussed in Section 2, is being met. This decision must be based on a full understanding of the contributors to the PRA results and the impacts of the uncertainties, both those that are explicitly accounted for in the results and those that are not. This is a somewhat subjective process, and the reasoning behind the decisions must be well documented. Guidance on what should be addressed follows in Section 2.2.5.4; but first, the types of uncertainty that impact PRA results and methods typically used for their analysis are briefly discussed. More information can be found in some of the publications in the Bibliography.

2.2.5.1 Types of Uncertainty and Methods of Analysis

There are two facets to uncertainty that, because of their natures, must be treated differently when creating models of complex systems. They have recently been termed aleatory and epistemic uncertainty. The aleatory uncertainty is that addressed when the events or phenomena being modeled are characterized as occurring in a "random" or "stochastic" manner, and probabilistic models are adopted to describe their occurrences. It is this aspect of uncertainty that gives PRA the probabilistic part of its name. The epistemic uncertainty is that associated with the analyst's confidence in the predictions of the PRA model itself, and it reflects the analyst's assessment of how well the PRA model represents the actual system being modeled. This has been referred to as state-of-knowledge uncertainty. In this section, it is the epistemic uncertainty that is discussed; the aleatory uncertainty is built into the structure of the PRA model itself.

Because they are generally characterized and treated differently, it is useful to identify three classes of uncertainty that are addressed in and impact the results of PRAs: parameter uncertainty, model uncertainty, and completeness uncertainty. Completeness uncertainty can be regarded as one aspect of model uncertainty, but because of its importance, it is discussed separately. The

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Safety culture: a survey of the state-of-the-art

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Abstract

This paper discusses the evolution of the term 'safety culture' and the perceived relationship between safety culture and safety of operations in nuclear power generation and other hazardous technologies. There is a widespread belief that safety culture is an important contributor to safety of operations. Empirical evidence that safety culture and other management and organizational factors influence operational safety is more readily available for the chemical process industry than for nuclear power plant operations. The commonly accepted attributes of safety culture include good organizational communications, good organizational learning, and senior management commitment to safety. Safety culture may be particularly important in reducing latent errors in complex, well-defended systems. The role of regulatory bodies in fostering strong safety cultures remains unclear, and additional work is required to define the essential attributes of safety culture and to identify reliable performance indicators. Published by Elsevier Science Ltd.

Keywords: Safety culture; Human reliability; Safety regulation

1. Introduction

The nuclear industry and the US Nuclear Regulatory Commission (NRC) explicitly recognized the importance of management and organizational factors to nuclear facility safety in the aftermath of the accident at Three Mile Island (TMI) Unit 2. Following the Chernobyl accident, the International Nuclear Safety Advisory Group (INSAG) introduced the term 'safety culture' to denote the management and organization factors that are important to safety [1]. Although INSAG intends 'safety culture' to capture all the management and organizational factors relevant to safe plant operation, many investigators use the term more narrowly. 'Safety culture' is often used to denote an element of organizational culture, that, in turn, is a component of the broader term 'management and organizational factors.'

Although major accidents often involve an unsafe act (or failure to act) by an individual, they may also involve conditions created by an organization that can magnify the consequences. The NRC's investigation of the accident at TMI reported to the Commissioners and the public that, "The one theme that runs through the conclusions we have reached is that the principal deficiencies in commercial reactor safety today are not hardware problems, they are management problems [2]." Later the report stated, "The NRC, for its part, has virtually ignored the critical areas of operator training, human factors engineering, utility management and technical qualifications." That sentence captures the basis for much of the NRC's regulatory agenda in the years following the accident, as well as the industry's agenda to improve plant operations.

The NRC's post-TMI action plan included a large number of issues under the general heading of human factors. The major categories included operator qualifications and training, staffing levels and working conditions, the man–machine interface, emergency operating procedures, human reliability, and organizational and management effectiveness.

Independent of the initiatives that the NRC undertook, the industry saw a need to improve the quality of nuclear operations. The Institute of Nuclear Power Operations was established by the electric utilities that owned and operated nuclear power plants to foster excellence in plant operations.

Confidence in facility management and human performance within the international nuclear power community was severely damaged by the Chernobyl accident in 1986. In its report of the Chernobyl post-accident review meeting [1], INSAG stated that, "The vital conclusion drawn is the importance of placing complete authority and responsibility for the safety of the plant on a senior member of the

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[1] The views expressed in this paper are the author's and do not necessarily represent the views of the Advisory Committee on Reactor Safeguards.

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operational staff of the plant. Formal procedures, properly reviewed and approved, must be supplemented by the creation and maintenance of a 'nuclear safety culture.'"

The present paper explores the nature of safety culture and its perceived importance in the management and regulation of hazardous technologies. Its purpose is to provide a tutorial, for non-practitioners of the human performance disciplines, that addresses the following questions:

- What is safety culture?
- How can it be measured?
- How is safety culture related to safety of operations?
- How is safety culture related to the regulatory process?

Note that addressing these questions does not imply a promise to provide answers to all of them. As one investigator observed, "...the sheer multiplicity of constituent elements of a safety culture and its precept of universal involvement imply that any attempt to monitor its health... is bound to be complex... [3]."

Because the term 'safety culture' was introduced by INSAG, we first look at INSAG's development of the idea, and the structure it designed for evaluation and implementation. Next, we consider the intellectual foundation of the concept, independent of the INSAG construct. We then discuss the larger issue of human performance, and the place of safety culture within that context. Since the ultimate objective is to establish a relationship between safety culture and the safety of facility operations, we next define the steps required to demonstrate such a link, and review some of the work that has been published toward that end. Finally, we look at the relationship between safety culture and the regulatory process, and identify areas where additional work would appear to be beneficial.

2. Evolution of the term 'safety culture'

Having introduced the term 'safety culture' into the nuclear safety discussion, INSAG expanded on its importance in INSAG-3 [4], explaining that "The phrase 'safety culture' refers to a very general matter, the personal dedication and accountability of all individuals engaged in any activity which has a bearing on the safety of nuclear power plants." INSAG-4 [5] develops the concept of safety culture in considerable detail, observing, "...the meaning of the term [in INSAG-1 and INSAG-3] was left open to interpretation and guidance was lacking on how Safety Culture could be assessed."

INSAG-4 defines safety culture as: "...that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance." It then explains that, "Safety Culture is attitudinal as well as structural, relates both to organizations and individuals, and concerns the requirements to match all safety issues with appropriate perceptions and action."

Since the definition of safety culture is related to personal attitudes and habits of thought and to the style of organizations, INSAG-4 suggests that, "...such matters are generally intangible; that nevertheless such qualities lead to tangible manifestations; and that a principal requirement is the development of means to use the tangible manifestations to test what is underlying." Arguing that "...sound procedures and good practices are not fully adequate if merely practiced mechanically...," INSAG-4 holds that "...Safety Culture requires all duties important to safety to be carried out correctly, with alertness, due thought and full knowledge, sound judgment and a proper sense of accountability."

The body of INSAG-4 is devoted to articulating what it terms universal features of a safety culture and the identification of broad characteristics (tangible evidence) of an effective safety culture. The approach to both topics, universal features and tangible evidence, is to provide detailed lists of the desired attributes.

Within the operating organization, INSAG looks first at the corporate policy level: "Safety Culture flows down from actions by the senior management of an organization.... The primary indication of corporate level commitment to Safety Culture is its statement of safety policy and objectives." Other indicators of safety culture should be found in regular reviews of the organization's safety performance and the evaluation of individual attitudes toward safety as part of the staff selection and promotion process.

To find tangible evidence of safety culture among the operating personnel of a particular power plant, INSAG suggests that the three aspects to be considered are (1) the environment created by local management, (2) the attitudes of individuals at all levels, and (3) the actual safety experience at the plant. The working environment should include defined safety responsibilities and detailed practices at all levels. Training and education should ensure staff knowledge about possible errors in each individual's area of activity. Safety concerns should be given a high level of visibility by plant inspections, audits, visits by senior officers, and safety seminars. Satisfactory facilities, including tools, equipment and information, should be provided to the staff.

Individual attitudes are reflected by adherence to procedures, stopping to think when facing an unforeseen situation, and management respect for a good safety attitude. Managers should take opportunities to show that they will put safety concerns ahead of power production if circumstances warrant. Development of local practices for enhancing safety, such as error reporting, should be encouraged.

Ultimately, in INSAG's view, the effectiveness of the organization's safety culture should be reflected in the performance of the facility. Plant performance indicators,

including plant availability, number of unplanned shutdowns, or radiation exposure, are a reflection of attention to safety. Significant events that occur should be analyzed to determine what they reveal about staff strengths and weaknesses. The rigor of the reviews, and the effectiveness of any resulting corrective actions, are important safety culture indicators.

INSAG-4's conclusion is that safety culture is now a commonly used term, and that it is important to give practical value to the concept. This includes identifying attributes that may be used to judge the strength of safety culture in specific instances. An appendix is included that identifies 143 questions INSAG suggests are worth examining when the effectiveness of safety culture is to be judged in a particular situation.

Following the publication of INSAG-4, the International Atomic Energy Agency (IAEA) published guidelines [6] "... for use by any organization wishing to conduct a self-assessment of safety culture." Titled "ASCOT Guidelines," (Assessment of Safety Culture in Organizations Team Guidelines), the document summarizes the concept of safety culture and then describes a process for assessing safety culture. The ASCOT guidelines restate the basic INSAG questions and then expand on them with approximately 300 'Guide Questions.'

Missing from the ASCOT guidelines, as well as from INSAG-4, is any indication of how an overall conclusion should be drawn from the collected answers to all the questions. Possibly, the intent of judging safety culture does not include an overall conclusion. It may be that in each area the intent is simply to identify deficiencies and make suggestions for improvements in those areas. Still, it would seem that a facility with a poor safety culture might be left with an overwhelming list of corrective actions. Unless some guidance is provided on how to proceed, the evaluation may provide little help. It would seem inevitable that the review team will conclude that a safety culture is superior, acceptable, or deficient, and attempt to provide the proper degree of motivation for corrective actions. The ASCOT guidelines do not appear to help in the formulation of overall conclusions.

The fundamental problem with INSAG's approach to safety culture is that it specifies in great detail what should be included, but provides little guidance on overall criteria for acceptability. Furthermore, no link is made (or even seems possible) between safety culture as INSAG defines it and human performance or human reliability. A positive relationship is simply assumed. One of the goals to be reached in risk informed regulation is to advance the state of the art in probabilistic risk assessment (PRA) to account for the probability of human error, and to further account for the contribution of human skills to recovering from accident sequences. The INSAG approach appears to make little contribution to either. It is also true that the INSAG work does not establish the link between a good safety culture and safe plant performance. The relationship is, again, simply assumed. While it seems plausible that the sum total of the indicators of a strong safety culture would imply safe plant operations, that is not the same as demonstrating a cause and effect relationship. The possibility remains that safe plant operations can be fostered, perhaps even more effectively, by other organizational characteristics.

3. Organizational culture

Although INSAG has borrowed the term 'culture' from either anthropologists or the organizational development community (who in turn borrowed it from anthropologists), the INSAG publications make no reference to the bodies of literature in those fields. In fact, no attempt is made to link 'safety culture' with 'culture' as the term is used elsewhere. Nevertheless, suggestions that 'culture' might help explain organizational behavior, and that management and organizational factors could influence safety performance, both predated INSAG's introduction of the term 'safety culture.'

Ostram et al. [7], note that "Heinrich's Domino Theory developed in the 1930s was based on the premise that a social environment conducive to accidents was the first of five dominos to fall in an accident sequence." Deal and Kennedy [8] attempted to establish why the structure of an organization often did not explain the control of work activities, and suggested culture as the undocumented influence. Uttal [9], after several books had been published on the human underpinnings of business, summarized the meaning of organizational culture as: a system of shared values (what is important) and beliefs (how things work) that interact with a company's people, organizational structures, and control systems to produce behavioral norms (the way we do things around here).

Bridges [10] raises a cautionary note regarding the current practice of assuming an organizational culture exists, can be reasonably well defined, and can be changed. He observed that there are several important differences between 'culture' as commonly used by anthropologists and 'culture' as applied to organizations by management consultants. He noted that, "Like many who borrow concepts from other fields, organizational writers have oversimplified matters to such an extent that their concept has lost much of its connection to the usages that are current in the field to which it belongs."

Apostolakis and Wu [11] questioned whether the term safety culture is appropriate by suggesting it is too narrowly drawn. "When the subject is culture, we must question the wisdom of separating safety culture from the culture that exists with respect to normal plant operation and power production. The dependencies between them are much stronger because they are due to common work processes and organizational factors." Reason [12] also notes that the quality of production and protection depend on the same organizational processes.

Despite the reservations of some investigators, safety culture seems to be accepted as an appropriate and useful concept, even though its relationship to 'culture' in the usual sense is tenuous. Ascribing the usually understood characteristics of 'culture' to 'safety culture' should be done with some caution. The term itself implies that it is a subset of a larger 'organizational culture.'

Safety culture may not capture all the management and organizational factors that are important to safe plant operation, but it has acquired a place in the literature. Although the literature does not support any single definition of safety culture, it is probably reasonable to settle on a model that represents organizational culture as a particular application of the larger concept of culture, and then considers safety culture as a subset of organizational culture. The definition chosen for 'safety culture' should then be consistent with its parent terms, 'culture' and 'organizational culture.' The ultimate objective is to establish a link between safety culture and safety of operations. That process requires not only a definition, but also a delineation of the characteristics or attributes of safety culture. Such attributes should be consistent with the chosen definition, but they are probably more important than the definition.

4. Safety culture in context

Safety culture, however defined, is part of the larger issue of human factors. In a 1988 study requested by the NRC, the National Research Council recommended a human factors research agenda to be undertaken by the NRC [13]. The recommended program included five major areas: human–system interface design, the personnel subsystem, human performance, management and organization, and the regulatory environment. The first two areas are primarily related to system design and personnel training, respectively, and are only indirectly related to safety culture. The next two areas, human performance and management and organization, are most closely related to the idea of safety culture. Under human performance, the National Research Council identifies the highest priority topic as causal models of human error. Under management and organization, it identifies two high priority topics: the impact of regulations on the practice of management, and organizational design and a culture of reliability. Equating 'culture of reliability' to what we are now calling 'safety culture' seems like a reasonable step.

Safety culture is also related to the last area mentioned by the National Research Council, regulatory environment, but not in a simple way. Regulatory activities influence the overall environment in which licensee organizations operate, and hence affect the organizational cultures that evolve. Regulatory activities also have the potential for being counterproductive, especially if they appear to shift the responsibility for safety from the operator to the regulator.

4.1. Human error

The focal point of human factors concerns is the performance of individuals. The term 'human error' is generally understood to mean an unsafe act by a system operator. The consequences of such an act may or may not be severe, depending on other circumstances. Such 'other circumstances' are often the product of organizational factors that established other important conditions that determine system response. In his taxonomy of human error, Reason [14] distinguishes between active errors, "whose effects are felt almost immediately," and latent errors, "whose adverse consequences may lie dormant within the system for a long time." Active errors are usually associated with system operators such as airplane pilots, air traffic controllers, or power plant control room personnel. Latent errors are normally associated with personnel removed from operations, such as design, construction and maintenance personnel.

Modeling human error is necessary to the complete understanding of the human contribution to system safety. Information from human error models and associated data gathering are an important input to the process of PRA. The probability of an operator committing an error and causing a system to fail to perform its intended function is as important as a component failure leading to the same result. Modeling unsafe acts, however, is only part of the story. The consequences of those acts often depend on latent errors. It seems reasonable to expect that safety culture, and probably other organizational factors, will have a significant influence on both the frequency of unsafe acts and the probability of latent errors.

4.2. Organizational accidents

In *Human Error*, Reason argues that most of the root causes of serious accidents are present within the system long before an obvious accident sequence can be identified. He contends that "... some of these latent failures could have been spotted and corrected by those managing, maintaining and operating the system in question." In a second book [12], he looks at the organizational functions involved in creating or mitigating accidents. He argues that "...human error is a consequence, not a cause. Errors ... are shaped and provoked by upstream workplace and organizational factors [12, p. 126]." He calls the accidents resulting from such upstream factors 'organizational accidents.' Understanding the management and organizational factors that can either reduce or identify and correct latent errors is an important element in reducing the frequency and consequences of accidents.

Typically, organizational accidents involve "...the interaction of latent conditions with local triggering events [12, p. 35]." Reason describes organizational accidents in terms of organizational factors, local workplace factors and unsafe acts. The organizational factors and local workplace factors

not only interact directly, but each may create latent condition pathways. Accidents with significant losses occur when all these conditions align in such a way that the defenses built into a system are overwhelmed. Reason maintains that latent conditions may be sufficient to cause accidents, and that they are always present in the system.

The NRC has developed a human reliability analysis method called ATHEANA (A Technique for Human Event Analysis) [15]. The issues addressed by the concept of safety culture in general, and latent errors in particular, provide what is called the 'error forcing context' for ATHEANA. The model provides a structured search process for human failure events, including detailed search processes for error-forcing context, and an improved representation of human–system interactions.

The ATHEANA process contributes to the objective of systematically identifying important management and organization factors that contribute to significant event sequences. The ATHEANA analysis of the Wolf Creek drain-down event [16] identified a number of management and organization factors that contributed to the occurrence of the event. Contributors included incompatible work activities, a compressed outage schedule, poor mental models of the systems and valves, heavy reliance on the control room crew to identify problems, and inadequate reviews of procedures prior to use.

The influence of latent errors was identified in a recent study by the Idaho National Engineering and Environmental Laboratory (INEEL) [17]. One objective of the study was to identify the influence of human performance in significant operating events. INEEL analyzed 35 operating events, 20 of them using PRA methods. Event importance, as measured by conditional core damage probability, ranged from 1.0×10^{-6} to 5.2×10^{-3}.

INEEL found that most identified errors were latent, with no immediate observable impact. The ratio of latent to active errors was 4:1. Latent errors identified included design deficiencies, failure to correct known problems, incomplete design change testing, inadequate maintenance practices and post-maintenance testing, and poor work package quality assurance. The INEEL discussion does not seem to distinguish between the latent error itself, such as a design deficiency, and the root cause of the latent error, such as inadequate design review or incomplete design change testing. Active errors included failures in command and control (such as loss of phone communications), and incorrect operator actions (such as incorrect system line-ups or acting without procedural guidance).

The INEEL findings are supported by other analyses. In discussing a human performance improvement program at Duke Power Company, one Duke Power senior manager observed that "If you analyze an entire event,... you'll find it wasn't just one mistake—it was five, six or seven mistakes that occurred and there weren't enough contingencies or barriers built in to prevent the event from happening [18]."

A structured assessment by Duke Power of human performance needs identified the need for focused human error reduction training for technicians and supervisors. Although the term 'safety culture' is not used in describing the program, it incorporates elements and issues practically identical to many of those addressed by INSAG-4. One element in the Duke Power program, for example, is 'individual commitment' which includes a questioning attitude, procedure use and adherence, communications, stopping when unsure, and an overall prudent approach. The same parallels exist for the manager's commitment portion of the INSAG model and the supervisors and manager's sections of the Duke Power program. Both deal with clear priorities, goals and responsibilities, clear lines of responsibility and authority, staff skills and competence and performance assessment.

Since the program was initiated, refueling outage times at Duke Power's McGuire nuclear power station have been reduced from about 90 days to about 33 days, and capacity factors have increased from about 72% to about 89%. These results, of course, are measures of efficiency, not safety. The similarity between the management and organization factors apparently responsible for the noted improvements and those factors identified with safety culture suggests that an attempt to relate safe operations to efficient operations might be worthwhile. It is often claimed that efficient, well-managed facilities from a production standpoint are also safe facilities. That notion is not universally accepted, and probably requires a more rigorous examination than it has been given to date. Such an examination may be valuable.

5. Relating safety culture to safety performance

As noted previously, one of the omissions in INSAG's structure for establishing and evaluating safety culture is the link between safety culture and safety of operations. The INSAG approach assumes, but does not attempt to demonstrate, a positive relationship between safety culture and facility safety. There are actually two parts to this demonstration. The first part is to establish a relationship between safety culture (or its associated attributes) and safety of operations. The second part is to determine if there are suitable performance indicators that can be used to infer changes in safety culture and, thereby, predict changes in safety performance. There is a substantial body of literature that addresses the first part of the problem. There is much less work that addresses the second part. No performance indicators to gauge safety culture and its impact on safety of operations appear to have been identified and validated.

An activity diagram for establishing a relationship between safety culture and safety, the first part of the problem posed above, is shown in Fig. 1. The objective is to identify one or more measurable attributes of safety

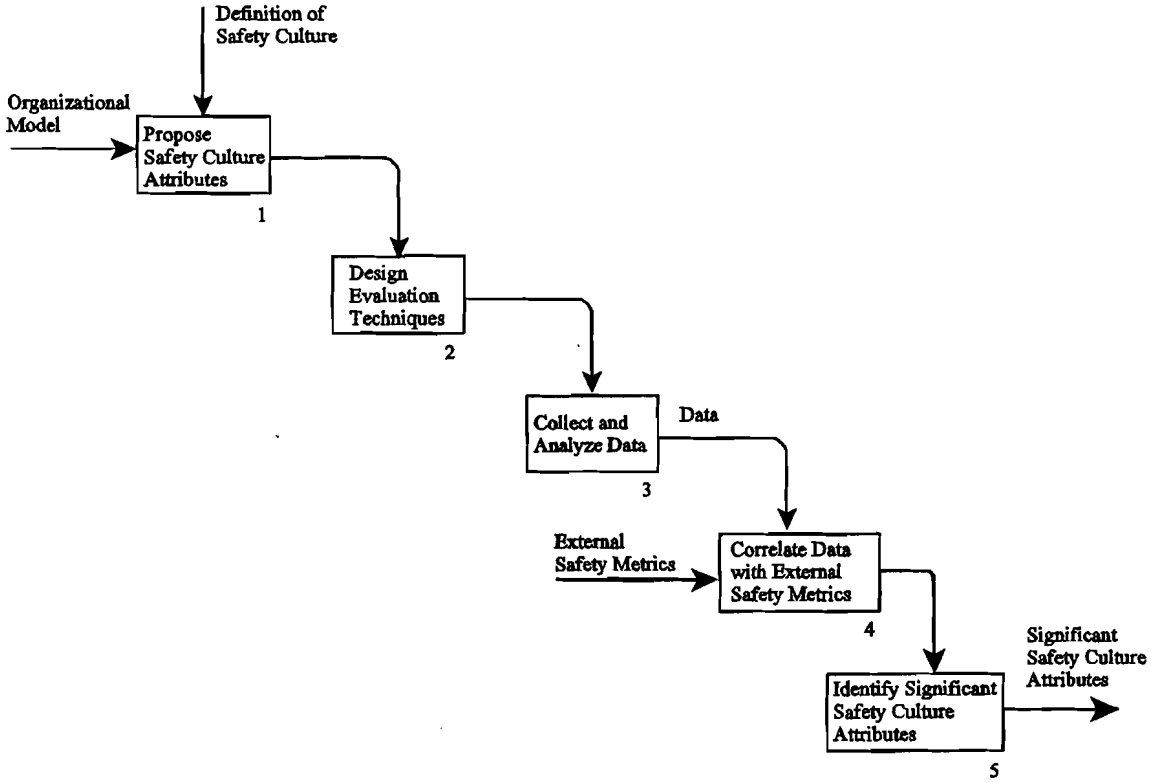

Fig. 1. Relating safety culture to safety performance.

culture that can be correlated with one or more measures of operational safety. The second part of the problem, identifying suitable performance indicators, is outlined in Fig. 2.

Research intended to show how management and organization factors affect safety of operations typically starts with a description of how a particular organization works, and attempts to identify specific, measurable organizational factors that influence safety. The process necessarily requires some measure of safety, such as the frequency of accidents. The analyst may begin by choosing an organizational model to represent how the organization works. The insights derived from that model, in conjunction with a suitable definition of safety culture, can be used to suggest attributes of safety culture that can be measured (step 1 in Fig. 1). Such attributes might include, for example, effectiveness of organizational communications, organizational learning, management attention to safety, and management expectations regarding compliance with procedures.

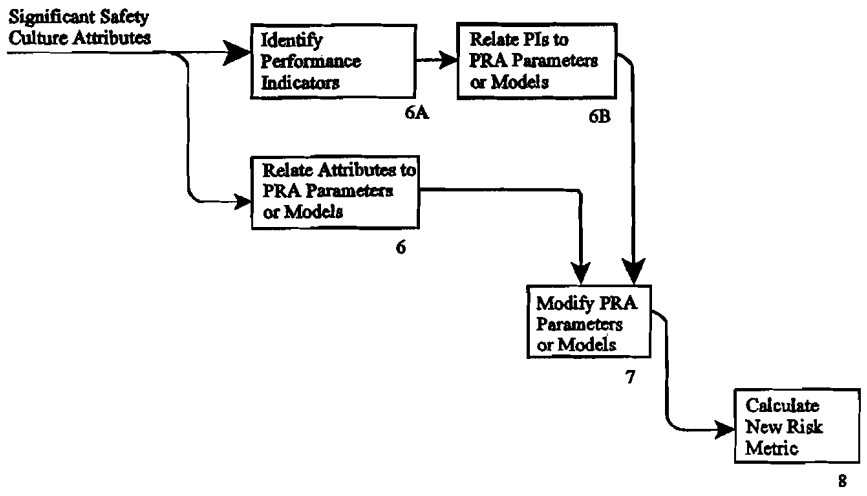

Fig. 2. Relating safety culture to risk metrics.

The next step in the process is to design methods to measure the proposed attributes in a real organization. This is typically done using audits, inspections, document reviews and personnel surveys. The tools and techniques used here often include those used by psychologists as well as those used by engineers. To continue the example, designing the measurement methods would involve finding a way to quantify 'management attention to safety' and the other proposed attributes of safety culture. Selection of the measurement techniques is obviously followed by data collection and analysis (step 3 in Fig. 1).

The next step is to correlate the attribute measurements with one or more measures of operational safety. This step obviously requires the analyst to select external safety metrics. The choices may be dictated by what measures are available. Early studies of US nuclear power plants used systematic assessment of licensee performance (SALP) evaluations, licensee event reports (LERs), and other performance indicators such as unplanned scrams, safety system actuations, and safety system failures.

Correlation of the safety culture attribute data with the chosen safety metrics (step 4) is usually done using regression analysis. The result typically will show that some of the proposed attributes have a statistically significant relationship with one or more of the chosen safety indicators. An organization with a low score on 'management attention to safety,' for example, might consistently have a relatively high rate of safety system failures. Other attributes may show no significant correlation. The output of the process at this point (step 5 in Fig. 1) is the identification of those attributes of safety culture that show a significant relationship to safety, at least as measured by the chosen safety metrics.

Addressing the second part of the problem, identification of suitable performance indicators and the impact of safety culture attributes on risk metrics, is outlined in Fig. 2. As shown in the lower path on Fig. 2, the significant safety culture attributes must be related to parameters in a PRA, such as a human error probability, a system failure probability, or a system unavailability. Essentially, a numerical value must be developed for each of the significant elements. An algorithm is then developed to relate the resulting quantification to a change in one or more PRA parameters, such as a system unavailability or failure rate. To pursue the example of the attribute 'management attention to safety,' the desired algorithm could relate a low score on this attribute to an increase in assumed equipment failure rates used as input to the PRA. It is also possible that a relationship identified between the significant attributes and the external safety metrics is not modeled in the PRA at all. In this case, the PRA model itself must be modified. The final step on this path is the calculation of core damage frequency or other chosen risk metric.

If the overall process described above is to be most useful in the safety assessment of hazardous facilities, it is desirable to identify easily obtainable performance indicators that will provide a reliable measure of the significant safety culture attributes. This possibility is illustrated in the upper path of Fig. 2. Evaluating the attribute 'management attention to safety,' for example, might require extensive data collection and data analysis. Once the relationship between 'management attention to safety' and safety system failure rates has been established, it may be possible to identify an easily observable surrogate for 'management attention to safety.' Such a performance indicator might be the fraction of employees participating in periodic safety training. This indicator could be monitored through record reviews, and would not require the personnel surveys and audits that might otherwise be necessary to measure 'management attention to safety.' If suitable performance indicators for the attributes of safety culture can in fact be identified, the performance indicators can also be related, in turn, to the PRA parameters or PRA models.

6. Modeling organizations

This section reviews some of the literature that addresses studies related to establishing a relationship between safety culture (or other management and organizational factors) and safety of operations. The models discussed in each of the studies reviewed address some of the activities represented in Figs. 1 and 2. None of them treat all activities, at least not with the same degree of thoroughness and rigor. Typically, a given study will address a few activities in detail, and acknowledge the need to address the remainder. Some studies consider safety culture to be a subset of the management and organizational factors that might affect safety, and others do not use the term 'safety culture' at all. The studies selected for review are a representative but limited sample of the available literature, not an exhaustive survey.

6.1. Chemical industry safety surveys and audits

Investigators in the chemical process industry have used safety audits and personnel surveys as the primary means of relating safety attitude or safety culture to operational safety. Investigators in this field have the advantage, if it can be deemed an advantage, that certain types of accidents occur with sufficient frequency to provide statistically valid measures of operational safety.

Donald and Canter [19] examined the relationship between employee attitudes and safety performance in the chemical process industry using the terms 'safety attitudes' and 'safety climate' instead of 'safety culture.' The authors use the term 'organizational climate' as the sum of perceptions employees have of their organizations. The climate represents the context in which behavior occurs and the basis of people's expectations.

Donald and Canter began their study by deriving six

factors associated with safety from the relevant literature:

- management commitment
- safety training
- open communication
- environmental control and management
- stable workforce
- positive safety promotion policy

Other factors were found, using expert judgment, that discriminated between factories (locations) in terms of safety climate. In order of decreasing discriminant power they were:

- importance of safety training
- effects of workpace (sic)
- status of safety committee
- status of safety officer
- effect of safe conduct on promotion
- level of risk at the workplace
- management attitudes toward safety
- effect of safe conduct on social status

These factors are summarized in the first column of Table 1.

The evaluation technique used by Donald and Canter was an employee survey based on three facets of safety attitude: people, attitude behavior, and activity. The 'people' facet was divided into five components: self, workmates, supervisors, managers, and safety representatives. Attitude behavior was divided into three components. The first was an employee 'knowing about' something related to safety, the second was an employee being 'satisfied with' something about safety, and the third was an employee 'carrying out' some action related to safety. The 'activity' facet addresses the degree to which an employee engages in activities important to safety. The elements of each of the facets were used to construct ten 'scales' to measure worker attitudes toward safety and their perception of other peoples attitudes. The ten scales are summarized in the second column of Table 1.

Table 1
Attributes related to safety in the chemical industry from Donald and Canter [19]

Attributes derived from the literature	Proposed attributes to be tested empirically
Management commitment	*People facet*
Safety training	Self
Open communications	Workmate
Environmental control and management	Manager
Stable workforce	Supervisor
Positive safety promotion policy	Safety representatives*
Attributes found using expert judgment	*Attitude behavior facet*
Importance of safety training	Satisfaction
Effects of workpace (*sic*)	Knowledge
Status of safety committee	Action
Status of safety officer	
Effect of safe conduct on promotion	*Activity facet*
Level of risk at the workplace	Active
Management attitudes towards safety	Passive
Effect of safe conduct on social status	

* Attribute marked with an asterisk did *not* correlate with low self-reported accident rates.

Question templates were designed to map all three facets into specific questions related to safety climate. One such template, for example, would be the combination of "workmate (people facet) is satisfied with (attitude facet) passive safety activity (activity facet)." An example of a question developed from that template is "To what extent are your workmates satisfied with the safety procedures they are required to follow?"

Each of the ten scales was represented by a set of questions based on the templates described earlier. In addition, each participant in the survey was asked about their involvement in accidents. These 'self-reported accident rates' were the safety metric chosen for the study. The survey was conducted at ten plants owned by the same company. The results indicated that the attitude scales were a reliable measure of safety climate. Only one scale, safety representatives, did not show a statistically significant correlation with self-reported accident rates. Overall, there was a "...clear and strong relationship between the safety attitude climate of a company and its accident performance."

Note that the work just described did not extend to the activities displayed in Fig. 2. There was no attempt at identification of performance indicators as surrogates for either the accident rates or the attributes of safety climate, nor was there any attempt to quantify the level of risk represented by particular values of the safety climate scales.

6.2. *Safety survey of a nuclear fuel reprocessing plant*

Lee [3] reported on an assessment of safety culture at the Sellafield nuclear reprocessing plant, which can perhaps be considered as belonging both to the nuclear industry and the chemical process industry. Lee noted that the concept of safety culture is not new, and had existed for some years as 'safety climate,' which in turn was one aspect of a broader 'organizational climate.'

Lee's description of the traditional approach to safety reflects the process that has been used within the US regulatory system. "The traditional approach to safety... has been retrospective, built on precedents. Because it is necessary, it is easy to think it is sufficient. It involves, first, a search for the primary (or 'root') cause of a specific accident, a decision on whether the cause was an unsafe act or an unsafe condition, and finally the supposed prevention of a recurrence by devising a regulation if an unsafe act, or a technical solution if an unsafe condition." Although maintaining that this process is necessary, Lee went on to note that it has serious shortcomings. Specifically,

"Regulations are proliferated to the point where they become incomprehensible and... resources are diverted to prevent the accident that has happened rather than the one most likely to happen."

Lee started with a list of characteristics of low accident plants distilled from a review of empirical research into the organizational aspects of safety. The list included a high level of communications, good organizational learning, a strong focus on safety, strong commitment to safety by senior management, a management leadership style that is democratic, cooperative, participative and humane, more and better quality training, good working conditions, high job satisfaction and a workforce retained for safe working habits.

These characteristics, summarized in the first column of Table 2, are similar to those used as a starting point by Donald and Canter (the first column of Table 1). Lee then identified 19 attitudes toward safety (safety culture attributes) to be tested empirically. Lee's attributes are listed in the second column of Table 2, and bear some similarity to those examined by Donald and Canter. The evaluation process involved both focus groups and an employee questionnaire consisting of 172 statements about safety. Respondents could indicate a range of agreement/disagreement on a seven-point scale. The safety metric chosen was self-reported rates of accidents involving three or more days of lost work.

Lee's results showed a strong correlation between positive safety attitudes and low accident rates. Of the 19 factors, 16 showed a statistically significant correlation, 15 of those at a very high level of significance.

Lee concluded that, "The concept of safety culture... now has widespread support. If it is a valid concept... [it] should be helpful in getting employees to understand the objectives of a safety management system... However, the sheer multiplicity of constituent elements of a safety culture and its precept of universal involvement imply that any attempt to monitor its health... is bound to be complex...."

6.3. An organizational analysis approach

Work begun for the NRC at Pacific Northwest Laboratory (PNL) in the early 1980s focused primarily on the relationship between the structure of the utility organization and safety performance. The first of these reports [20] addressed identifying appropriate organizational factors (step 1 of Fig. 1) and possible external safety metrics (the input to step 4).

Drawing on work done in organizational analysis, Osborn et al. [20], proposed a model based on categories of variables they called 'organizational contingencies' and 'intermediate outcomes.' Under the heading of organizational contingencies, potential important organizational factors were grouped into four types: environment, context, governance, and design. The utility environment includes general economic trends, regulation by the state, regulation by the NRC, support from vendor organizations, and interfaces with corporate parents. The second factor, the utility's context, includes its history, size and technology. The third factor, organizational governance or management philosophy, is characterized by three types: (1) traditional, which emphasizes a bureaucratic approach including administrative control, written policies, and elaborate written procedures; (2) modern, which emphasizes values where individual judgment is to be used to implement policy; and (3) federal, which stresses negotiation and

Table 2
Organization factors related to safety from Lee [3]

Characteristics of low accident plants	Proposed safety attitudes (safety culture attributes) to be tested empirically
High level of communication	Confidence in safety procedures
Good organizational learning	Personal caution over risks
Strong focus on safety	Perceived level of risk at work
Strong senior management commitment to safety	Trust in workforce
Democratic, cooperative, humanistic management leadership style	Confidence in efficiency of 'permit to work' system*
More and better quality training	General support for 'permit to work' system
Clean, comfortable working conditions	Perceived need for 'permit to work' system*
High job satisfaction	Personal interest in job
Workforce retention is related to working safely	Contentment with job
	Satisfaction with work relationships
	Satisfaction with rewards for good work
	Personal understanding of safety rules
	Perceived clarity of safety rules*
	Satisfaction with training
	Satisfaction with staff suitability
	Perceived source of safety suggestions
	Perceived source of safety actions
	Perceived personal control over safety
	Satisfaction with design of plant

* Attributes marked with an asterisk did *not* correlate with low accident rates.

Table 3
Management and organizational factors related to safety performance

Organizational analysis approach Marcus et al. [22]	Organizational process approach Jacobs and Haber [25]
Environmental conditions	*Administrative knowledge*
General environment	Coordination of work
Abundance of resources	Formalization
Amount of volatility	Organizational knowledge
Amount of interdependence	Roles and responsibilities
Task environment	
Abundance of resources	*Communications*
Amount of volatility	External
Amount of interdependence	Interdepartmental
	Intradepartmental
Contextual conditions	
Size (staff and budget)	*Culture*
Technological sophistication	Organizational culture
Technological variability	Ownership
	Safety culture
	Time urgency
Organizational governance	
Traditional, modern or federal	*Decision making*
	Centralization
	Goal prioritization
Organizational design	Organizational learning
Mechanistic, organic or diverse	Problem identification
	Resource allocation
Intermediate outcomes	*Human resource allocation*
Efficiency	Performance evaluation
Compliance	Personnel selection
Quality	Technical knowledge
Innovation	Training

integration of differing views through conflict resolution. The fourth factor, organizational design, includes how work is divided among units; the nature of controls placed on individuals, managers and operational units; coordination mechanisms; and developmental mechanisms, which reinforce and direct decisions by individuals.

The second category of variables, called 'intermediate outcomes,' includes four factors: compliance, efficiency, quality and innovation. These factors appear to be included in the model to account for organizational characteristics closely related to safety and to external regulation. These organizational factors, the output of step 1 in Fig. 1, are summarized in the first column of Table 3.

The safety metrics chosen for the PNL work were typical of the early attempts to identify such indicators. Included in the preliminary list are LERs, inspection and enforcement data, operating and outage data, SALP scores, personnel exposure, and operator exam scores.

A second report [21], published about a year after the preliminary work, claimed some success with the proposed approach, although the results were still labeled as preliminary. Specifically, the report concluded that plant performance data could be used to create reliable indicators of plant safety performance, that plant safety performance indicators are potentially useful for identifying causes of poor performance, and that organizational structure appears to be an important predictor of plant safety performance.

The approach described above is focused on the structure of the organization. It is based on a body of work in organizational analysis that appears to have virtually no overlap with the proponents of corporate culture. The organizational factors considered are not components of organizational culture or safety culture, and have different properties. It is interesting to note, however, that the later work by these investigators acknowledges the possible importance of the concept of organizational culture. Specifically, work done following the Bhopal, Challenger and Chernobyl accidents prompted the authors to note, "Collectively, these analyses suggest that relationships that emerge from the day-to-day operation of technologies are potentially as important as the more general state conditions and management philosophy concerns described earlier. [T]hese management relationships... are those unplanned continuing dynamics of the organization that allow it to operate with continuity and react to unanticipated conditions. They arise because individuals shape and mold the formal organization, interpret the environment and context, implement the management philosophy and generally add variety to that planned into the system [22, p. 51]."

6.4. An organizational process approach

The approach proposed and developed by Haber et al. [23], is based on organizational processes as opposed to organizational structure. As with the organizational structure approach pursued at PNL, the underlying idea is to seek statistically valid relationships between organizational factors and safe plant operations. The three-step process used was (1) development of a description of the human organization of a nuclear power plant, (2) identification of organizational and management functions and processes related to safety performance, and (3) the development of methods for measuring organizational and management factors. The overall concept was designated Nuclear Organization and Management Analysis Concept.

The assessment of organization and management factors involved three types of data collection: a functional analysis, a behavioral observation technique and an organizational culture assessment. The functional analysis provides a description of the work flow, behavioral observation identifies patterns of communication, and the culture assessment describes the environment of the organization. Two demonstration studies, one at a fossil power plant and one at a nuclear power plant, identified five organizational factors for further investigation: (1) communication, (2) organizational culture, (3) decision-making, (4) standardization of work processes, and (5) management attention, involvement and oversight.

Organizational culture in this work was described as "...

the beliefs, perceptions, and expectations that individuals have about the organization in which they work and about the values and consequences that will follow from one course of action or another. Consequently, culture highly influences behavior within the organization." 'Safety culture' is considered to be an element of organizational culture. Organizational culture in the demonstration studies was measured using employee questionnaires.

Jacobs et al. [24], adopted a similar viewpoint based on organizational processes. The paper identifies five organizational factors as relevant to safe operations: culture, administrative knowledge, communications, decision-making, and human resource allocation. This is similar, but not identical, to the list developed by Haber et al. In addition to identifying the five factors, Jacobs assigns several dimensions to each as shown in the second column of Table 3. A somewhat later joint paper by Jacobs and Haber [25] uses Jacob's list of organizational factors and dimensions.

Table 3 lists the organizational factors chosen for investigation in the organizational analysis approach described earlier [22], and those proposed by Jacobs and Haber. A comparison of the two columns shows the emphasis on structure and conditions in the first column and the emphasis on process in the second column. Both approaches are designed to relate management and organizational factors to safety performance. The organizational analysis approach was designed to rely as much as possible on publically available records. The organizational process approach, on the other hand, depends heavily on inferring organizational characteristics from surveys and interviews of a broad spectrum of personnel in the organization. It attempts to determine how an organization works, as opposed to how it is structured.

6.5. Work process analysis

Davoudian et al. [26] have proposed an approach to modeling the organization that uses elements of both organizational structure and organizational process. The ultimate goal is to develop a methodology for incorporating organizational factors into PRAs. This work has evolved over the last few years [11,27]. The analysis begins with asking the question "how is the organization supposed to work?" and then addressing "how well is it working?" The categories and dimensions of important organizational factors as articulated by Jacobs and Haber [25] are adopted. An examination of work processes is then proposed as a way to analyze and possibly quantify the importance of those factors.

The work process analysis methodology starts with the observation that the structure of an organization is determined by two basic elements: division of labor and coordination of effort. Division of labor is accomplished by creating work units, typically based on functional specialization. Examples are operations, maintenance, instrumentation and control, and health physics. Coordination is accomplished by both formal and informal mechanisms, including policies, procedures, scheduled meetings and unscheduled meetings. Work processes within a functional unit tend to be standardized and controlled by written procedures. The objective of the work process analysis methodology is to identify the organizational factors that can impact the performance of particular tasks, and ultimately to quantify those impacts as changes in PRA parameters (failure rates, human error probabilities or system unavailabilities).

The first step in the work process analysis model (WPAM) is the identification of front-line and supporting work processes. Front-line process are those that have a direct influence on the operability of plant hardware, such as plant operation, maintenance, and modifications. Supporting work processes include training, procurement and quality control. For each work process, the following basic question is posed: how can an accumulation of organization failures lead to an unsafe plant condition?

Each task in a given work process can be influenced by several organizational factors. In fact, one of the strengths suggested for this approach is its ability to identify organizational deficiencies that could disable dissimilar components. If the analysis is to be extended to quantification of the impacts on human error rates or system unavailability, it is necessary to rank the organizational factors according to their degree of influence on each task. One method of performing the ranking is the analytical hierarchy process. This involves assigning relative weights to each pair of pertinent factors (pairwise comparison). Presumably other ranking methods could be used.

In a 1999 paper [28], Weil and Apostolakis propose that the 20 organizational factors identified by Jacobs and Haber [25] can be reduced to six without impairing the effectiveness of the work process analysis methodology. The six factors retained are: communications, formalization, goal prioritization, problem identification, roles and responsibilities, and technical knowledge. These six were chosen by identifying factors that affected a large number of tasks and/or were often cited as contributing to errors, and by excluding factors that logically could be combined into one of the remaining factors.

6.6. A model based on expert elicitation

The Swedish Nuclear Power Inspectorate (SKi) has sponsored a study to develop a risk based performance monitoring system for nuclear power plants using expert elicitation to identify organizational and operational-based safety related performance indicators [29]. The model is based on a probabilistic safety assessment of the plant. Starting with a proposed list of 78 performance indicators, a final list of five high worth indicators is derived. The five indicators are:

- annual rate of safety significant errors,

- annual rate of maintenance problems,
- ratio of corrective to preventive maintenance on safety equipment,
- annual rate of problems with repeated root cause, and
- annual rate of plant changes that are not incorporated into design-basis documents prior to the next outage.

These indicators are proposed as a suitable measure of safety culture. Ultimately, the assessment of safety culture (superior, above average, average, below average, or inferior) can be used to modify equipment failure rates or system unavailabilities.

The SKi process is particularly interesting because it replaces virtually all of the activities represented in Fig. 1 with expert elicitation. Step 1 is represented by the initially proposed list of 78 performance indicators. Steps 2–5, are replaced by the expert elicitation process, with the output being the final list of five high worth indicators. Since the methodology includes an algorithm for quantifying the impact of the performance indicators on risk metrics, it provides a means of addressing the upper path of Fig. 2.

6.7. A summary of the empirical evidence

There is a substantial body of literature dealing with the relationship between safety culture and safety of operations. That literature is fragmented, however, and it is often difficult to understand how one piece of work relates to another, if at all. The scope, depth, terminology and perspective vary widely from one study to the next.

The first source of difficulty is terminology. There is general agreement on the concept of safety culture, and some agreement on its attributes. Many of the studies relating management and organization factors to safety of operations do not use the term 'safety culture.' If it is used, it may denote a narrowly defined element of a larger set of management and organization factors being investigated. One study can only be compared with another by looking at the organizational attributes that are actually measured. The study of safety culture might benefit substantially if a consensus could be developed on its definition, and, most importantly, its measurable attributes.

A second source of difficulty is the availability of suitable safety metrics. The chemical processing and transportation industries have sufficiently high occurrences of unwanted events that it is possible to correlate management and organization factors with accident rates. Other activities, including nuclear power generation, have sufficiently low accident rates that the accident rates provide no basis for comparing one facility to another. Instead, investigators select performance indicators, such as the number of unplanned scrams, as surrogates for safety performance.

Olson et al. [30] illustrate the issue by distinguishing among three categories of information: (1) plant performance indicators, (2) penultimate measures of safety, and (3) ultimate measures of plant safety. They suggest that the ultimate measures of plant safety are the unwanted events: core melt, large releases of radionuclides, and large population exposures. The penultimate measures of safety are potentially significant events, releases of radionuclides, and personnel exposures. (Analyses of potentially significant events to determine conditional core damage probability or conditional large early release frequency can partially bridge the gap between the penultimate and ultimate measures of safety.) Plant performance indicators might include the number of LERs, operating and outage data, and the number of violations of NRC regulations. The use of performance indicators as a measure of safety should include establishing a relationship between the indicator and the likelihood of an unwanted event. Current NRC work to identify risk based performance indicators is intended to address this issue [31].

No studies relating safety culture and safety of operations were identified which addressed all of the activities outlined in Figs. 1 and 2. Studies of the chemical process industry addressed all of the activities in Fig. 1, and provided empirical evidence that safety attitudes have a positive relationship to safety of operations. Those studies did not address identifying performance indicators (Fig. 2).

Studies of nuclear power plants focused on identifying management and organization factors important to safety of operations, but they lack the extensive field data collected in the chemical process industry studies. The work started at PNL by Osborn et al. [20], involved extensive empirical analyses relating organization factors to performance indicators, but did not examine attributes of safety culture. The work begun at Brookhaven National Laboratory by Haber et al. [23], did address organizational factors related to safety culture, but data collection and analysis concentrated on measuring those attributes and validating the measurements. Data collection appears to have been limited to one fossil and two nuclear power plants, and very little was reported on establishing an empirical relationship between the organizational factors and indicators of safety performance.

Overall, substantial work has been done to validate the idea that safety culture and other management and organization factors have a strong relationship to safety of operations. Most of the empirical work has been done outside the nuclear industry. Some investigators believe that results cannot be extrapolated from one industry to another without justification that does not now exist [32]. It appears that Lee [3] has characterized the situation correctly: "There has been little direct research on the organizational factors that make for a good safety culture. However, there is an extensive literature if we make the *indirect* assumption that a relatively low accident plant must have a relatively good safety culture" (emphasis in original). The proponents of safety culture as a determinant of operational safety in the nuclear power industry are relying, at least to some degree, on that indirect assumption.

7. Regulatory perspectives

Regulatory organizations have an interest in safety culture because it is now widely believed that there is a relationship between safety culture and safety of operations. The most obvious link suggested by work done to date is that a good safety culture is expected to reduce human error rates. Reason [12] suggests that well-defended technologies, those that use several layers of defense in depth such as nuclear power plants, may be especially vulnerable to an unsafe culture. He points out that the effect of a poor safety culture is to create gaps or holes in the defenses, which are not readily apparent (latent errors), thus making the system vulnerable to a serious accident when the right initiating event occurs. Defenses in depth make the system more opaque to the operators, and the operators are more remote from the processes they are controlling. An important question remains as to whether safety culture should be addressed by the regulatory process. That question probably cannot be answered without considering *how* it might be addressed by the regulatory process.

INSAG [5] asserts that safety culture is attitudinal as well as structural, and that it relates to both organizations and individuals. Lee [3] suggests that the safety improvements to be achieved through engineering are limited, and that additional improvements require addressing the "hearts and minds of the management and workers." Studies sponsored by the NRC [33,34] have shown a positive correlation between management and organizational factors and selected safety indicators. Studies outside of the nuclear industry [19,35] have shown strong positive correlations between organizational characteristics associated with safety culture and low accident rates.

Although there is no universally accepted definition, there is some common ground among investigators on the elements of safety culture. Most investigators appear to agree that the elements include good communications, organizational learning, senior management commitment to safety, and a working environment that rewards identifying safety issues. Some investigators would also include management and organizational factors such as a participative management leadership style. The regulatory dilemma is that the elements important to safety culture are difficult, if not impossible, to separate from the management of the organization.

Historically, the NRC has been reluctant to regulate management functions in any direct way. Licensees have been even more reluctant to permit any moves in that direction. The argument is, of course, that licensees are responsible for the safe operation of their facilities, and they must be permitted to achieve safety in their own operating environment in the best ways they know. The closest NRC has come to evaluating management performance is the SALP program, which the agency has discontinued. Throughout its life it was criticized by licensees as lacking objectivity.

7.1. Safety culture as a basis for safety regulation

One of the most comprehensive reviews of the relationship between safety culture and safety of operations was undertaken for the United Kingdom Health and Safety Executive by its Advisory Committee for the Safety of Nuclear Installations (ACSNI) [36].

ACSNI concludes, from USNRC sponsored work, that the key predictive indicators of safety performance are effective communication, good organizational learning, organizational focus on safety, and external factors such as the financial health of the organization and the impact of regulatory bodies. It holds that, "The best safety standards can arguably only be achieved by a programme which has a scope well beyond the traditional pattern of safety management functions." It characterizes the evolution of safety regulation as follows:

> "There are three phases in the history of attempts to regulate general industrial safety.
> First, there is a stage of concentration on the outcome; if a worker or a member of the public is harmed, those considered responsible are punished.
> Second, there is a stage of prescribing in advance the detailed action that industry must take. For example the organisation must provide guards of certain types for specific machines.... This stage is an advance because it attacks points of danger before actual harm occurs....
> In the third stage, industry is canvassed to develop a 'safety culture'... This stage of regulation... concentrates on the internal climate and organization of the system [and] also emphasizes the need for every individual to 'own' the actions being taken to improve safety...."

In examining the regulator's role in influencing licensee organizational behavior, the ACSNI human factors study group maintained that, "The behavior of the regulators will affect the culture of the licensees.... The regulators need to act in such a way as to encourage 'ownership' of safety by the whole staff of the licensee [36, p. 47]."

A theme that runs through the ACSNI study is that the most effective safety cultures will develop in less prescriptive regulatory structures. "The most impressive achievements appear in companies where the pressure for safety has been generated from within the organization, apparently independent of external standards [36, p.16]."

A subsequent report [37] notes that "It is recognized that there are a number of prescriptive regimes, such as the U.S. Nuclear Industry, where the encouragement of a positive safety culture is still essential. It is considered that those Operators with good Safety Cultures, within the US regulatory regime, tend to self-regulate around the constraints of the regulatory regime, to attain levels of safety which are beyond those minima specified in the regulations. The manner in which the Regulator can encourage such self-regulation is not clear [37, p. 34]."

One aspect of this idea is explored in some detail in an earlier paper by Marcus [38], in which he examines the implementation of certain NRC requirements at several U.S. nuclear power plants. His conclusion was that, "...nuclear power plants with relatively poor safety records tended to respond in a rule-bound manner that perpetuated their poor safety performance and that nuclear power plants whose safety records were relatively strong tended to retain their autonomy, a response that reinforced their strong safety performance."

7.2. International activities

The IAEA has continued to develop the concept of safety culture as an important contributor to safety of operations, and therefore as an important issue to be addressed by the regulatory process. A 1998 publication is devoted to offering "...practical advice to assist in the development, improvement or evaluation of a progressive safety culture [39]." A revision to INSAG-3 was issued in 1999 to provide, among other things, "A more comprehensive treatment of safety culture and defense in depth [40]." INSAG-13 [41] was also issued in 1999 to "...build upon the ideas outlined in 75-INSAG-4 [Safety Culture] and to develop a set of universal features for an effective safety management system in order to develop a common understanding."

The Nuclear Energy Agency (NEA) has also become engaged in promoting safety culture as an important part of safety regulation. A 1999 publication, "The Role of the Nuclear Regulator in Promoting and Evaluating Safety Culture [42]," suggests signs that a regulator should look for to determine the strength of a licensee's safety culture. It also provides suggestions for regulatory response to a weakening safety culture, although the suggestions are very general. A subsequent report deals more specifically with the issue of regulatory response [43].

NEA has also issued a 'state-of-the-art' report [44] on the identification and assessment of organizational factors related to nuclear power plant safety. Volume 1 lists and discusses 12 organizational factors:

- external influences,
- goals and strategies,
- management functions and overview,
- resource allocation,
- human resources management,
- training,
- co-ordination of work,
- organizational knowledge,
- proceduralization,
- organizational culture,
- organizational learning,
- communication.

This list is similar, but not identical, to the list of attributes proposed by Jacobs and Haber (Table 3, second column).

The second volume of the report summarizes the regulatory framework used in nine OECD countries, including France, the United Kingdom and the United States. In each case the discussion addresses how the regulatory process considers management and organization factors. Most of the regulatory programs discussed include some evaluation of management and organization factors. By contrast, the NRC program does not involve direct evaluation of management performance.

Volume 2 of the NEA report [44] also provides summaries of research programs on management and organization factors. The programs described are directed at identifying management and organizational factors important to safety of operations, incorporating management and organization factors into PRAs, or evaluating the attributes of safety culture within a licensee's organization.

7.3. Safety culture and NRC's regulatory process

The Commission Policy Statement on the Conduct of Nuclear Power Plant Operations [45], makes 'safety culture' part of the NRC's regulatory agenda. Issued in 1989, the policy statement includes the provision that "Management has a duty and obligation to foster the development of a 'safety culture' at each facility and to provide a professional working environment, in the control room and throughout the facility, that assures safe operations." The policy statement then defines safety culture using the definition from INSAG-4.

Current NRC programs to develop risk informed regulatory processes and performance based reactor oversight appear to be in consonance with the idea of some degree of self-regulation. The reactor oversight program [46] identifies a level of performance, as measured by a set of performance indicators, where regulatory involvement will be limited to a baseline inspection program. The program identifies seven cornerstones of safety performance, each monitored by one or more performance indicators. The four cornerstones for reactor safety are initiating events, mitigating systems, barrier integrity and emergency preparedness. In addition to the cornerstones, the staff has identified three 'cross-cutting' elements that are part of each cornerstone. These are human performance, management attention to safety and worker's ability to raise safety issues (safety-conscious work environment), and finding and fixing problems (corrective action programs). There are currently no performance indicators associated with these cross-cutting issues.

The staff recognizes that the new oversight program will involve a shift in the NRC role from improving human reliability to monitoring human reliability. This appears to be consistent with the thought of allowing more of what might be termed 'self-regulation.' On the other hand, the staff equates the term 'safety culture' with 'safety conscious

work environment.' This appears to be a much narrower concept of safety culture than is used by most writers in the organizational safety field.

Two questions are suggested here. The first is whether the NRC is giving sufficient attention to the staff skills, knowledge and abilities that will be required in a risk-informed, performance based regulatory scheme. If the NRC is to encourage safety culture, it may require a different perspective on the part of the front-line inspection staff. The second question is whether appropriate attention is being given to identifying performance indicators for human performance, safety culture, or other relevant management and organizational factors.

The ACSNI study group [36] concluded that research is required particularly in two areas. "Firstly, work is necessary simply to increase the number of validated culture and performance indicators available. Secondly, studies are required to establish the extent to which the indicators remain valid once they have been identified and used as indicators."

8. Conclusions

There is a clear consensus among writers in the field of safety management that worker attitudes toward safety make a difference. What is not clear is the mechanism by which attitudes, or safety culture, affect the safety of operations. Statistical evidence that unambiguously links safety culture or specific attributes of safety culture with the safety of operations is surprisingly rare, especially within the nuclear industry.

Pidgeon [47] examines the key theoretical issues underlying the concept of safety culture. He notes that, "...some 10 years on from Chernobyl, the existing empirical attempts to study safety culture and its relationship to organizational outcomes have remained unsystematic, fragmented, and in particular underspecified in theoretical terms." He identifies four theoretical issues that must be addressed if the concept of safety culture is to realize its promise. The first is the paradox that culture can act simultaneously as a precondition for safe operations and an incubator for hazards. The second issue is that in complex and ill-structured risk situations, decision makers are faced not only with the matter of risk, but with fundamental uncertainty characterized by incompleteness of knowledge. The third issue is the organizational construction of acceptable risk. The fourth is the issue of organizational learning and the political need to assign blame for disasters. Pidgeon's paper stresses the importance of safety culture as a concept uniquely capable of improving safety in complex systems.

From the narrow perspective of the nuclear power industry, an important next step in understanding the relationship among safety culture, safety of operations and safety regulation would be to develop consensus on the essential attributes of safety culture and to identify suitable performance indicators. Consensus may not be easily reached, but investigators seem to have made too little use of past work, and constructed new frameworks rather than building on what has been done. Performance indicators will be even more difficult. Some work is underway to determine the degree to which the performance indicators in the reactor oversight program capture human performance issues. The results of that work might provide some insights into how performance indicators could be developed.

The NRC regulatory program must assure that licensee's root-cause analyses and corrective action programs are capable of identifying safety culture issues. Models for human performance, such as ATHEANA [15], will not be realistic until the influence of the plant's safety culture on the 'error-forcing context' is assessed [48].

Ultimately, the NRC will have to arrive at an understanding of how its regulatory process can affect the safety cultures of its licensees, both positively and negatively. The role of the regulator needs to be determined, including the possibility that there is no role other than monitoring.

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RISK-INFORMED LICENSING FOR ADVANCED REACTORS

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ABSTRACT

Recent developments in the United States nuclear industry include pre-application discussions between potential licensees and the U.S. Nuclear Regulatory Commission (NRC) for new plants, some based on non-light water reactor technology. One of the major challenges to the successful deployment of new nuclear plants in the United States is the regulatory process, which is largely based on water-reactor technology. With the initiation of pre-application discussions for new plant designs, the industry and the NRC are addressing the applicability of the current set of regulatory requirements to the proposed new designs. One outcome of the focus on proposed new plants is the development of new licensing approaches. The current efforts for existing plants and proposed new plants focus on adapting the current set of requirements and standards for near-term application. Beyond these current efforts, our team has been pursuing the application of a more aggressive risk-informed approach to all regulatory requirements and industry standards, as well as to the regulatory process, focusing upon those issues that affect the design and licensing of new nuclear power plants. We have extended the previous development of our framework and have been investigating approaches for defining quantitative risk criteria that have the potential to provide a consistent basis for regulatory decisions, independent of the reactor concept, within our framework.

KEYWORDS

Advanced nuclear systems, risk-informed regulation, new nuclear reactors

BACKGROUND

Recent developments in the United States nuclear industry include pre-application discussions between potential licensees and the U.S. Nuclear Regulatory Commission (NRC) for new plants, some based on non-light water reactor technology. One of the major challenges to the successful deployment of new nuclear plants in the United States is the regulatory process, which is largely based on water-reactor technology. Furthermore, a growing awareness within government and industry is that many of the

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current regulatory requirements and industry standards are not contributing significantly to safety and, therefore, have needlessly driven the costs of nuclear plants into a range that will not be economically competitive in the deregulated U.S. power industry. Moreover, the overly prescriptive nature of these requirements and standards inhibits the introduction of new, more advanced technologies. To address this concern, the U.S. nuclear industry and the NRC have been working to apply risk-informed regulation to address requirements that affect the operation and maintenance of existing nuclear plants (U.S. NRC, 1998; 1999; 2000). With the initiation of pre-application discussions for new plant designs, the industry and the NRC are also addressing the applicability of the current set of regulatory requirements to the proposed new designs. One outcome of the focus on proposed new plants is the development of new licensing approaches. The current efforts for existing plants and proposed new plants focus on adapting the current set of requirements and standards for near-term application. Beyond these current efforts, our team has been pursuing the application of a more aggressive risk-informed approach to all regulatory requirements and industry standards, as well as to the regulatory process, focusing upon those issues that affect the design and licensing of new nuclear power plants (Durán et al., 2000; Apostolakis et al., 2001).

Our approach is based on the rationalist process for defense in depth and a full application of probabilistic risk assessment (PRA) to identify systematically the regulatory requirements and industry standards that truly add to safety and reliability. The rationalist process for defense in depth uses PRA techniques to quantify uncertainty and to explicitly account for defense in depth features in reducing uncertainties to acceptable levels. The rationalist relies on PRA methods to provide an integrated and systematic analysis of the plant that explicitly addresses sources of uncertainty. The process envisioned by the rationalist is: (1) establish quantitative safety goals, such as health objectives, core damage frequency, and large release frequency; (2) design and analyze the plant using PRA methods to establish that the safety goals are met; (3) evaluate the uncertainties in the analysis, including those due to model inadequacies, system performance and reliability, and lack of knowledge; and (4) determine what steps (i.e., defense in depth measures, new design features) to take to address those uncertainties (Sorenson et al., 1999). The result of this approach was the development of a framework for risk-informed regulation and design for new nuclear power plants (Durán et al., 2000). To develop risk-informed regulations, implementation of the framework is achieved by defining functional system characteristics, within the context of how PRA is performed, to determine what areas need to be regulated to assure safety. Implementation for design is achieved by specifying design configurations and using PRA to evaluate the design, then iterating with subsequent design changes to achieve the desired level of safety and reliability. A master logic diagram (MLD) was developed to be used in taking a top-down approach to identify the safety functions, and systems, structures, and components (SSCs) that are required to maintain safety and to identify the accident initiators and system response failures that could compromise safety.

Our initial regulatory framework, however, was based on light-water reactor technology. Our more recent efforts have extended this framework for more general application to other advanced reactor concepts. A key feature of this framework is that it depends on establishing quantitative metrics, and then performing formal analyses, including an assessment of uncertainties, as far as the analytical methodology permits and evaluating the results of the analyses to the quantitative metrics. One issue in extending the framework has been that current quantitative criteria such as core damage frequency may not be applicable to advanced reactor concepts that are not water-based. Therefore, we have been investigating approaches for defining quantitative risk criteria that have the potential to provide a consistent basis for regulatory decisions, independent of the reactor concept, within our framework.

This paper will present an overview of recent developments in our framework for risk-informed regulation and design of advanced nuclear power plants. We will present an updated version of the MLD, revised and extended for more general application to any advanced reactor concept.

Additionally, we will discuss the development and application of quantitative risk criteria as a basis for consistent reactor safety regulation, independent of reactor concept.

EXTENDING THE FRAMEWORK FOR OTHER REACTOR CONCEPTS

Implementation of the initial version of the framework for regulation was achieved by defining functional system characteristics, within the context of how PRA is performed, to determine what areas need to be regulated to assure safety. Our initial efforts defined the functional system characteristics based on light-water reactor technology (Durán, 2000). To extend the framework for more general application to other reactor concepts, we focused on defining the functional system characteristics in more generic terms, for example a reactivity increase or decrease. Figure 1 provides the extended MLD. Additionally, the extended MLD for the framework includes explicit consideration of all operational modes and categorization of internal and external initiating events by frequency of occurrence (high, moderate, low). These features demonstrate how important considerations of licensing decisions that are addressed by a PRA can be integrated into the framework in a coherent manner. To exercise the extended MLD for a non-light-water reactor concept, additional efforts, discussed in Apostolakis et al. (2001), have applied this generic MLD for the pebble bed modular reactor (PBMR).

QUANTITATIVE CRITERIA FOR LICENSING

In the initial development of the framework, existing quantitative goals were used as a starting point. Established quantitative health objectives (QHOs) and related subsidiary goals (Seale, 1998) have been to guide the development of risk-based regulation and design. The intent has been to investigate approaches to developing regulatory requirements in such a way that compliance will provide reasonable assurance of meeting specific quantitative goals and the goal of protection of the public.

To delineate quantitative goals, the PRA strategies of the framework must be expressed using quantifiable measures of risk. One method of assessing the level of protection against accidents at a given nuclear power plant is simply to compare PRA results to the QHOs for early fatality and latent cancer risks:

- the risk of an early fatality as a result of a plant accident should be less than 5×10^{-7}/year for members of the public located within 1 mile of the exclusion area boundary, and
- the risk of dying from cancer as a result of a plant accident should be less than 2×10^{-6}/year for members of the public residing within 10 miles of the plant.

For new plants, QHOs and related subsidiary goals set forth in this discussion apply to mean risk measures quantified in full-scope PRAs. Unfortunately, the QHOs are difficult to apply for the purposes of risk-informed licensing. Simply replacing existing regulations with the QHOs would be a completely rationalist approach and would not consider limitations and uncertainties inherent in PRA. As such, it is appropriate to investigate allocation of risk by establishing subsidiary goals. The quantitative goals discussed here provide targets for the initial framework development. Compliance with regulation and design requirements developed by implementing the framework would provide a reasonable expectation that the quantitative goals will be met.

The QHOs are the highest level quantitative goals. The QHOs were originally set as a measure of "safe enough," and in that sense they go beyond adequate protection. Given this position of the Commission, no risk arguments exist for setting quantitative goals more stringent than the QHOs. While there is no basis for being more stringent than the QHOs, the limitations and uncertainties

inherent in PRA, which tend to grow as postulated accidents proceed in time, influence the quantitative allocation among the three PRA strategies originally established for the framework.

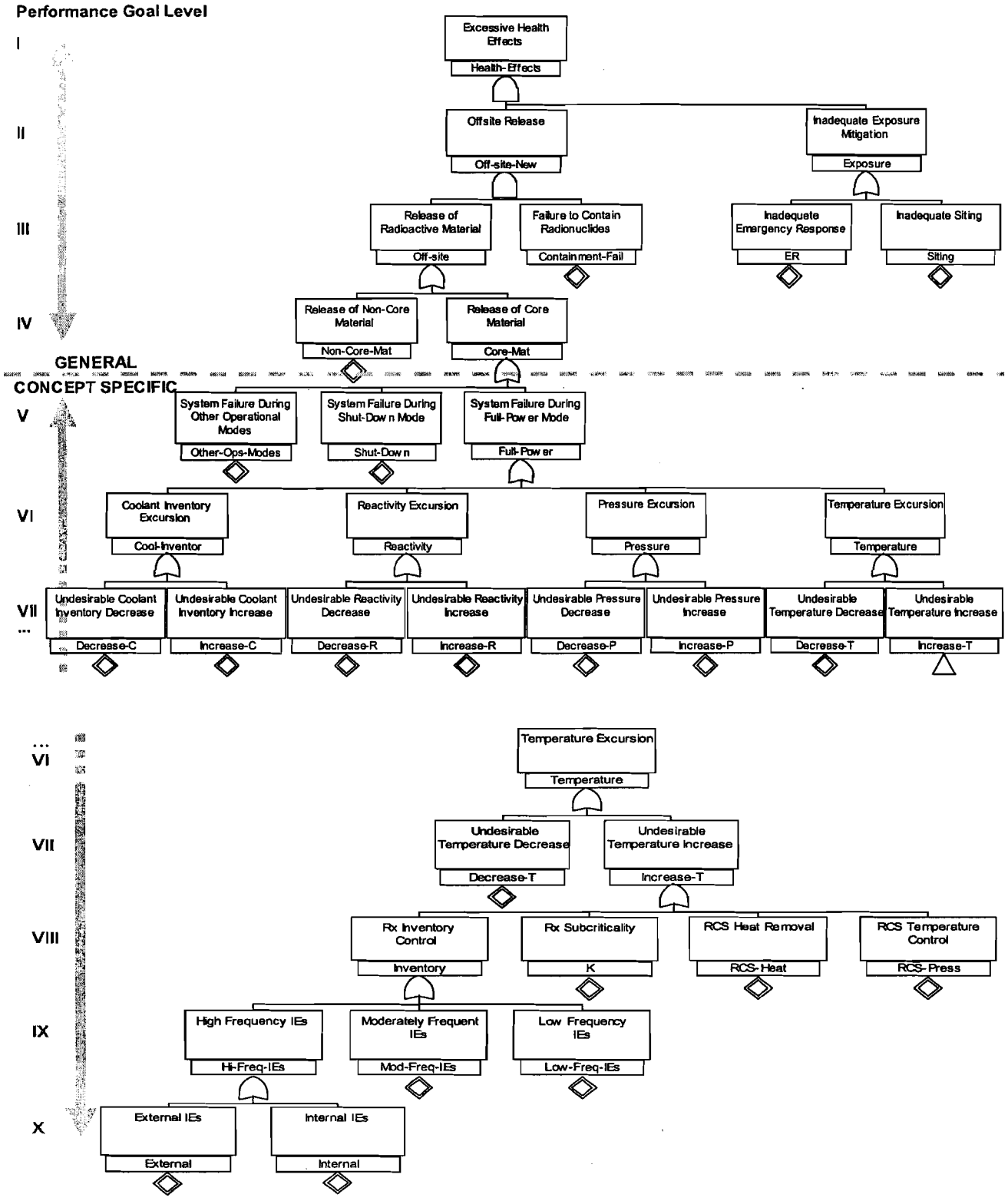

Figure 1: Generic master logic diagram, extended for application to any reactor concept.

Because public risks are dominated by accidents that involve core damage and containment failure, subsidiary goals based on other risk measures associated with the calculations for Level 1, Level 2, and Level 3 PRAs were developed for the framework. The subsidiary goals are consistent with the QHOs. A summary of the proposed goals is provided below:

(1) For the strategy to limit core damage frequency (Level 1 PRA):
- the probability of core damage should be less than 10^{-4}

(2) For the strategy to mitigate releases of radionuclides (Level 2 PRA):
- the conditional probability of a large release (either early or late) should be less than 0.1

(3) For the strategy to mitigate consequences (Level 3 PRA):
- the conditional probability of an early fatality for an individual should be less than 0.1
- the conditional probability of a latent cancer for an individual should be less than 0.1

These are consistent with NRC's efforts on risk-forming current reactor regulations (U.S. NRC, 2000). Yet, these goals may not be applicable to all reactor types. For example, the coated fuel pebbles in a PBMR are designed to retain fission products in normal and accident conditions; hence, the goal of limiting core damage frequency may not be directly applicable for the PBMR. Furthermore, according to its designers, the fuel can never get hot enough to melt even in accident conditions; hence, the goal of limiting large release frequency may not be applicable to the PBMR.

Ideally, any subsidiary goals developed to support the QHOs will be applicable to any reactor type. Further, Kress (2002) identifies the need for international consensus on the technical basis to be used to develop consistent risk criteria so that the actual risk status of each plant will be quantified in a way that can be easily communicated to the public and that will satisfy broadly accepted risk limits on an individual plant basis.

One type of subsidiary goal that can be applied to any reactor type is expressed as a frequency vs. consequence diagram. This idea was first proposed by Farmer (1967) and has been expanded on by others over the years (Ballard, 1993). Farmer (1967) proposed performing multiple safety evaluations, producing "a spectrum of events with associated probabilities and associated consequences." In Farmer's example, the reactor-years between accidental releases of ^{131}I was plotted against Curies of ^{131}I released. He pointed out that, on this type of a plot, "all parallel lines of equal slope –1 join points of equal risk in terms of curies released per year. One such line might be used as a safety criterion by defining an upper boundary of permissible probability for all fault consequences" (1967).

In the years since Farmer made these suggestions, this approach has been further developed. It is now much more common to plot "F(C) against C, where F(C) is the frequency of events with consequences greater than or equal to C" (Ballard, 1993). Also, consequences are more commonly expressed in terms of harm to humans (e.g., fatalities, latent cancers), although calculating such consequences introduces significant sources of uncertainty such as weather, population distributions, and effectiveness of emergency measures. However, the consequences of interest can also be selected to provide subsidiary performance measures that support the QHOs or other high level criteria. For example, the consequence could be Curies of particular radionuclides released (e.g., ^{131}I, ^{90}Sr, ^{137}Cs, noble gases, alpha-emitters with a half-life greater than 5 years, etc.), or percentage release of particular radionuclides, or dose to a worker or a member of the public. Regulatory probabilistic safety studies conducted in Switzerland (Schmocker et al., 1996) use both Cs release fractions and sum of fractions of total core inventory which is released for each release class as consequences. Any curve proposed as a safety goal that limited the probability of certain consequences would have to be constructed very carefully so as to accurately support the QHOs or other high level criteria. The use of

frequency vs. consequence diagrams to set safety goals that are applicable to all reactor types continues to be examined as this framework is developed further.

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(6)

Risk-Informed Regulation Implementation Plan

Presentation to ACRS

July 10, 2002

Overview of Presentation

- Introduction & 10 CFR 50.46 Rulemaking Activities
  - Mark Cunningham, RES
- Plan for Improving Coherence Among Risk-Informed Activities
  - Michael Johnson, NRR
- Risk-Management Technical Specifications
  - William Beckner, NRR
- Upcoming Items of Interest
  - Chris Grimes, NRR & Mark Cunningham, RES

2

Introduction & 10 CFR 50.46 Rulemaking Activities

Mark Cunningham, RES

Overview of June 2002 RIRIP

- Summary of Upcoming Activities - Reactor Safety Arena
  - Special Treatment Requirements (10 CFR 50.69)
  - Risk-Informing Part 50 (10 CFR 50.44 & 50.46)
  - Fire Protection Rule Revision
  - Improve Coherence Among Risk-Informed Activities
  - Risk-Informed Environment Initiative
  - Risk-Management Technical Specification Initiatives
  - Develop Regulatory Guide and Standard Review Plan to assess PRA adequacy
  - Pressurized Thermal Shock

4

Overview of June 2002 RIRIP

- Summary of Upcoming Activities - Waste Safety and Materials Safety Arenas
  - Amending Regulations – medical use of byproduct material
  - High-Level radioactive waste disposal at Yucca Mountain
  - Revision of Inspection Manual
  - Decommissioning policy and guidance documents

5

Overview of June 2002 RIRIP

- New Activities
  - Waste Safety and Materials Safety Arenas
    - Develop Guide for Performing Risk Analyses
    - Develop Safety Goals
    - Evaluate Low-Level Source Material (thorium and/or uranium)
    - Amend Part 63
    - Cross Cutting Risk Assessment of Spent Fuel Management
- Accomplishments
- More Detailed Descriptions of activities and milestones

6

Identifying possible changes to NRC's reactor safety requirements

Changing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"

- Suggested changes to 10 CFR 50.46
  - Replace current requirements with a performance-based requirement
  - Revise requirements for ECCS evaluation model
    - Allow for more realistic analyses
  - Propose a risk-informed voluntary alternative to 10 CFR 50.46 for ECCS reliability

7

Recent and Upcoming Activities

- Evaluation of plant-specific approach for reliability evaluation (April 2002)
- Evaluation of changes to acceptance criterion and evaluation model (June 2002)
- Evaluation of generic approach for reliability evaluation (July 2002)

8

PLAN FOR IMPROVING COHERENCE AMONG RISK-INFORMED ACTIVITIES

Michael Johnson, NRR
Stu Magruder, NRR
Mary Drouin, RES

Background

- SRM: for current reactors –
  - "provide its plan for moving forward with risk-informed regulation to address regulatory structure convergence with our risk-informed processes"
- Stakeholders believe we are inconsistent
- NRC staff is often frustrated

10

Goals

- Develop a common understanding of risk-informed regulatory objectives
- Staff and stakeholder buy-in

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Need to Develop Plan That Will:

- Utilize ongoing efforts
- Identify goals and associated end products
- Identify approach for achieving goals
- Approach will identify:
  - Inefficiencies
  - Unnecessary regulatory burden
  - Safety concerns
  - Interface with advanced reactors

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Next Steps

- Outline in RIRIP, June 2002
- Briefing for Commission TAs
- Interact with stakeholders to solicit input
  - Public meetings/workshop
  - ACRS
- Develop detailed plan

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Risk-Management Technical Specifications

William Beckner, NRR

Overview

- Staff and industry briefed ACRS 4/28/00
- Eight initiatives developed by owners groups through NEI task force
- Major objective:
  - Bring 10 CFR 50.65(a)(4) risk assessment process into configuration management under technical specifications

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Status Highlights

- Initiative 2 – Missed Surveillance
  - Manage risk of missed surveillance
  - Offered for adoption 9/28/01
  - 34 applications for 57 units
- Initiative 3 – Mode Change Flexibility
  - Permit upward mode change
  - Model safety evaluation nearing completion

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Status Highlights (cont.)

- Initiative 1 – TS Actions End States
  - Permit repair in hot shutdown
  - Safety evaluations complete (CEOG, BWROG)
- Initiative 4 – Completion Times From Configuration Risk Management Program
  - Major reliance on 50.65(a)(4) to control completion times
  - Initial concept developed

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Initiatives

1. TS Actions End States
2. Missed Surveillance
3. Mode Change Flexibility
4. Completion Times from CRMP
5. STI Methodology
6. LCO 3.0.3 Actions and Times
7. Non-TS Support Features
8. Scope of TS

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Upcoming Items of Interest

Chris Grimes, NRR
Mark Cunningham, RES

Upcoming Items of Interest

- 10 CFR 50.69 (Special Treatment)
- RG on PRA Standards & Industry Guidance
- Draft Guidance on Performance-Based Regulation
- Advanced Reactors

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THE ENERGY DAILY Wednesday, July 10, 2002 1

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Wednesday, July 10, 2002 ED Volume 30, Number 131

Senate Overrides Nevada Protest

Yucca Mountain Gets Final Nod

BY JEFF BEATTIE

After 20 years of debate, the Senate Tuesday decisively approved plans to bury the nation's high-level nuclear waste beneath Yucca Mountain, crushing emotional objections from Nevada officials and marking an historic victory for the U.S. nuclear industry.

The Senate vote represented final congressional approval for Yucca Mountain, which was first picked by Congress in 1982 for geological study by the Energy Department to see if it was suitable for long-term nuclear waste disposal.

The House earlier this summer overrode Nevada's protests that DOE had failed to adequately assess the site, which the state contends will leak radioactivity into soil and groundwater.

The Senate vote means DOE now can file a construction and operation plan for the waste repository with the Nuclear Regulatory Commission, which must certify that the repository will meet Environmental Protection Agency limits on leakage over the thousands of years the wastes will remain highly radioactive. DOE expects to file its plan by 2004.

The multi-billion dollar project—located 90 miles northwest of Las Vegas—also must clear a growing pile of lawsuits filed by Nevada officials against DOE, EPA and NRC, among others.

Nonetheless, the Senate's decision gets the United States over a colossal "not-in-my-backyard" waste disposal problem that has confounded other nuclear-invested countries around the globe.

It also hands a thundering win to the Bush administration, which has lined up foursquare behind the highly

(Continued on page 2)

Big Names Power Up New Renewable Energy Association

BY GEORGE LOBSENZ

What do C. Boyden Gray, big industry lobbyist and ex-aide to former President Bush, and Amory Lovins, alternative energy guru, agree on?

The need for a new, big-bucks trade association that can "bring renewable energy into the mainstream of America's economy and lifestyle" and otherwise spread the gospel about solar, wind, hydro, geothermal, biomass, biofuels, waste energy and hydrogen energy systems.

Gray and Lovins are among a host of heavy-hitters on the advisory board helping to put together the American Council for Renewable Energy (ACRE), which opens a two-day organizing conference today at a Washington, D.C., hotel.

Others key figures on the advisory board are Richard Truly, head of the Energy Department's National Renewable Energy

(Continued on page 3)

Brownell Sees Phased Implementation Of Standard Market Design

BY TINA DAVIS

Standard market design may need to be phased in for some parts of the country, FERC Commissioner Nora Brownell said Tuesday, adding that there could be a date certain for full implementation of the market but in the meantime the agency wants to encourage regions to take interim steps towards competitive wholesale markets.

"You don't have to issue the rules and then wait until some date certain before you start to implement what's good for markets," Brownell said at a meeting on standard market design sponsored by *The Energy Daily* and Sullivan & Worcester.

She added that the perception the Federal Energy Regulatory Commission's rules for the market will be based on one single platform is wrong.

There were, she said, "some perceptions this design was based on one model. That is not the case."

In many respects, the FERC white paper on market design shares many elements with market rules adopted by PJM Interconnection LLC, the mid-Atlantic regional transmission organization (RTO) seen as one of the most advanced grid operators. But Brownell said

(Continued on page 4)

EnCana Selling Two Big Oil Pipelines

EnCana Corp. announced Tuesday it was selling a major pipeline system that brings Canadian crude oil to the Rocky Mountain and Midwest regions as well as its 70 percent interest in a pipeline serving Canada's rapidly developing oil sands deposits.

Calgary-based EnCana said it did not need the 1,717-mile Express pipeline system or the 297-mile Cold Lake pipeline to meet its own transportation needs and that it wanted to focus resources on exploration and production operations.

"After a strategic review of EnCana's assets, we have determined that we do not need to own these pipeline systems to assure ourselves of sufficient transportation capacity to meet EnCana's requirements," said Bill Oliver, president of EnCana's midstream and marketing division. "And given our rich portfolio of upstream growth opportunities, we look forward to re-deploying capital into core exploration and production initiatives that are more consistent with our strategic direction."

The Express pipeline system consists of two major pipelines. The 24-inch Express pipeline runs 785 miles from Alberta's oil transportation hub at Hardisty to Casper, Wyo. It delivers up to 172,000 barrels per day of Canadian crude to Montana, Wyoming and Utah. The 932-mile Platte pipeline can deliver up to 150,000 barrels per day of oil from Casper to Wood River, Ill., serving refineries in Colorado, Kansas and Illinois.

The 297-mile Cold Lake pipeline system has two legs. The west leg between the Cold Lake oil sands region and Edmonton, Alberta, consists of a 24-inch blend pipeline and a 12-inch diluent pipeline. The 24-inch line can deliver up to 235,000 barrels per day of blended bitumen to Edmonton and the 12-inch line moves condensate from Edmonton to the Cold Lake area. The south leg can deliver up to 200,000 barrels per day of blended oil from Cold Lake to Hardisty, Alberta, where it connects with Express and another pipeline.

Net book value of the pipelines, held through Alberta Energy Company Ltd., an indirect wholly owned subsidiary of EnCana, is $1.3 billion.

Final Nod For Yucca Mountian...

(Continued from page one)

controversial project to clear the way for new nuclear power generation.

Though there had been some question in recent weeks about whether the Senate would back Yucca, the final debate Tuesday was anti-climatic, with repository opponents indicating from the outset that they did not have the votes to stop the project.

However, the critics bitterly charged the Bush administration and other Yucca supporters were unwisely ramming the project through Congress.

"It may be years before we know if it is safe to store nuclear waste at Yucca Mountain," said Senate Majority Leader Tom Daschle (D-S.D.). "If you're thinking of voting for this proposal, next time your state may be where the special interest groups want to bury their radioactive trash.

"If we let them do it this time, without sufficient proof that it's safe, think how much easier it will be next time time."

And Assistant Majority Leader Harry Reid (D-Nev.), called Energy Department assertions that Yucca was safe "a great big crock of...chicken soup."

Yucca proponents, led by Sen. Frank Murkowski (R-Alaska), warned that failure to approve Yucca would leave the nation with no plan for handling dangerous nuclear waste.

"We may not be saying this is the end of the nuclear industry, but we certainly would be saying we have no resolution on how to manage its waste stream," said Sen. Larry Craig (R-Idaho).

And Sen. Jeff Bingaman (D-N.M.), chairman of the Energy and Natural Resources Committee, broke with many in his party to bluntly state that Yucca's time had come.

"Failure to approve the resolution terminates this nation's nuclear waste program. Either the waste will stay where it is...or Congress would have to pass a new law to ask the DOE to look for a new site," said Bingaman.

In the end, the issue was decided on a procedural vote on whether to proceed with Senate consideration of a resolution overriding Nevada's objection. After senators voted 60-39 to approve that motion, Yucca opponents agreed to a simple voice vote on the resolution itself.

Tuesday's vote marks a stunning reversal for a project that was, by most accounts, spinning its wheels badly just years ago.

The repository was originally supposed to open in 1998, and more than a dozen utilities have sued the DOE over failure to take spent nuclear fuel that has piled up alongside their reactors.

At the Yucca Mountain site itself, opponents say studies have produced mounting evidence that geology there is a poor barrier, forcing the Energy Department to rely on engineered barriers to a worrisome degree.

But the Bush administration—reversing a history of limp presidential support for the plan—began finalizing long-running safety analyses shortly after arriving in Washington. Citing those documents, President Bush in February declared his support for the site.

DOE says the repository could open by 2010 or shortly after, although most observers expect that date may slip.

Still, that places the United States alongside Finland, with only four reactors, or Sweden, home to 11, as the only western countries likely to build a permanent, underground home for high-level waste within the next two decades. Authorities in those countries hope to open repositories in 2020 and 2015, respectively.

Several other western countries have in recent years backed off repository plans, in part due to public concern.

494th ACRS Full Committee:

Advanced Reactor Research Plan

July 11, 2002

John H. Flack, REAHFB
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

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Outline

- Introduction

- Discussion of Technical Areas:

  - ☐ Framework.....Mary Drouin, PRAB
  - ☐ Fuels Analysis.....Stuart Rubin, REAHFB
  - ☐ Materials Analysis.....Joseph Muscara, MEB
  - ☐ Reactor Systems Analysis...Donald Carlson, REAHFB
    Richard Lee, SMSAB

- Summary and Future Plans

Introduction

- Primary focus: non-LWR research infrastructure
- Next update will include:
  - ACR-700
  - ESBWR, SWR-1000
  - Generation IV
- NRC research vs. applicant's responsibility
- Key collaborations (DOE, National, International)
- Prioritization (PIRT, PBPM)

NRC Advanced Reactor Research Plan (Structure)

Structured around 9 Key Research Areas:

1. **Framework*** (tools)
2. Accident Analysis (PRA, human factors, instrumentation & control)
3. **Reactor Systems Analysis*** (T/H, nuclear analysis, severe accident analysis)
4. **Fuels Analysis*** (fabrication and performance)
5. **Materials Analysis*** (high-temperature metals, graphite)
6. Structural Analysis (external events, concrete performance)
7. Consequence Analysis (environmental impact)
8. Nuclear Materials and Waste Safety (enrichment process, fabrication)
9. Safeguards and Security

* major research areas

FRAMEWORK
(STATUS)

- FY 02: Limited work due to Commission direction
- FY 03+: Commission approval to proceed, funding available
- Working on plan for development of framework that will include:
  - Identification of issues (policy and technical) needing resolution
  - Approach (work that will be done) to resolve issues
  - Identification of needed resources
  - Schedule for completion of each activity
- Will interact with stakeholders to solicit their input
- Draft plan – September 2002
- Implement plan – October 2002

BENEFIT OF A FRAMEWORK

- Framework provides process, approach, and guidelines for licensing and regulation of advanced reactors.
- Current regulatory structure/framework has limited applicability.
- Incorporate PRA results and insights and make requirements more realistic.
- Framework facilitates development of design and operating requirements in a consistent, systematic, and structured manner.

ADVANCED REACTOR FRAMEWORK

- Start with current Framework developed for risk-informing Part 50.
- Use experience gained (lessons learned) from the 40 years of licensing and regulating the current LWRs.
- Framework for advanced reactors, as with current framework:
  - ☐ Have both qualitative and quantitative aspects
  - ☐ Have the top-down hierarchal structure
  - ☐ Integrate defense-in-depth at two levels
  - ☐ Provide quantitative guidelines to define what is meant by "safe enough"
- Both policy and technical issues will need to be resolved.

EXAMPLE ISSUES NEEDING RESOLUTION

- How is the framework to be used?
- Should level of safety be raised for new plants (explicitly/implicitly)?
- Should additional cornerstones (besides reactor safety) be included?
- Should environmental risk metrics (land contamination) be considered?
- Should criteria apply to single units or entire sites (what about mixed sites – current reactors and advanced reactors on same site)?
- Are the QHOs the appropriate safety goal?
- What are the appropriate safety goal surrogates?"
- For the surrogates chosen, what are the appropriate quantitative guidelines?
- What is the appropriate level of defense-in-depth, safety margin, etc.?
- Considerations will differ from those of current reactors where margins, defense-in-depth layers are well established.

HTGR Fuel Safety Research Importance

- The safety objective of HTGR fuel is to reliably contain and retain the radiologically important fission products within the TRISO coated fuel particles during normal operation, design-basis accidents, and potential severe accidents that are beyond the design-basis.

- Research would focus on TRISO coated particle fuel performance, testing methods, analytical tools, and fabrication expertise and would provide experimental data on safety margins and failure points.

Fuel Irradiation Testing

Objectives: (1) explore the limits/margins of TRISO fuel performance and fission product retention during irradiation, (2) assess irradiation test methods, (3) validate analytical models.

- Irradiation conditions beyond the expected design basis:
  - Irradiation temperature
  - Burn-up
  - Fast fluence
  - Particle power level
- Accelerated vs real-time irradiation testing
- Fission gas release monitoring during irradiations
- Pre-irradiation and post-irradiation examinations

Fuel Accident Condition Testing

Objectives: (1) explore the limits/margins of TRISO fuel performance and fission product retention during accidents, (2) assess accident simulation test methods, (3) support analytical models.

- Accident Conditions Beyond the Expected Licensing Basis:
  - Heatup events
  - Reactivity pulse events
  - Chemical attack events
- Heat-up test methods: ramp/hold vs accident profile simulation
- Fission product release monitoring during accident simulations
- Post-accident examinations

Fuel Performance Analysis Tool Development

Objective: provide NRC staff with an independent capability to predict HTGR fuel performance during normal operation and accidents:

- Predict TRISO particle behavior and failure
- Calculate fission product release (accident source term input)
- Account for statistical variations
- Expand irradiated materials properties data base
- Apply three-dimensional modeling
- Account for chemical interaction effects

Fuel Fabrication Knowledge and Information

Objective: Provide NRC staff with in-depth knowledge of the key factors of fuel fabrication that ensure quality and performance of fuel over the plant (fuel supply) lifetime:

- Fuel fabrication *process* factors (particle coatings)
- Fuel *product* factors (kernel, coated particle and element)
- Fuel process and product *quality control* measures

Materials Analysis
High-Temperature Metals

Area of Interest:

- Pressure boundary Integrity
- Internal component integrity

Examples of Issues and Needs:

- Data bases and applicability of national codes and standards
- Potential adverse effects of coolant impurities, including oxygen, on degradation
- Aging and sensitization of alloys during elevated temperature exposures
- Degradation by carburization, decarburization, and oxidation.
- Need research results to calculate failure probabilities for PRAs
- Treatment of connecting pipe as a vessel
- Inservice inspection and continuous monitoring

Materials Analysis
Graphite

Area of Interest:

Integrity and performance of graphite core components

Examples of Issues and Needs:

- Codes & standards for nuclear-grade graphite
- Data on high levels of irradiation for current graphites
- Predictive capability of irradiated properties from non-irradiated properties
- Data on oxidation kinetics
- Applicability of graphite sleeve properties to large block graphite properties

---

- Staff assignee to work with graphite experts in the UK

HTGR Reactor Systems Analysis

Area of Interest:

Prediction of reactor system conditions, responses, and source terms over the spectrum of operating conditions and accidents

Technical Disciplines:

- Nuclear Analysis
- Thermal-Hydraulics Analysis
- Severe Accident and Source Term Analysis

HTGR Reactor Systems Analysis - Nuclear Analysis (1)

HTGR versus LWR nuclear analysis:

- Particle fuel, >5% initial enrichment
- Inert He coolant, graphite moderator (longer neutron migration)
- Long annular cores with absorber elements in reflector
- Little or no in-core instrumentation

Examples of HTGR modeling and validation issues:

- Temperature coefficients, reactivity transients
- Absorber worth, axial stability
- Graphite annealing heat sources
- Both fertile and fissile particles, burnable poisons
- Hot spots, seismic compaction
- Burnup measurements & discharge criteria

HTGR Reactor Systems Analysis - Nuclear Analysis (2)

Nuclear Analysis to Support TRISO Fuel Testing:

- Worst credible reactivity transients (e.g., prompt pulses)
- Applicability of delayed out-of-pile accident testing
- Applicability of irradiation in test reactors versus HTGRs

Infrastructure Activities:

- Nuclear data libraries (ENDF/B-VI)
- Scoping studies for local neutronics and depletion modeling effects
- PARCS geometry modifications, MIT-INEEL core depletion code
- PIRT to further identify and rank needs

HTGR Reactor Systems Analysis
Thermal Hydraulics

Phenomena and issues - Examples:

- Helium viscosity increase with temperature
- Fluid flow in porous and solid structures (e.g., hot spots)
- Thermal flow stratification and mixing
- Natural circulation inside pressure boundary and in RCCS
- Conduction cooldown - conductive and radiative heat transfer
- Turbomachinery modeling

Infrastructure Activities

- Development of TRAC-M for HTGR analysis (build upon GRSAC and THATCH codes)
- CFD (FLUENT) for component analysis and model development
- PIRT exercises to develop data and modeling needs

HTGR Reactor Systems Analysis
Severe Accident and Source Term Analysis

Phenomena and Issues - Examples:

- Air ingress and Moisture ingress
- Graphite oxidation and dust
- Fission product inventories andchemistry
- Release mechanisms from fuel - normal and accidents
- Fission product transport and deposition - normal and accidents

Infrastructure Activities:

- Development of MELCOR for HTGR analysis (build upon GRSAC)
- CFD (FLUENT) for component analysis and model development
- PIRT exercises to develop data and modeling needs

Summary

- NRC Research Role in Advanced Reactors
- ACRS comments and feedback
- Consideration of AECL ACR-700, ESBWR, SWR-1000
- Additional stakeholder interactions
- Transmit plan to Commission in Fall 2002.

U.S. Nuclear Regulatory Commission

PRESENTATION TO ACRS ON EARTH SCIENCES & EARTHQUAKE ENGINEERING

by

Michael Mayfield

Daniel Dorman

Andrew J. Murphy

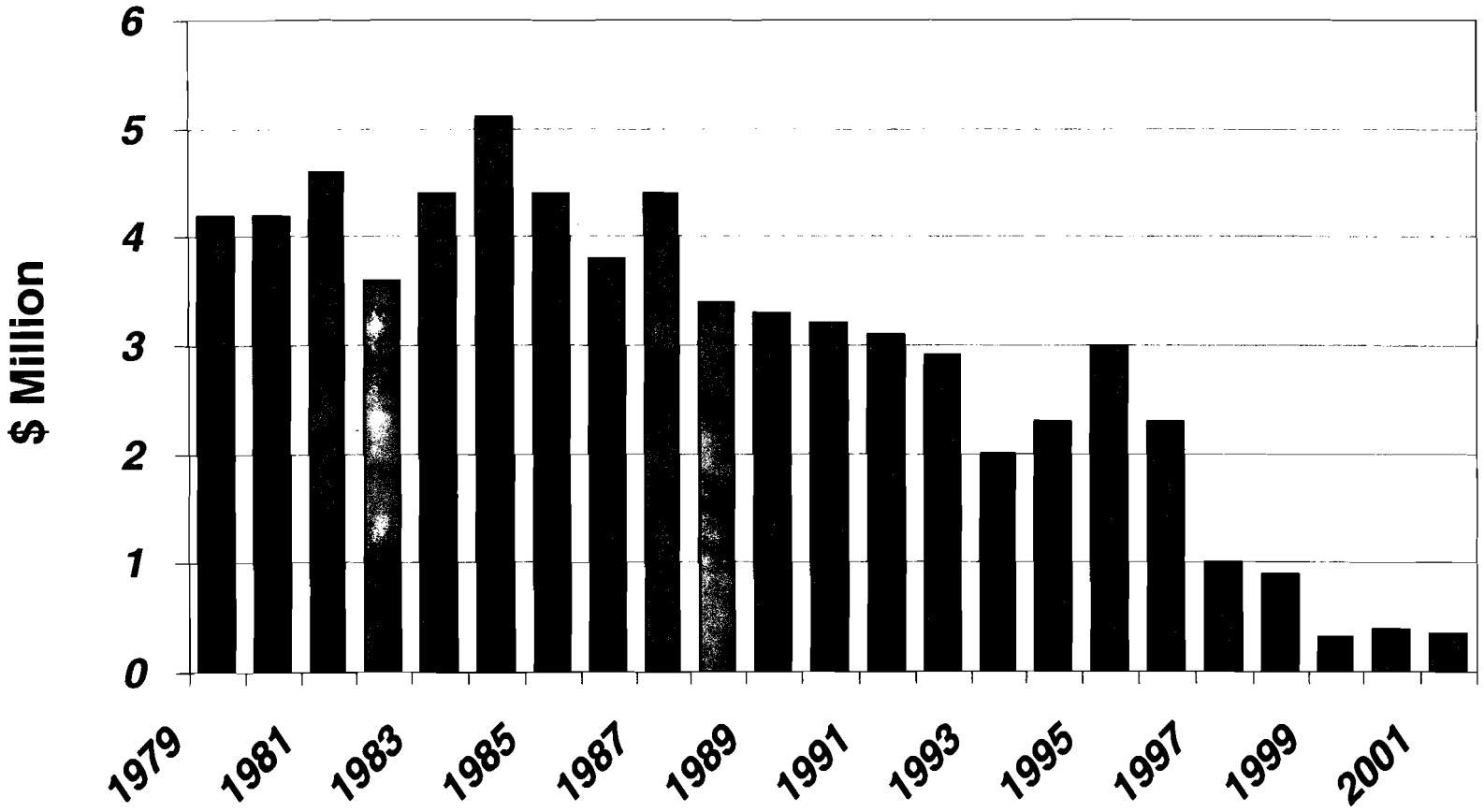
July 11, 2002

Outline

- Past Activities
  - Earth Sciences
  - Earthquake Engineering
  - Regulatory Products & Outcomes
- Current Activities
- Funded Future Activities
  - Open Issues

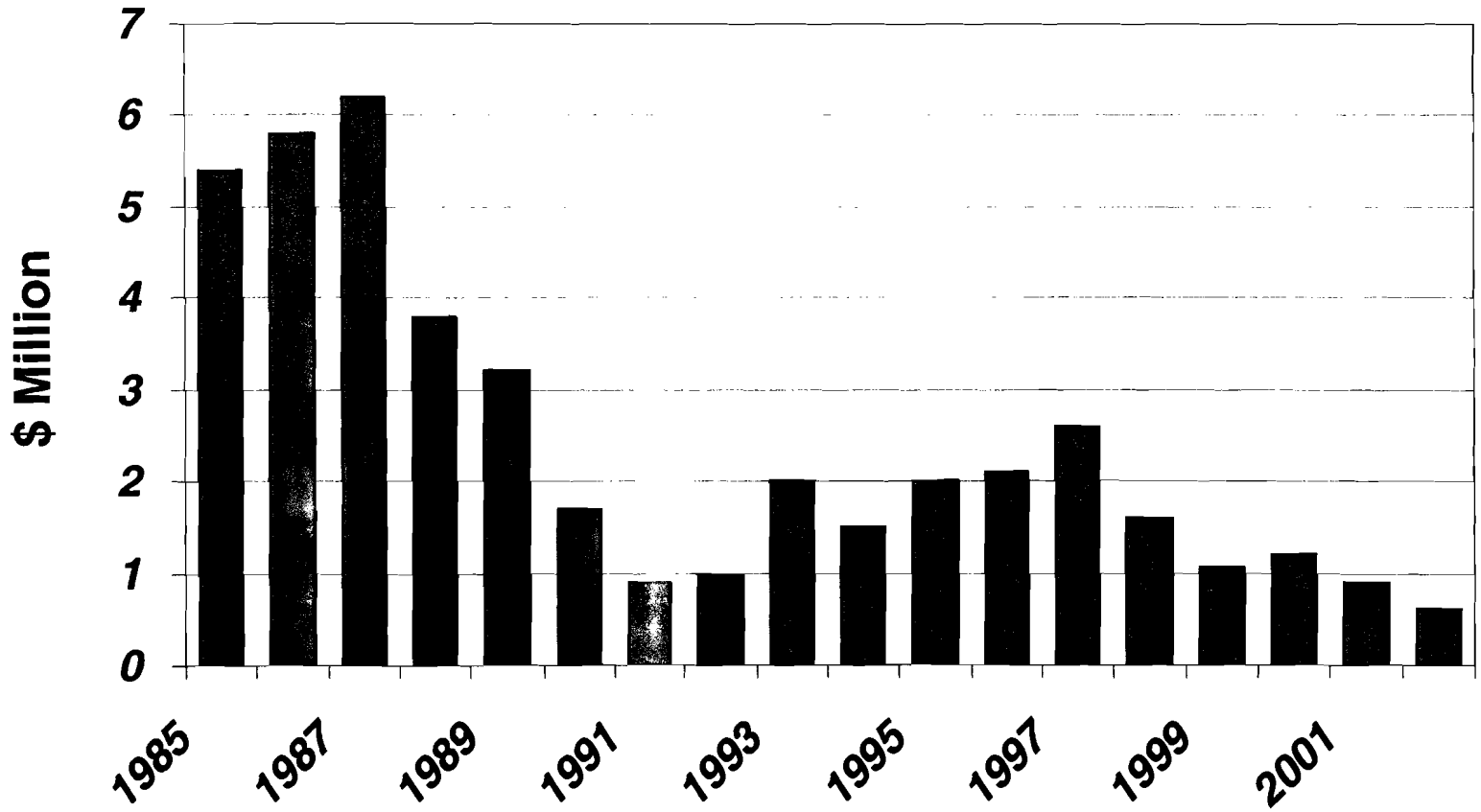
U.S. Nuclear Regulatory Commission

Earth Science Budget



U.S. Nuclear Regulatory Commission

Earthquake Engineering Budget



Earth Sciences - Solid

- Seismicity – Paleoseismicity
  - Geology
  - Seismographic Networks
    - Microearthquake Networks
    - National Seismographic Network – ANN
  - Seismic Source Zones
  - Ground Motion Propagation – Attenuation
  - Probabilistic Seismic Hazard Assessments Guidance

Geological Studies

- Cooperative Studies with State Surveys
- Close Coordination for Regional Studies
  - New England
  - New Madrid
  - Charleston – US Geological Survey
- Principal Conclusion: Low Level Correlation Between Seismicity & Surface Geology

Seismographic Networks

- Regional Microearthquake Networks
  - Northeast
  - Southeast
  - New Madrid
  - Northwest
- Telephone Divestiture
- National Seismographic Network with USGS

Probabilistic Seismic Hazard Assessments

- SEP & Charleston Earthquake Issue
- Original LLNL & EPRI Results
- Utility of Results
- Senior Seismic Hazard Analysis Committee
  - Trial Implementation
  - Full Implementation

Earthquake Engineering

- Fragility of Structure & Components
- Margins Projects
- Soil-Structure Interaction Projects
- Response of Aged Structures, Systems, & Components

Fragility of Structures & Components

- SSMRP
- PFDRP with EPRI
- Electrical Components – Relays, Cabinets, Switchgear
- Concrete Anchorage
- Cooperative Program with NUPEC on Containments, Structures, & Components

Seismic Margins Programs

- NRC - Fault Tree-Event Tree Approach
- EPRI – Success Path Approach
- Individual Plant Examination for External Events – A principal Use

Soil-Structure Interaction

- CARES Code
- Lotung Experiment
- Hua Lien Experiment
- Cooperative Efforts with NUPEC
  - Building Models & Real Earthquakes
  - Building Models & Shake Table

Response of Aged Structures, Systems & Components

- Change in Properties
- Dynamic Response
- Risk Significance

Regulatory Products & Outcomes

- Seismic PRA & Margins Methodologies
- Seismic Hazard Assessments
- IPEEE
- Part 100.23 & Reg. Guide 1.165 – OBE Analysis Optional
- Batch of New & Revised Reg. Guides
- Consensus Codes & Standards

Budget History

- Highly Leveraged Programs
  - With US Geological Survey & State Agencies
  - With Domestic Partners – DOE, EPRI, NSF, US Army Corps
  - With International Partners – Canada, NUPEC, CEA, IPSN

Current Activities

- Ground Motion Project
- Seismic Source Zone Characterization
- Collaborative Studies of Fragility with NUPEC
- Regulatory Products & Outcomes

Future Funded Activities

- Current Activities
- SHHAC Implementation

Continuing & Emerging Issues

- New Data &Interpretations
  - East Tennessee Seismic Zone (GSI)
  - Implications of Recent Earthquakes – Turkey & Taiwan
  - Coordination of New PSHAs - US GS & EPRI
    - Evaluation & Use
- New Technology
  - Buried or Deeply Embedded Structures
    - Ground Motion Input Guidance
    - Soil-Structure Interactions – Building & Interconnects
  - Fragility of New Structures & Components

Continuing & Emerging Issues cont.

- Performance-Based/Risk-Informed Analysis & Design
  - Hazard/Risk Consistent Analysis to Predict Probabilistic Response & Loads for Linear & Non-Linear Analysis
  - Development of Performance- Based Target & Limit States, e.g., Displacement-Based Design
  - Development of & Compatibility with Codes & Standards, e.g., ASCE and Revision of Regulatory Documents

Current Outlook

- Earth Science & Earthquake Engineering Research Program Has Fallen to or below a Level to Sustain Core Competency
- Succession Planning for Staff and Contractors is Critical
  - Evolving Technology & New Data & Interpretations
  - New Siting - ESPs

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POWER UPRATES

DEVELOPMENT OF REVIEW STANDARD FOR EXTENDED POWER UPRATES

ACRS 494th Meeting

July 11, 2002

- Background
- ACRS Feedback
- Review Standard
- Benefits of a Review Standard
- Extended Power Uprate Review Standard Effort
- Schedule
- Conclusions

BACKGROUND

- Maine Yankee Lessons Learned
- Template Safety Evaluations
- SECY-01-0124, dated July 9, 2001
- Commission Meeting with ACRS, December 5, 2001
- ACRS Letters on EPU Reviews
- SECY-02-0106, dated June 14, 2002

ACRS FEEDBACK

- Documentation
- Reload Analyses
- Independent Calculations
- Anticipated Transients Without Scram
- Fuel

ACRS FEEDBACK

- Operator Action Times
- Material Degradation
- Containment Response
- Large Transient Testing
- Probabilistic Risk Assessment
- Communication with Inspection Staff

REVIEW STANDARD

- Clearer Definition of Scope of Review
- Technical Review Criteria
- Process Guidance
- Model Safety Evaluations

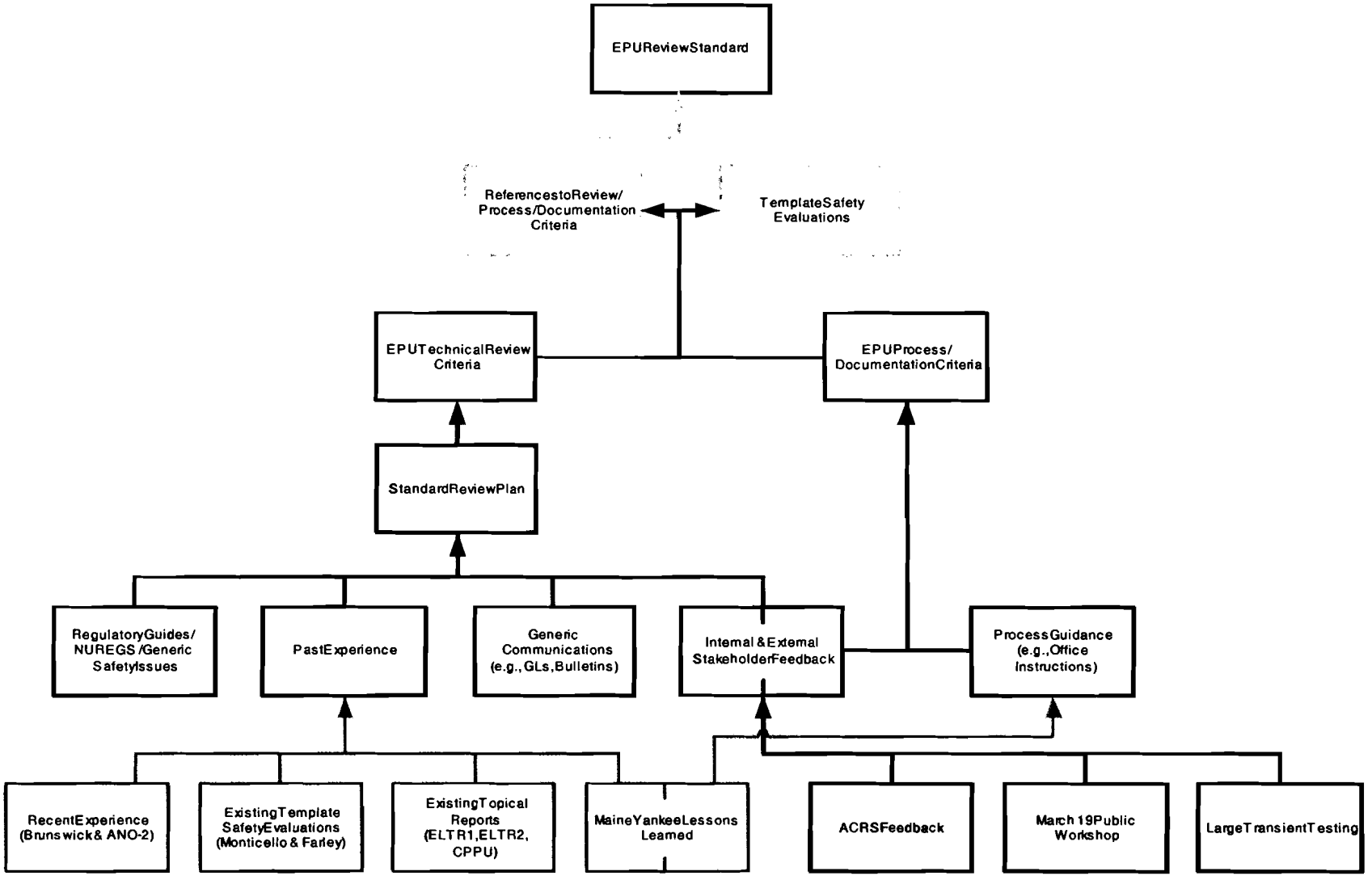
BENEFITS OF A REVIEW STANDARD

- Comprehensive Guidan Document
- Retain Institutional Knowledge
- Provide Technical Review Criteria and Process Guidance for New Hires
- Update Existing Review Criteria (e.g., Existing SRP)

BENEFITS OF A REVIEW STANDARD

- Consistent with NRR's Vision for Centralized Work Planning
- Improve Focus, Consistency, Completeness, and Thoroughness of Review
- Improve Documentation of Review

Extended Power Uprate Review Standard Effort



CurrentExperienceResultingfromMaineYankee

SCHEDULE

- Issue Review Standard for Interim Use and Public Comment – December 2002
- ACRS Review Following Public Comment
- Issue Final Review Standard – Early 2004

CONCLUSIONS

- The Staff is Developing a Review Standard for Extended Power Uprates
- Development of the Review Standard will Address ACRS and Other Stakeholder Feedback Received to Date
- Review Standard is Expected to Result in Improved Focus, Consistency, Completeness, and Thoroughness of Reviews, and Better Documentation of Reviews.
- Development of a Review Standard is Consistent with and Goes Beyond the Committee's Recommendation to Develop an SRP

Probabilistic Fracture Mechanics Techniques Used in the Re-Evaluation of the Pressurized Thermal Shock Rule (10CFR§50.61)

Mark Kirk, Shah Malik, Nilesh Chokshi

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

Materials Engineering Branch

Richard Bass, Claud Pugh, Terry Dickson, Paul Williams

Oak Ridge National Laboratory

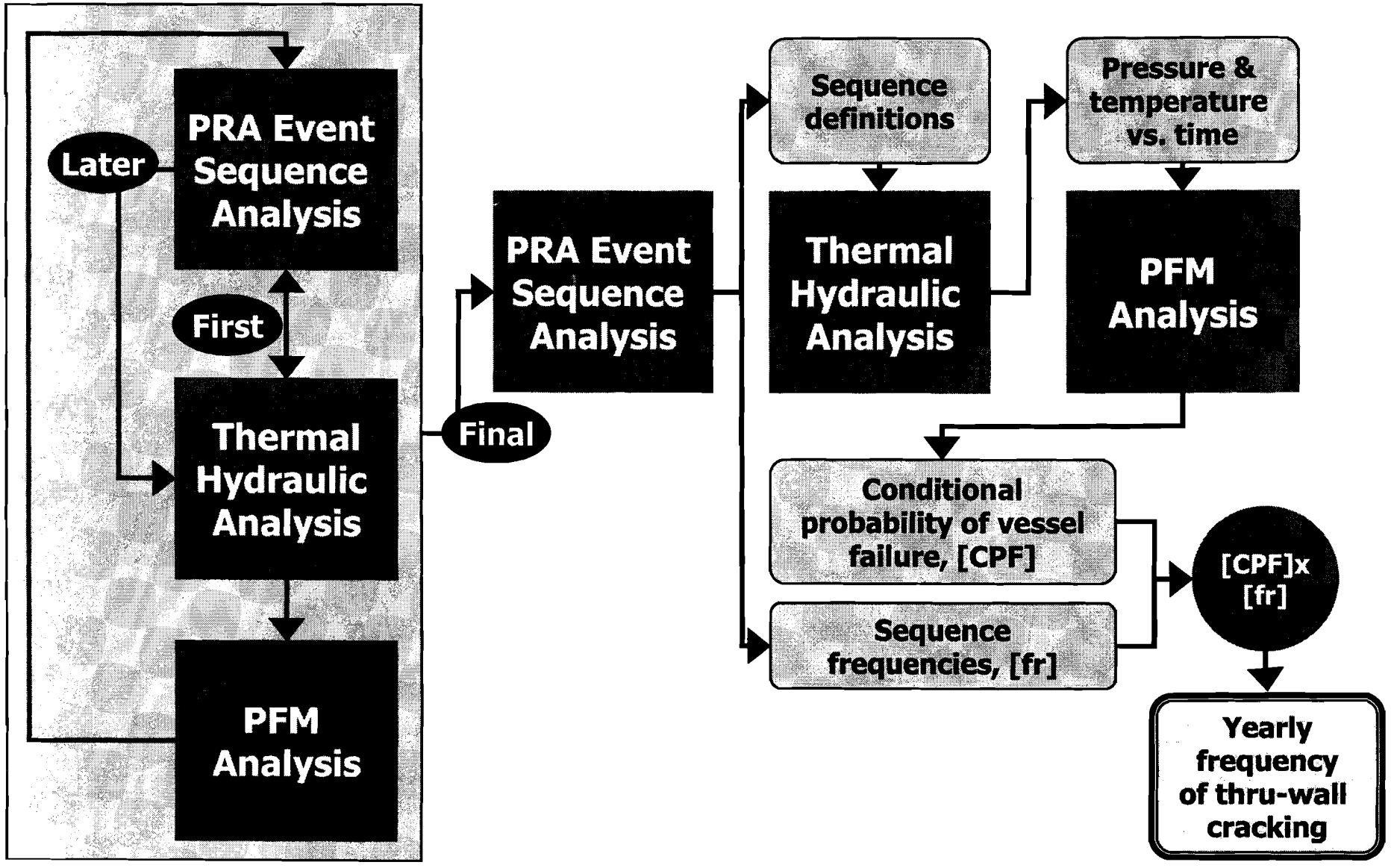
ACRS Briefing – Rockville, MD – July 12th, 2002

VG 1

Background

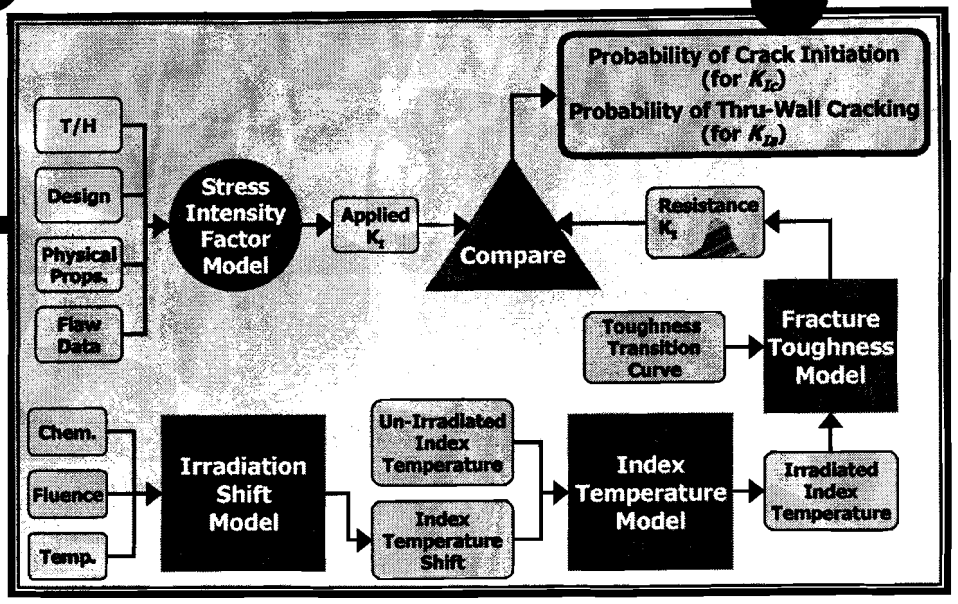
- NRC RES has briefed ACRS several times over the past few years regarding the use of probabilistic fracture mechanics techniques as part of the process used to assess the technical basis for updating the PTS rule (10CFR§50.61)
- ACRS requested that RES provide additional background concerning
  - The appropriateness of using linear elastic fracture mechanics in such assessments
  - Validation of LEFM applicability to nuclear RPV fracture assessment

PTS Re-Evaluation Process



Probabilistic Fracture Mechanics Summary

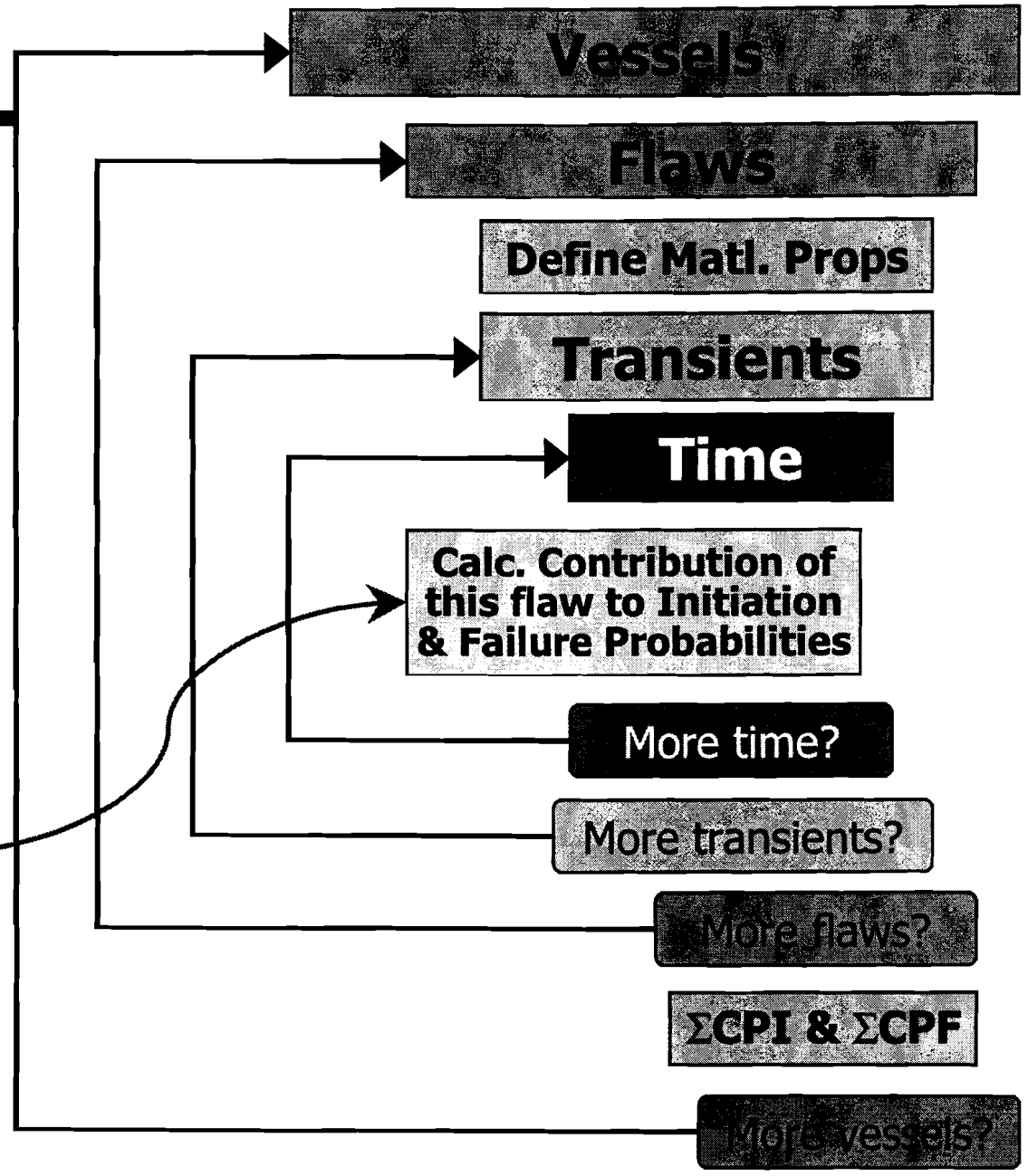
- **Toughness**
  - **Referenced to toughness data & physical understanding**
    - ✓ **Significant conservative bias in un-irradiated index temperature removed**
    - ✓ **Non-conservatism in arrest model removed**
    - ✓ **Aleatory nature of toughness uncertainty quantified**
- **Embrittlement**
  - **Referenced to toughness data & physical understanding**
    - ✓ **Correlation with better empirical/physical basis**
    - ✓ **Slight biases in in CVN-based shift estimates removed**



- **Fluence**
  - **Spatial variation in fluence recognized, significant conservatism associated with max fluence assumption removed**
- **Flaws**
  - **Based on significantly more data than before**
  - **Most flaws now embedded rather than surface flaws**
  - **More flaws than before**

FAVOR Looping Structure

- **FAVOR features a nested Monte-Carlo loop structure implemented in FORTRAN that simulates uncertainties in input variables**
- **Innermost loop is a DETERMINISTIC CALCULATION**
- **We will show that this deterministic calculation is appropriate to predicting RPV failure**



Evidence from Large-Scale Experiments Supporting Applicability of LEFM to Integrity Assessments of Nuclear Reactor Pressure Vessels

B. R. Bass, T. L. Dickson,
C. E. Pugh, and P. T. Williams

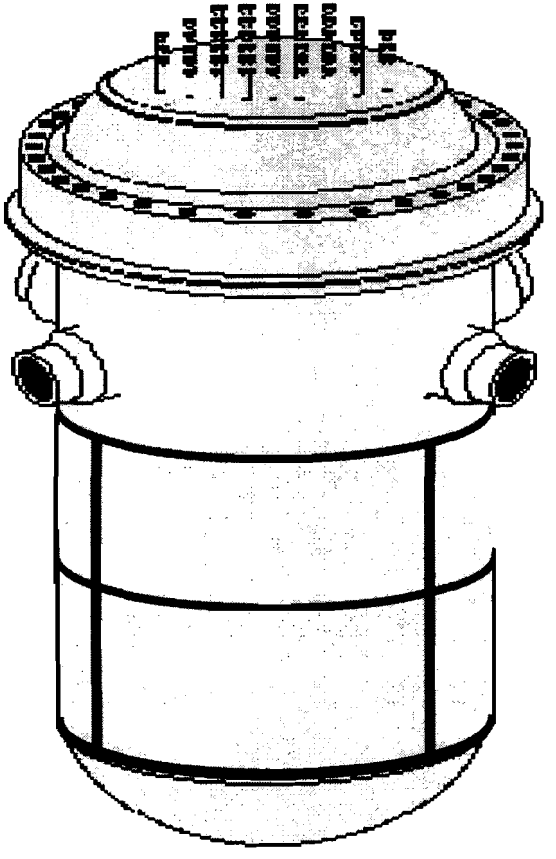
Oak Ridge National Laboratory

July 12, 2002

NRC Headquarters, Rockville, Maryland

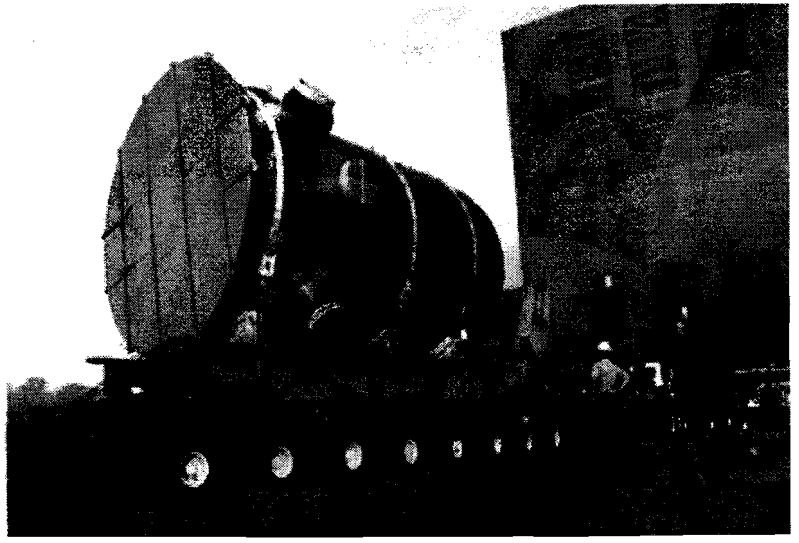
This Presentation is Organized to Review Key Phases of NRC's Determination of RPV Fracture-Prevention Technology

- To illustrate systematic and evolutionary nature of NRC's Research re RPV fracture-prevention technology
- To briefly review early large-scale tests re applicability of LEFM
- To review in more detail the pressurized thermal-shock experiments (esp. PTSE-1)
- To present original HSST and FAVOR analysis results for PTSE-1



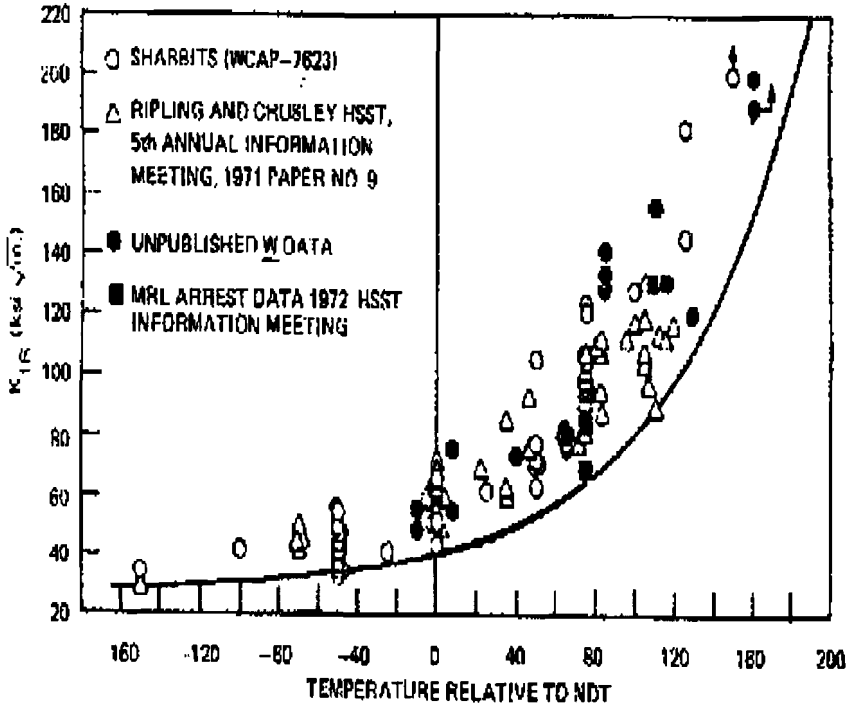
NRC/AEC's Confirmatory Research on RPV Technology has from Its Beginning been a Systematic Process

- Began in 1967 after intense planning that included representation from PVRC, ASME, vendors, universities, labs, etc.
  - Produced pressure vessel technology report (ORNL NSIC-21, December 1967)
- First priority of ensuing research was to establish basic fracture characteristics of thick sections of RPV steels
- Large-scale vessel tests were the major means for evaluating applicability of methods
- Fracture models and computational methods were integral part of process



Early Emphasis on Properties Generation and Fracture Testing/Modeling Established Fracture Design Curves Used by ASME to this Day

- Procured 500,000 pounds of RPV steel
- Large number of fracture-toughness tests (participants included MRL, Westinghouse, SwRI, BCL, B&W, etc.)
- Dependence of temperature, load-rate, variability, etc., were studied
- Produced data that defined the toughness transition curve which became basis for ASME K_{IR} Curve (See WRC Bulletin 175)
- Data and test methods contributed to various ASTM Standards (e.g., E-399, E-1737, E-1820, E-1221, and E-1921)

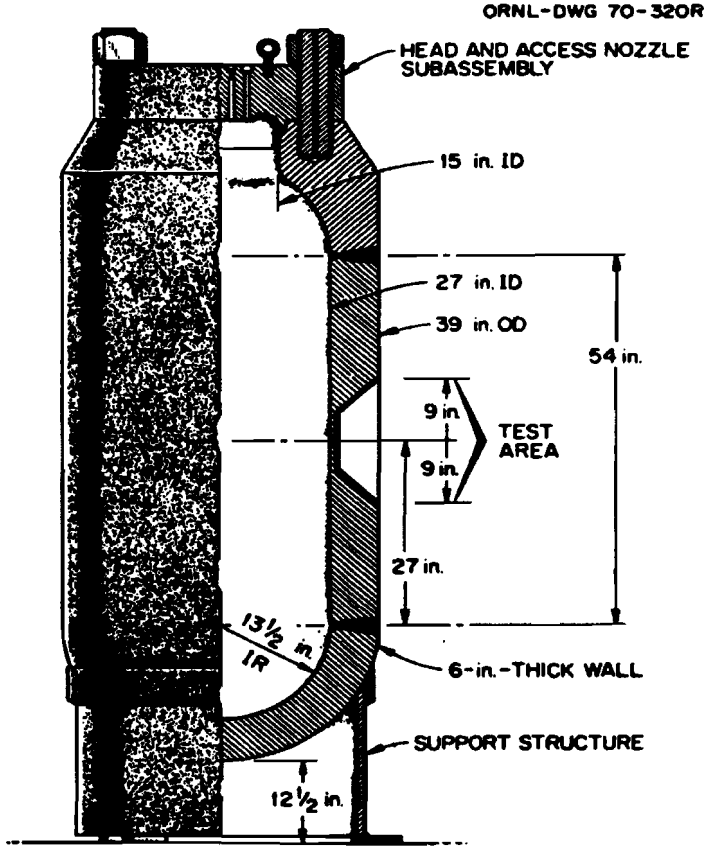


Key RPV Integrity Issues were Addressed by the Three Sets of Large-Scale Cylinder Experiments Performed at ORNL over Two Decades

- Intermediate test vessels (ITVs) (1972-1982)
  - Crack initiation through full transition region
  - Confirm margins against fracture for RPVs
- Thermal-shock experiments (TSEs) (1975-1985)
  - thermal-only loading (large-break LOCAs)
  - initiation – arrest – reinitiation
- Pressurized thermal-shock experiments (PTSEs) (1982-1988)
  - coordinated thermal shock and internal pressure
  - initiation – arrest – reinitiation
  - warm-prestressing

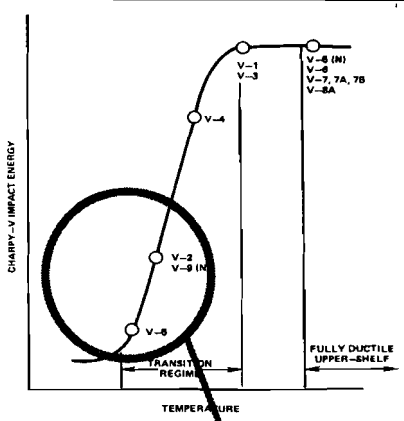
A Series of Thick-Cylinder [Intermediate Test Vessel (ITV)] Tests was Performed

- Ten 6-in Thick Vessels Procured (A508, Class 2 Steel) (Three with Nozzles)
- The ITV Tests Took on Added Importance after it was Decided that Full-Scale Tests would be Cost Prohibitive
- 12 Tests Performed using Nine Vessels
- To Examine Fracture Behavior of Forged, Plate, and Weld Materials
- To Demonstrate Capability to Predict Transition-Temperature Behavior of Thick Vessels
- To Verify Methods of Fracture Analysis for Thick Sections over Range of Conditions

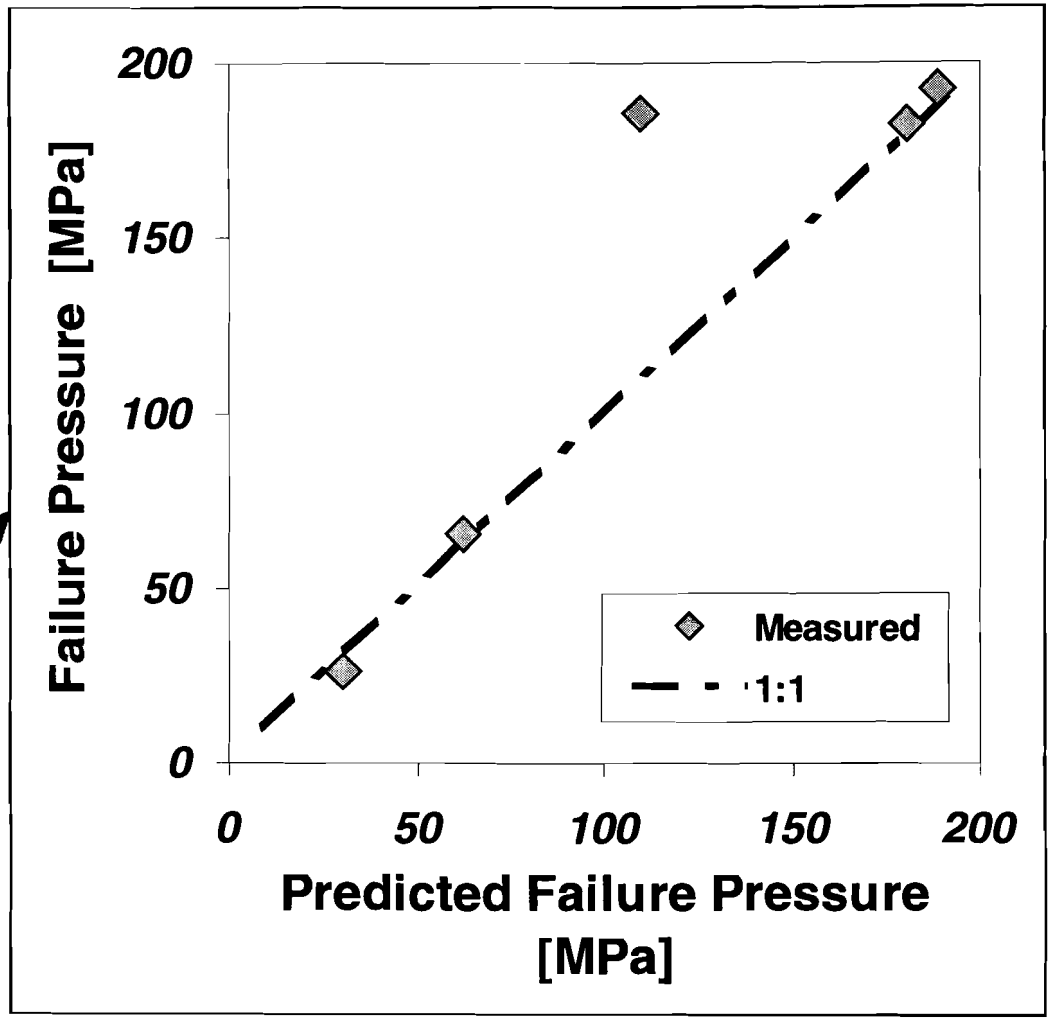


The Failure Pressures Measured During Intermediate-Scale Vessel Tests Were Predicted Well By LEFM

Tested from Lower to Upper Shelf

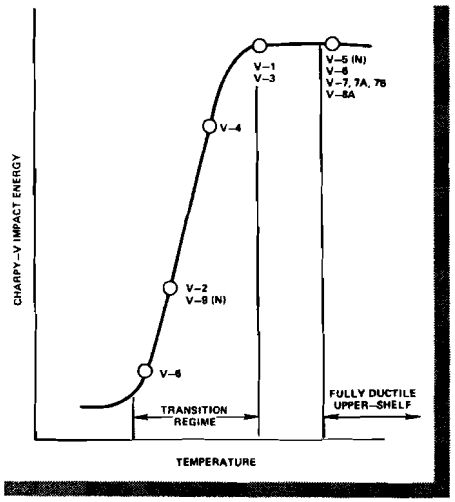


Results from ITVs Performed in Fracture Mode Transition

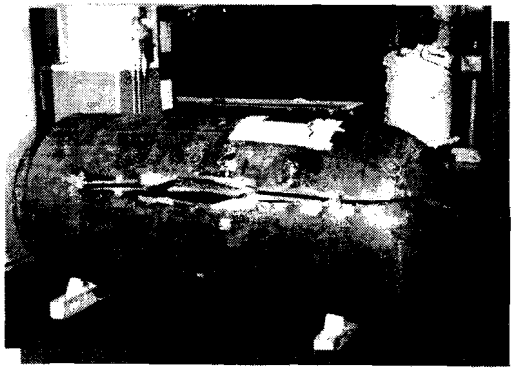


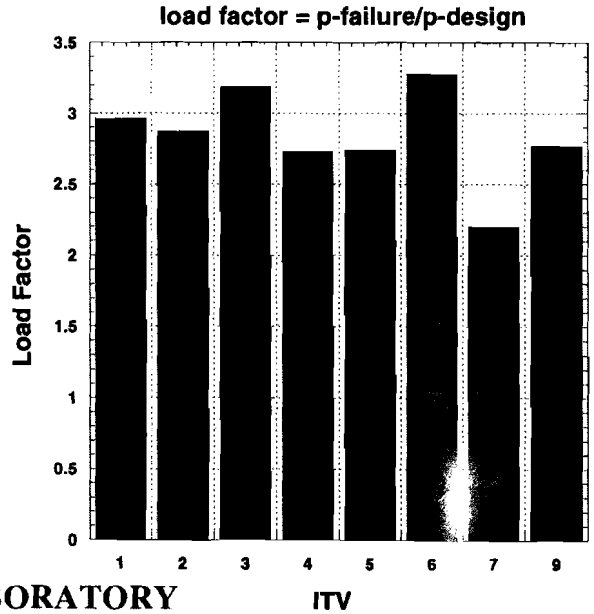
ITV Experimental and Analytical Results Supported Existence of Intended Margins for Full-Scale RPVs

ITV Program

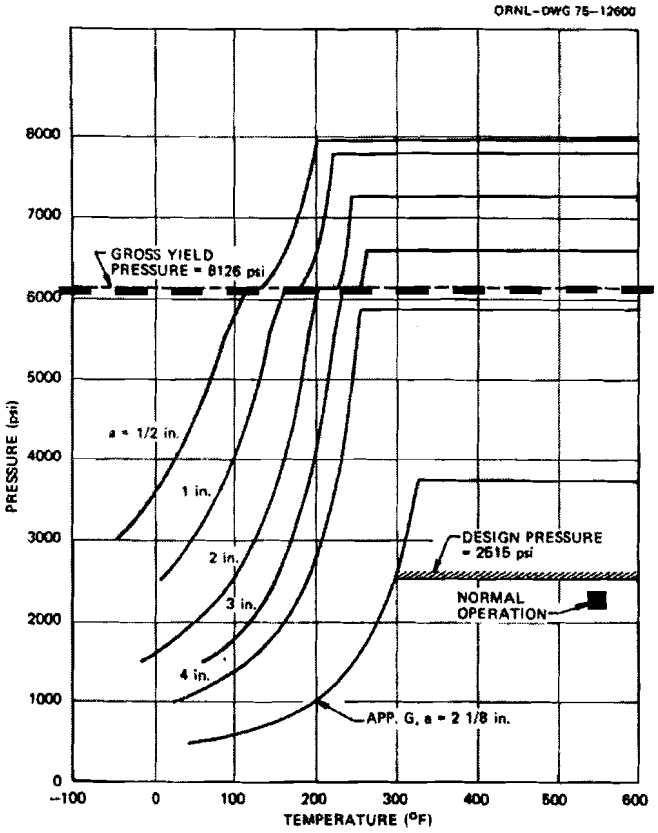


Flawed ITVs (1-7,9) tested
Over range of temperatures and fracture modes; load factors varied from 2.2 to 3.3



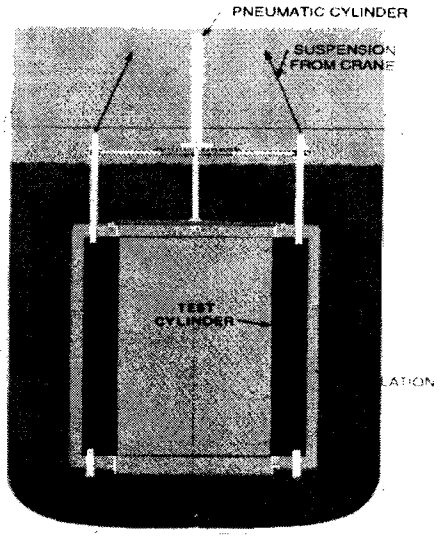


RPV Assessment



After ITV Testing was well underway, a Program of Eight Thermal-Shock Experiments (TSEs) was Initiated using 6-in. Thick Specimens

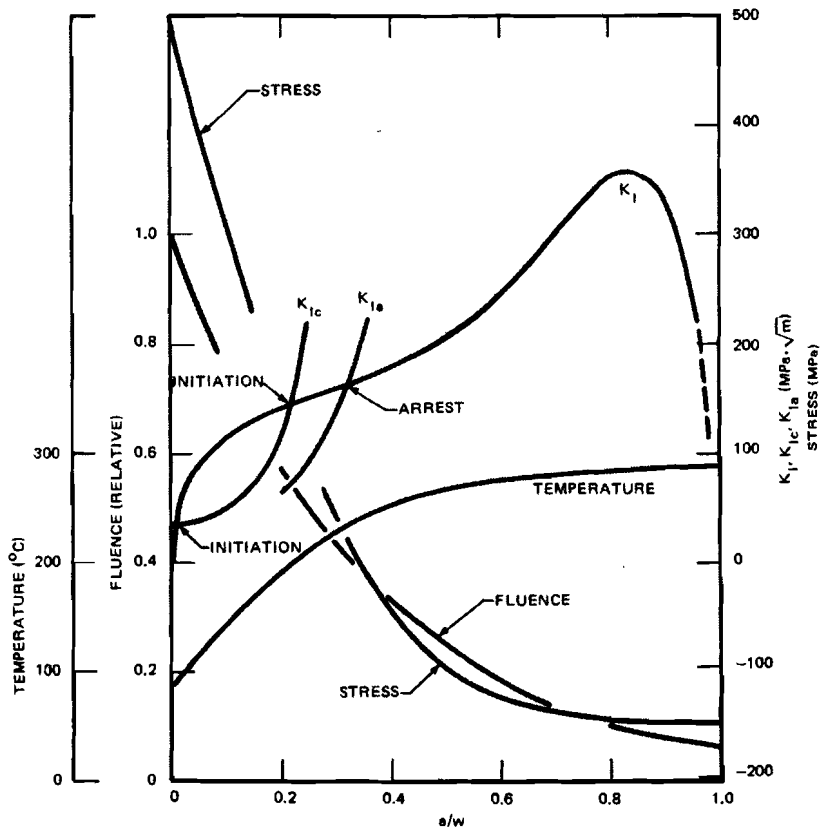
- **Objective: Evaluate initiation-arrest events under OCA conditions**
  - **RPV steels tested**
  - **Thick wall cylindrical sections**
  - **Through thickness thermal gradients**



- **Tests used to**
  - **Evaluate analysis methods**
  - **Establish appropriateness of LEFM**

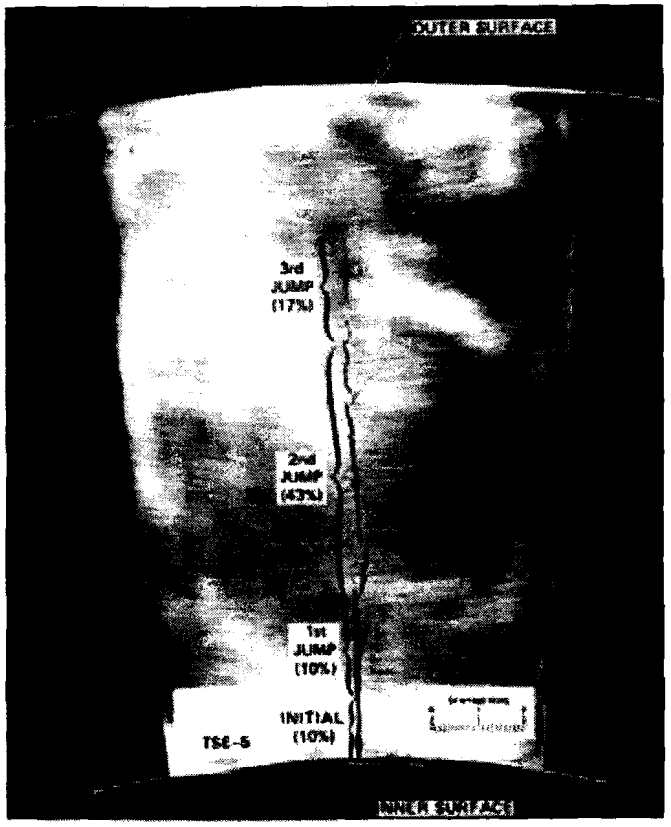
Typical LBLOCA (and TSFs) gives Rise to Transient Through-Wall Variations in a Thick Vessel

- Important Through-Wall Factors Include Temperature, Stress, Fluence, Stress-Intensity Factor, and Fracture Toughness
- Multiple Initiation-Arrest Events Predicted
- Both a Shallow and a Deep Flaw can Initiate
- Warm Prestressing can Occur for Deep Cracks
  - Limits Propagation
- Arrest in Rising K-Fields Predicted
- Arrest Predicted at K_I Values above ASME Limit



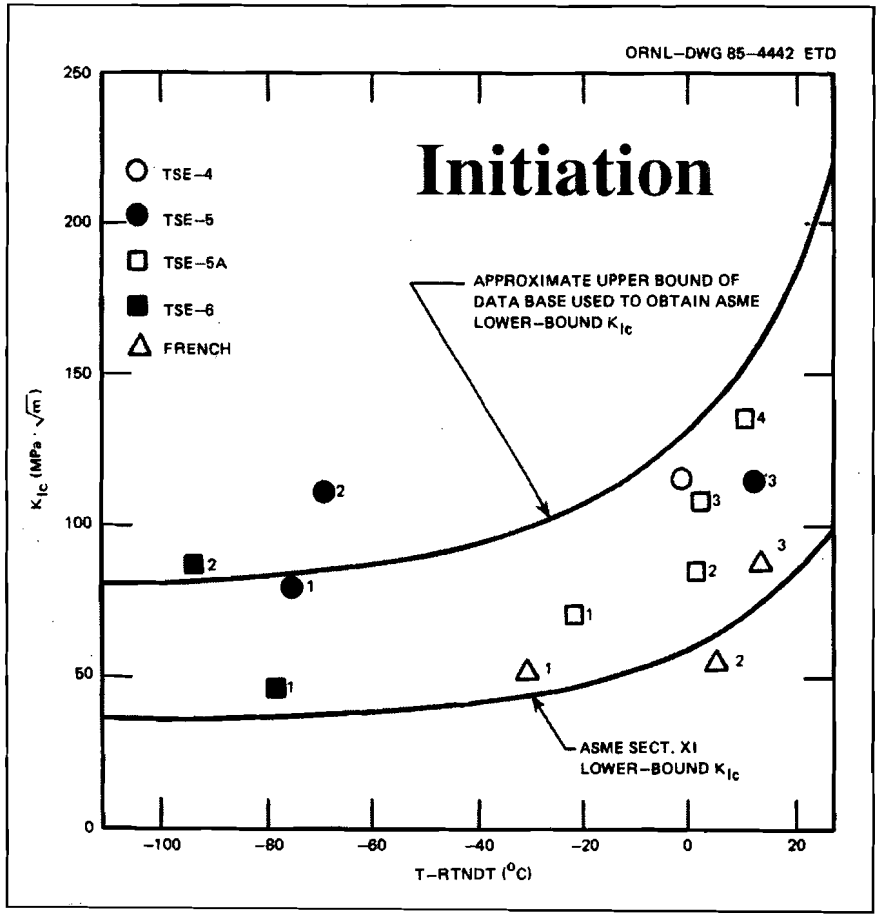
Consistent with Predictions, TSE-5 Experienced Three Run-Arrest Events of a Long Surface Flaw

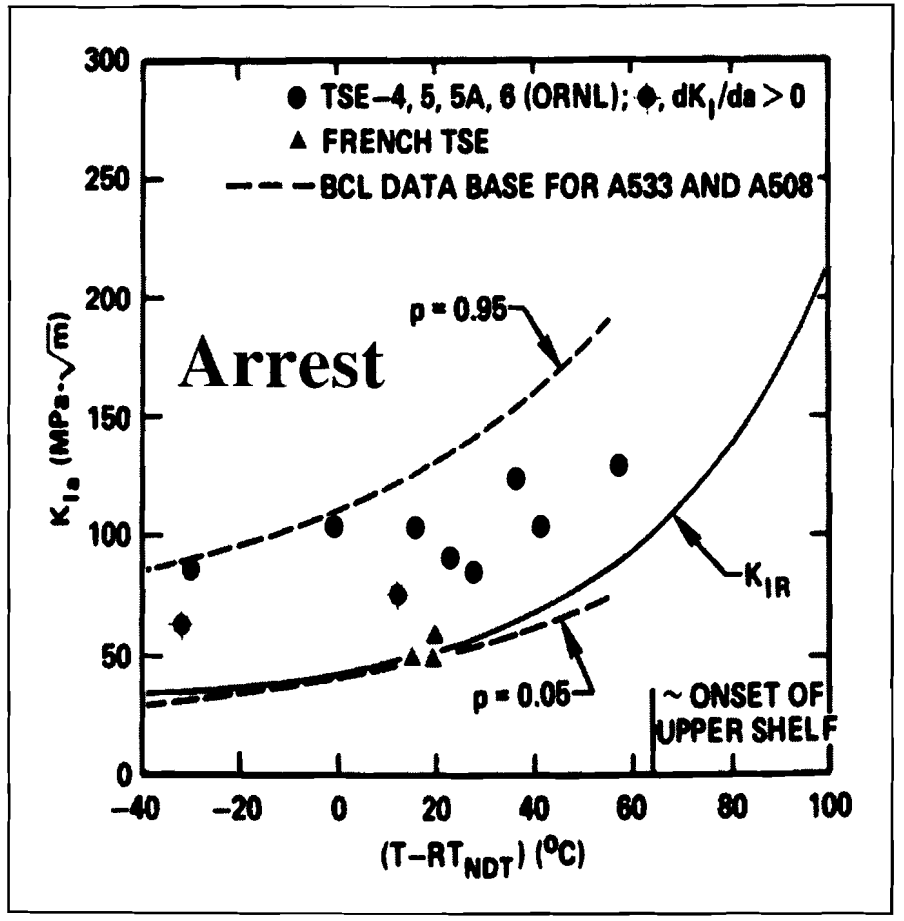
- Pretest Analyses Predicted the Long Flaw to Propagate from a/w = 0.1 to a/w = 0.5 (or 0.7 if WPS were not to become effective prior to 3rd initiation)
- COD Output Identified Times of Fracture Events
- Posttest Sectioning Shows the Dimensions of Three Events
- Jumps were Relatively Long
- Actual Flaw Propagated from a/w = 0.1 to 0.8



Initiation and Arrest Toughness Data from Large Scale TSEs Agrees Well with Small Specimen Data

(to within material scatter)

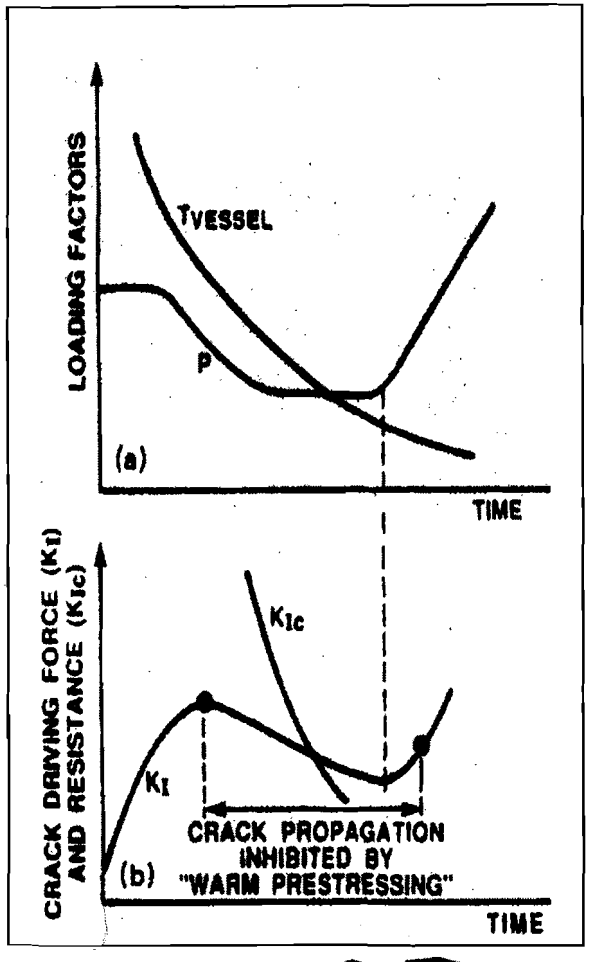




→ Points from large scale tests
→ Curves from small scale specimens

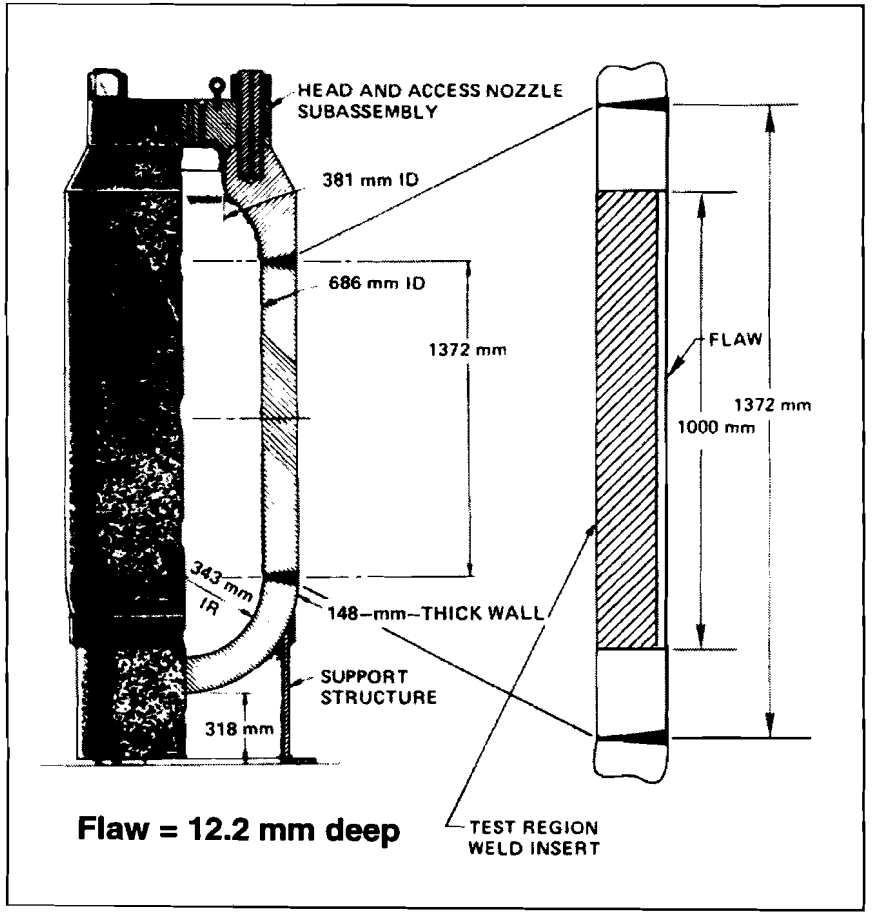
HSST Pressurized-Thermal-Shock Experiments (PTSEs) were Performed for Confirmation or Development of Fracture Analysis Methods

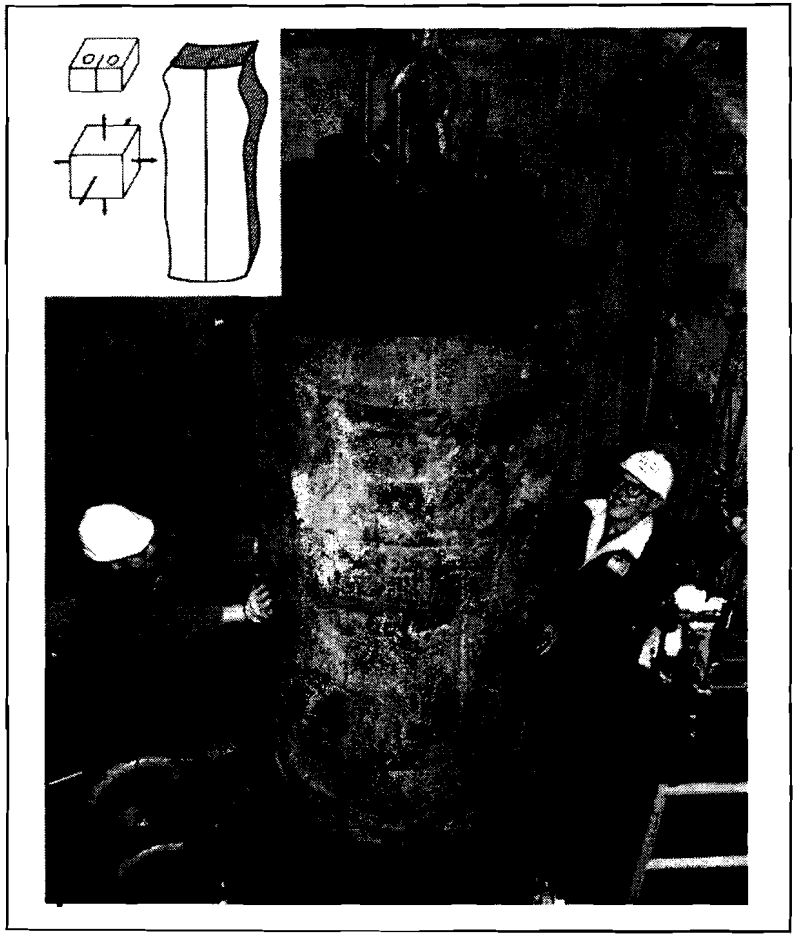
- The Experiments Investigated
  - warm-prestressing effects
  - nature of cleavage crack arrest at temperatures near or above onset of Charpy (ductile) upper shelf
  - behavior of low upper-shelf energy steel
- Long Surface Cracks Inserted into Thick (6-in) Wall Vessels
- Flawed Vessels Subjected to Coordinated Thermal-Shock and Internal Pressure Loading
- Experiments Designed and Analyzed Using
  - small-specimen fracture toughness data
  - OCA LEFM analysis code (a precursor to FAVOR)
- Experiments also Analyzed using Current FAVOR Code



UT-BATTELLE

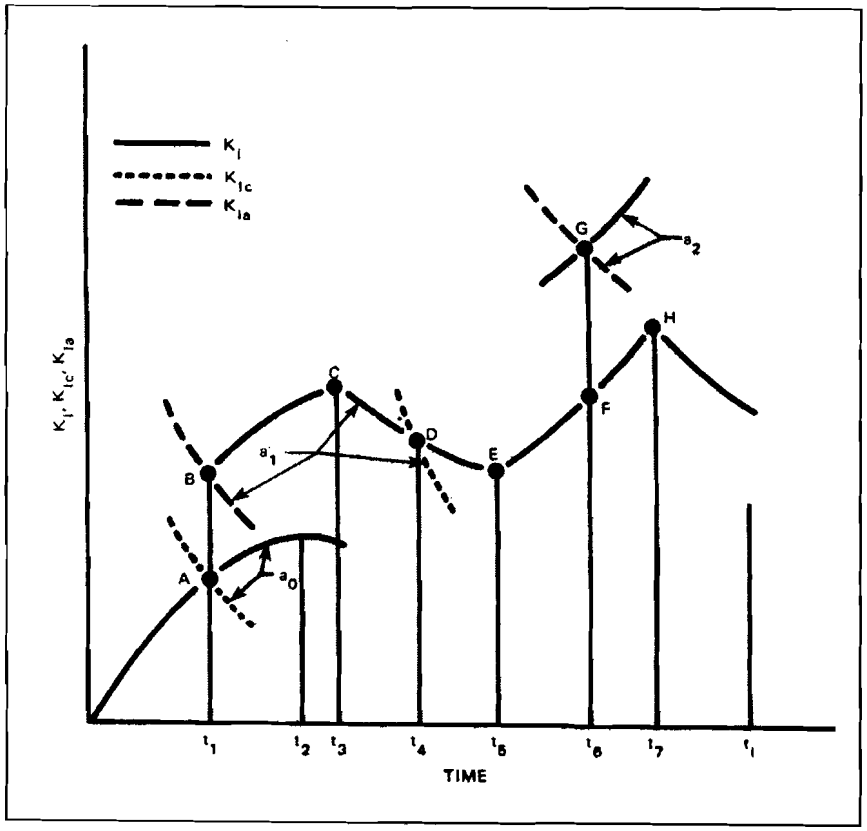
The Test Vessel was Thick Enough (148 mm) and Long Enough to Provide Crack-Tip Restraint Similar to that in a Full-Scale RPV





Test Plan Devised for PTSE-1 Focused on Warm-Prestress Effects and Cleavage Arrest of a Fast-Running Crack on the Charpy Upper Shelf

Planned Transient

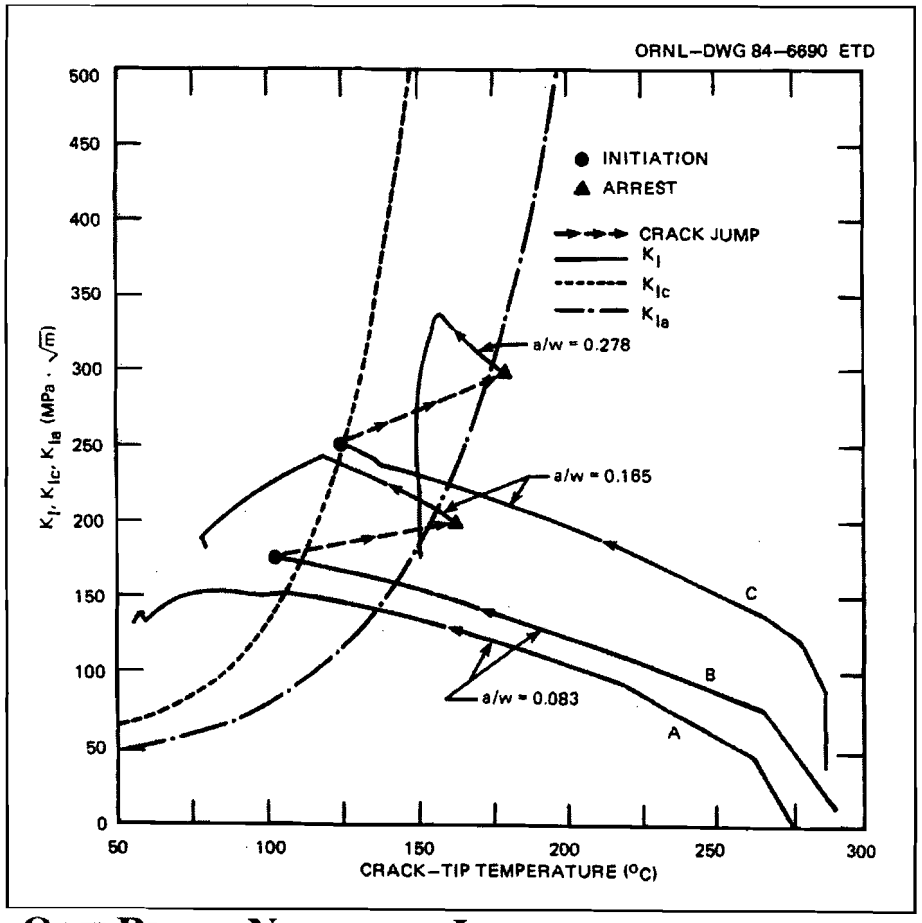


Test Plan Envisioned a Single Transient

- A-B: initiation and arrest of a cleavage fracture
- C-D-E: WPS eventually relieved by increasing the pressure (E-F)
- F-G: cleavage re-initiation and arrest on Charpy upper shelf

Actual experiment was conducted in three transients

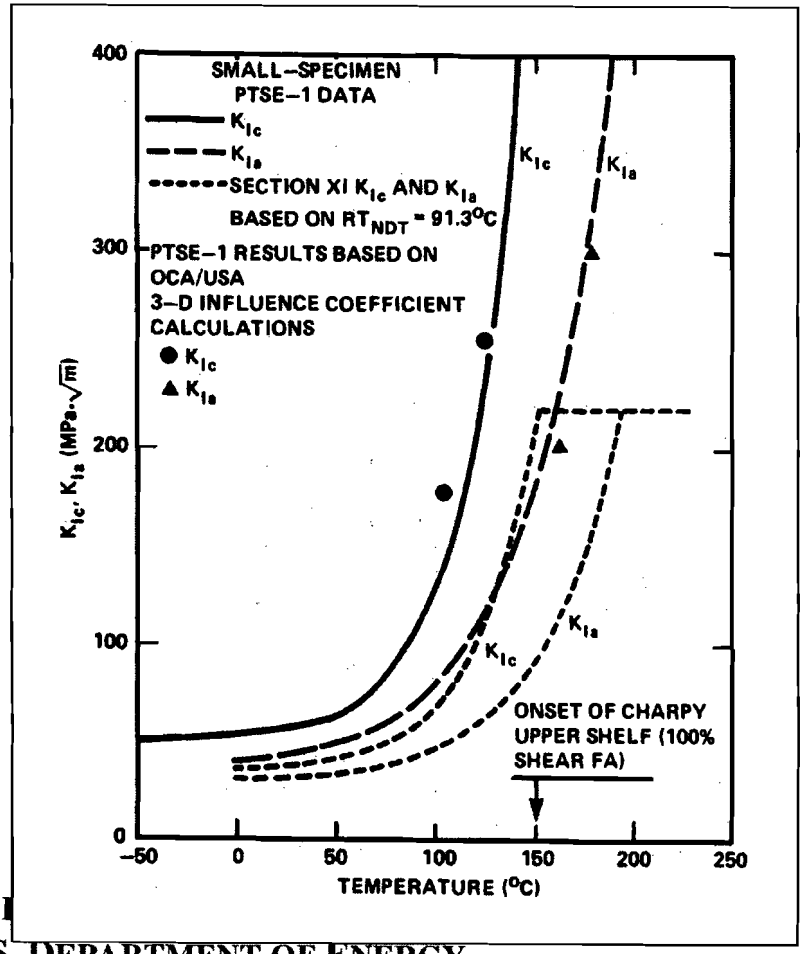
OCA Fracture Assessments of PTSE-1 Transients were Consistent with Small-Specimen Fracture-Toughness Data and Indicated Effectiveness of WPS



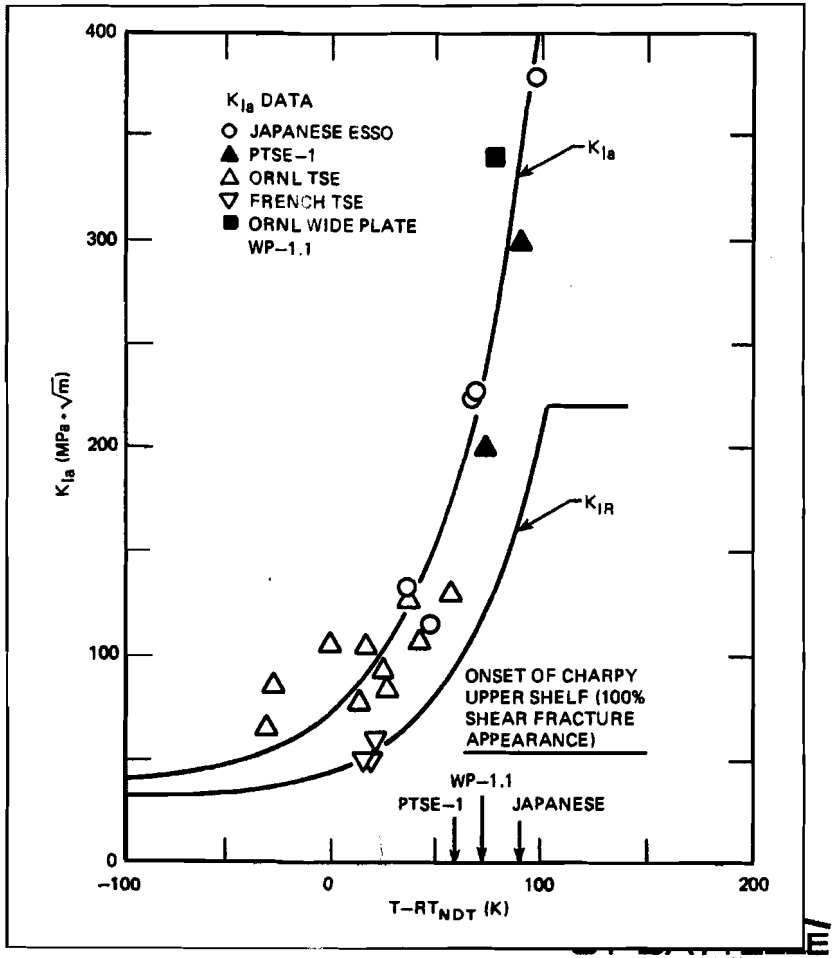
- Initiation and arrest events in PTSE-1B and -1C occurred at points close to K_{Ic} and K_{Ia} curves from small-specimen data
- WPS indicated in both the PTSE-1A and -1B transients

K_{Ic} and K_{Ia} Values from PTSE-1 are Close to Fracture-Toughness Values from Small Specimens; Results are also Consistent with Other Large-Specimen Data

Comparison with small specimen data

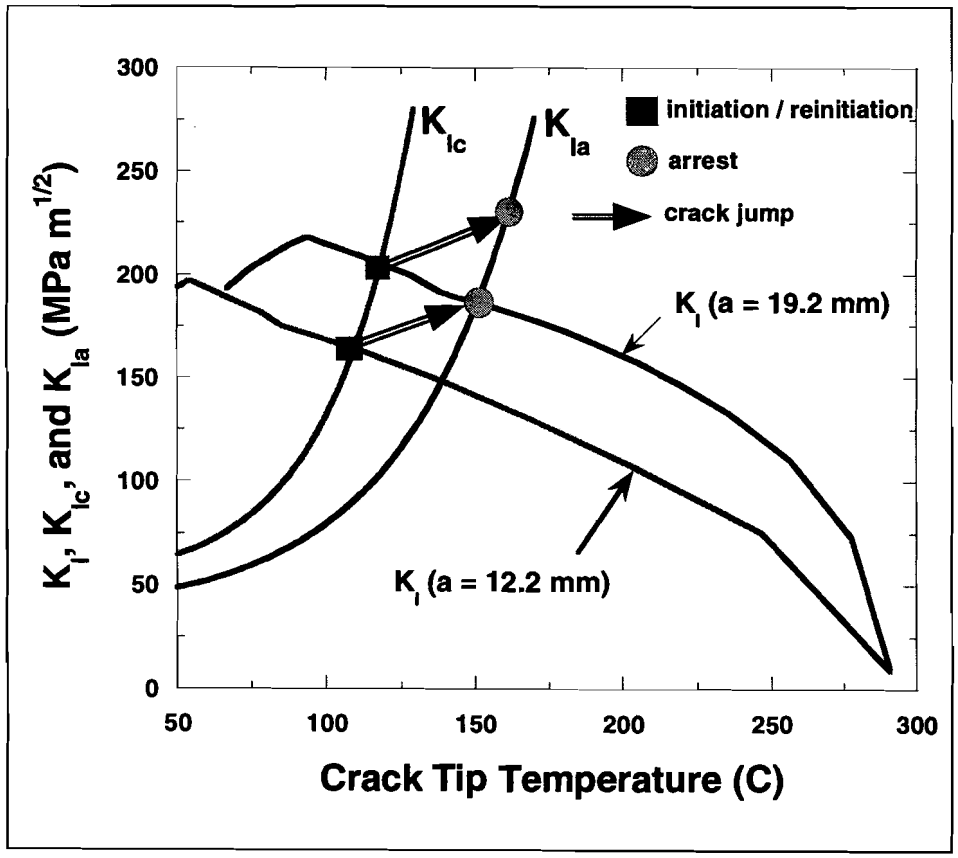


Comparison with other large specimen data

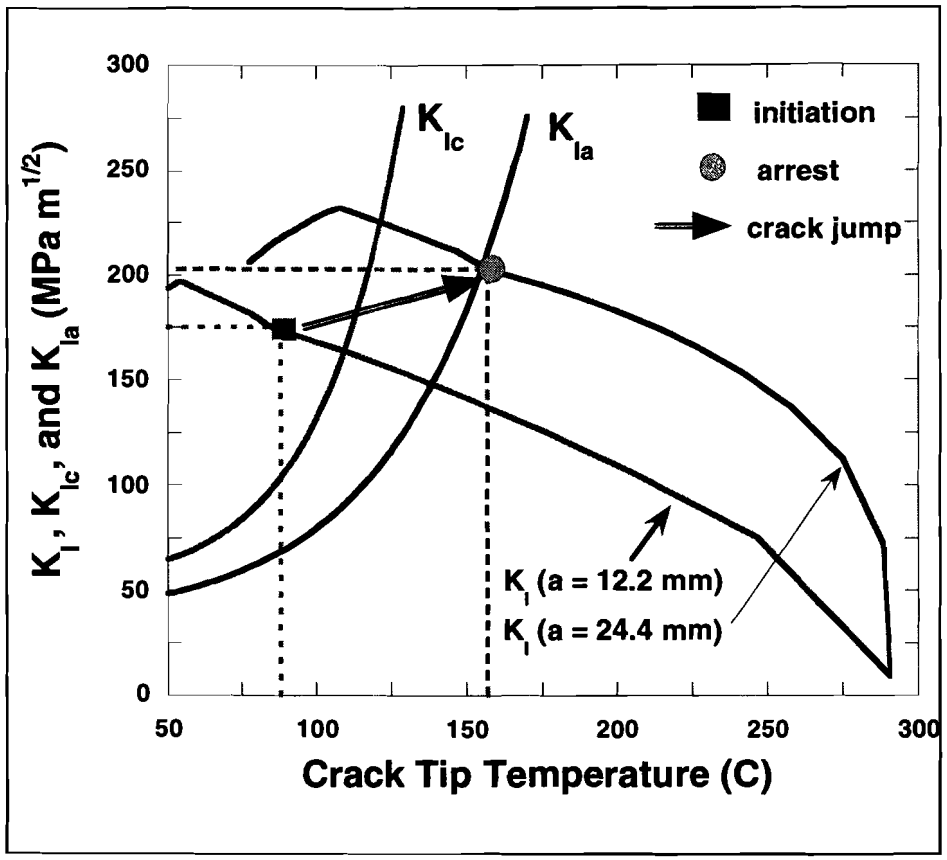


FAVOR Analysis of PTSE-1B Transient Produced Results in Good Agreement with Original Posttest Interpretations

"Classic" LEFM Prediction



FAVOR Assessment



Accomplishments and Conclusions from Large-Scale (ITVs, TSEs, and PTSEs) Experiments

- Cleavage fracture was observed in these large-scale tests consistent with the implications of small-specimen data
- Warm prestressing inhibited cleavage-fracture initiation in these experiments where ($dK_I/dt < 0$)
- The observed cleavage-crack behavior in these thick-section experiments has been well described by LEFM methodology (as embodied in the OCA/FAVOR codes)
- Should a consensus develop for analyzing additional large-scale experiments using FAVOR , then the European network's "simulated PTS experiment" NESC-1 would be a good choice

Summary

- NRC research programs conducted have established
  - The calculational methodologies, and
  - The empirical data

  needed to enable assessments of RPV fracture resistance under both routine operation and accident conditions using linear elastic fracture mechanics (LEFM)
- LEFM predictions of crack initiation and failure agree well with the results of prototypic large scale RPV experiments, suggesting that LEFM is an appropriate methodology for use in assessments of RPV fracture resistance

Future: Need for Research

Motivation

- **Regulatory**
  - **Licensees making exemption requests based on new technologies, no systematic way to deal with these**
  - **Focus on minimizing vessel risk can elevate overall plant risk (e.g. restrictive HU/CD windows ↑ P(pump trip))**
- **Commercial**
  - **A de-regulated energy market produces different operating decisions than have been made historically**
  - **License extension to 60 years**

Activities

- **FAVOR^{EP} development**
  - **EPFM driving force w/ constraint adjustments**
  - **Master Curve**

  benefits ...
  - **Better accuracy for shallow flaws**
  - **Removes considerable material uncertainty characteristic of RT_{NDT}-based methods**
- **International projects**
  - **Large scale experiments in fracture mode transition (FALSIRE, NESC)**
  - **Predictive PFM calcs. (ICAS, PROSIR)**

(12)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
494	16	16.1

Title **PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES**

Authors
JOHN T. LARKINS

List of Documents Attached

16

Instructions to Preparer	From Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	JOHN T. LARKINS/

G:PlanPro(ACRS):ppmins.494
July 10, 2002

SUMMARY MINUTES OF THE
ACRS PLANNING AND PROCEDURES MEETING
TUESDAY, JULY 9, 2002

The ACRS Subcommittee on Planning and Procedures held a meeting on May 1, 2002, in Room T 2 B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 3:00 p.m. and adjourned at 4:30 p.m.

ATTENDEES

MEMBERS

G. Apostolakis
M. Bonaca
T. Kress

ACRS STAFF

J. T. Larkins
S. Bahadur
H. Larson
S. Duraiswamy
R. P. Savio
J. Gallo
S. Meador
G. Cronenberg
P. Boehnert (part-time)

NRC STAFF

I. Schoenfeld, OEDO (part-time)

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the July ACRS meeting

Member assignments and priorities for ACRS reports and letters for the July ACRS meeting are attached (pp. 7-9). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the July 2002 ACRS meeting be as shown in the attachment (pp. 7-9).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through October 2002 is attached (pp. 7- 9). The objectives are to:

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

During this session, the Subcommittee discussed and developed recommendations on the items that require Committee decision, which are included in Section II of the Future Activities list (pp.10-11).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate. The Committee should decide on the Subcommittee's recommendations on items in Section II of the Future Activities List.

3) Quadripartite Meeting Update

The Quadripartite meeting is scheduled to be held on October 23-25, 2002, in Berlin, Germany. On May 16, 2002, RSK sent the latest agenda (pp. 12-15) for this meeting.

As confirmed at the April meeting, the following technical papers have been prepared by cognizant members for discussion at the Quadripartite meeting:

- Safety Culture and Safety Management (Bonaca/Powers)
- Risk-Informed Regulation (Apostolakis/Kress)
- Thermal-Hydraulic Analysis and Code Issues (Wallis/Ransom)
- Stress Corrosion Cracks in Pressure Retaining Components in Nuclear Power Plants (Ford/Shack)
- Risk Analysis of Spent Fuel Storage (Kress/Powers)

In connection with the Quadripartite meeting trip, 9 members agreed to visit a MOX facility in France prior to the meeting in Berlin, Germany.

RECOMMENDATION

The Subcommittee recommends the following:

- The Committee should discuss and endorse the technical papers prepared by cognizant members during the July 2002 ACRS meeting. Subsequently, Dr. Larkins should send these papers to RSK.

- Dr. Larkins should keep the Committee informed of the travel arrangements to the MOX facility in France and to the Quadripartite meeting in Germany.

4) Celebration of the 500th ACRS Meeting

As agreed to by the members, invitations were sent to the NRC Commissioners to participate at the 500th ACRS meeting ceremony, which is scheduled for March 4-5, 2003. (This is also coincidental with the Committee's 50th Anniversary.) So far, NRC Chairman Meserve, Commissioner Dicus, and Commissioner McGaffigan, as well as Bill Travers, EDO, have agreed to participate. Invitations have also been sent to those who are expected to serve as panel members.

Drs. Hal Lewis, Robert Seale, Bill Stratton, Mr. Dave Ward, and Dr. David Okrent (health permitting), have agreed to participate in the celebration.

As decided by the Committee at the June 2002 meeting, invitations have been sent to Mr. Ralph Beedle, NEI, and Mr. Bert Wolf, GE, to participate in the celebration

RECOMMENDATION

The Subcommittee recommends that the Executive Director keep the Committee informed of further developments.

5) Workshop on Nuclear Regulatory Decisionmaking Process in Switzerland

The Swiss Federal Safety Inspectorate (HSK), in coordination with the International Atomic Energy Agency (IAEA), is organizing a Workshop on Nuclear Regulatory Decisionmaking Process (pp. 16-22) to be held in Switzerland on October 13-16, 2002. As a result of the Planning and Procedures Subcommittee discussion in April, the Committee asked the Executive Director to contact HSK to see if the workshop on Nuclear Regulatory Decisionmaking Process could be scheduled for October 16-18, 2002, instead of October 13-16. The HSK has accommodated our request. Additionally, they have included ACRS participation in 2 places in the agenda. One would be a presentation on ACRS Perspective on Regulatory Decisionmaking and the second is a Panel Discussion. Both are on Wednesday, October 16, 2002. At the May meeting, the Committee had approved Drs. Bonaca, Kress, and Ransom to participate at this Workshop. In addition, the Committee stated that Dr. Kress, with assistance from Dr. Apostolakis, should prepare a paper, for Committee review and presentation at the workshop, on the ACRS Perspectives on Regulatory Decisionmaking and that Dr. Bonaca should represent the ACRS on the Panel discussion on Issues and

Opportunities for the Evaluation and Improvement of Nuclear Regulatory Decisionmaking Process. Dr. Kress plans to provide a draft paper for Committee review at the September ACRS meeting. A proposed outline of points prepared by Dr. Bonaca is attached (pp 23-26).

RECOMMENDATION

The Subcommittee recommends that Dr. Kress submit his paper to the Committee for review and endorsement during the September ACRS meeting and that the Committee provide feedback to Dr. Bonaca on the outline of points that he plans to make at the Panel discussion.

6) Role and Use of PRA in the ACRS Review Process

During its June 2002 meeting, the Committee discussed the issues raised by several members regarding the role and use of PRA in the ACRS review of regulatory issues (e.g., power uprates, license renewal, etc.). The Committee decided that a "White Paper" addressing the role and use of PRA in the ACRS review process would be helpful, and suggested that the ACRS Executive Director get a specialist under contract to prepare a draft "White Paper." Accordingly, paper work is being processed to obtain a specialist under contract to complete a draft "White Paper" by October 2002.

Barbara Jo White of the ACRS staff has forwarded a statement of work to the Division of Contracts and is working with a contract specialist to expedite getting a contract issued.

RECOMMENDATION

The Subcommittee recommends that the ACRS management keep the Committee informed of further developments.

7) EDO Responses to ACRS Reports

During the June 2002 meeting, the Committee discussed the EDO response of May 8, 2002 (pp. 27-28) to the comments and recommendations included in the ACRS report, dated March 19, 2002 regarding Risk-Informing Special Treatment Requirements of 10 CFR Part 50 (Option 2) (pp. 29-33). In addition, the Committee discussed the May 6, 2002 EDO response (pp. 34-38) to the ACRS report of March 14, 2002 (pp. 39-43) regarding Arkansas Nuclear One, Unit 2 Extended Power Uprate. In both cases, members were not satisfied with certain aspects of the EDO response and decided to continue discussion of this matter during the July ACRS meeting. Dr. Apostolakis agreed to propose a course of action after further discussion of this matter at the July meeting.

RECOMMENDATION

The Subcommittee agreed with the proposal by Dr. Apostolakis that:

- The Committee write a letter to the EDO, addressing those aspects of the EDO response to the ACRS report on Arkansas Nuclear One, Unit 2 Extended Power Uprate that the Committee does not agree.

- Regarding the EDO response to the ACRS report on Risk-Informing Special Treatment Requirements of 10 CFR Part 50, there is no need to write a separate letter; however, some of those issues that the Committee disagrees with the EDO could be addressed in the ACRS letter on the draft final revision 1 to Regulatory Guide 1.174 and SRP Chapter 19.

8) Subcommittee Report

The Thermal-Hydraulic Phenomena Subcommittee held a meeting on June 26, 2002. The Subcommittee discussed: (1) selected Office of Nuclear Regulatory Research's (RES) thermal-hydraulic research programs; to wit: the Phase Separation Research Program being conducted at Oregon State University (OSU), the RES TRAC-M code consolidation and documentation programs, and the Rod Bundle Heat Transfer test Program; and (2) proposed resolution of Generic Safety Issue (GSI)-185, "Control of Recriticality Following Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors."

Regarding GSI-185, the Subcommittee recommended that RES perform additional work to verify the thermal-hydraulic modeling assumptions pertaining to the "ex-vessel mixing" methodology employed to determine the extent of boron dilution seen in the vessel for the case of natural circulation flow.

Regarding the OSU Phase Separation Research Program, the Subcommittee agreed that additional work is necessary. Specifically, RES needs to verify the applicability to a full-scale plant of the correlation developed by OSU for modeling of entrainment in a horizontal pipe with an upward-oriented branch line.

RECOMMENDATION

The Subcommittee recommends that Dr. Wallis provide a brief report to the full Committee during the July meeting regarding matters discussed along with the concerns and issues raised by the Subcommittee and future course of action.

9) Report on the Visit to Watts Bar Nuclear Plant and Region II

Members of the ACRS Subcommittees on Plant Operations and on Fire Protection toured the Watts Bar Nuclear Plant on June 18 and held a meeting with Region II personnel on June 19, 2002, to discuss operational and fire protection issues. The Planning and Procedures Subcommittee believes that a report by the Chairman of the joint Subcommittees on Plant Operations and Fire Protection would be helpful to the members and the ACRS staff who did not participate in the tour or the meeting.

RECOMMENDATION

The Subcommittee recommends that Mr. Rosen provide a report to the Committee regarding the Watts Bar tour and meeting with the Region II personnel, with emphasis on concerns and issues brought to the attention of the Subcommittees both by the licensee and the Region II personnel and any follow-up items resulting from this tour and the meeting.

10) Members Projected to Exceed the 130 Legal Day Limit

Recently, there have been issues raised regarding some members exceeding the 130 legal day limit. Because members are classified as "Special Government Employees," we are required by law to plan work so that each member stays within the 130 legal day limit. After reviewing the compensation records, it appears that 4 of the 11 members may exceed the 130 legal day limit. There may be unique situations which require a member to exceed the limit, however, the "Special Government Employee" status of the member will change after exceeding the 130 legal day limit. This will have impact on member's outside consulting/contract activities.

RECOMMENDATION

The Subcommittee recommends that the workload for the 4 members be reviewed and to the extent feasible reduced so that these members do not exceed 130 legal day limit. Members who are projected to have fewer than 130 days could perhaps pick up some of the workload.

11) Member Time and Labor

As discussed at the last Planning and Procedures Subcommittee meeting, reporting compensation time by labor category is critical to the office operation. The members must use the agency's labor accounting codes when reporting their time.

RECOMMENDATION

The Subcommittee recommends that the codes and labor categories be provided in alphabetical order for use by members when filling out their compensation sheet. In the case of Full Committee meetings, each member will be provided with a completed sheet reflecting the meeting categories, which the member will then submit with the appropriate compensation voucher. All Subcommittee meetings and other work will be broken out by using the new labor codes by the member.

12) Member Issue

Travel Request

- Mr. Rosen has requested Committee approval to attend the NEI Fire Protection Forum scheduled to be held on August 28-30, 2002 in Seattle, Washington, at the Hotel Monaco (pp. 44-50).

RECOMMENDATION

The Subcommittee recommends that the Committee approve Mr. Rosen's travel request and that Mr. Rosen prepare a trip report summarizing the proceedings of the Forum along with his observations.

6A

ANTICIPATED WORKLOAD
July 10-12, 2002

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	All Members	Larkins	Meeting with the NRC Commissioners (7/10, 2:00-4:00 p.m.)	--	--	--
		Cronenberg/Larson	Risk-Informed Regulation Implementation Plan	--		--
		Boehnert	Response to EDO response to the ACRS report on ANO.2 Power Uprate	B	To provide feedback	--
		Cronenberg/ Boehnert	Draft Final Reg. Guide 1.174 and SRP Chapter 19	A	To meet the CTM schedule	1
Ford		Cronenberg/ El-Zeftawy	Application of Fracture Mechanics Methods to Reactor Vessel Integrity Assessment	--	--	--
		Savio	Format and content for the 2003 ACRS report on the NRC Safety Research Program	--	--	--
Kress	Ford	Savio	PTS Reevaluation Project: Risk Acceptance Criteria	A	To meet the CTM schedule	1
		El-Zeftawy	Advanced Reactors Research Plan	B	To provide feedback	1
Powers	--	Kobetz/Duraiswamy	Overview of NRC Research Activities in the Seismic Area	--	--	--
Rosen	--	Weston	Subcommittee Report-Visit to Watts Bar and Region II	--	--	--

ANTICIPATED WORKLOAD
July 10-12, 2002 CONTINUED

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Wallis	Sieber	Boehnert	Development of Review Standard for Reviewing Core Power Uprate Applications **[Information Briefing]**	--		--
	--	Boehnert	Subcommittee Report on matters discussed during the June 26, 2002 T/H Phenomena Subcommittee meeting.	--	--	--

ANTICIPATED WORKLOAD
September 12-14, 2002

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Cronenberg/Larson	Risk-Informed 10 CFR Part 50 Pilot Program (Option 2)	A	To meet the CTM schedule	--
Leitch	Bonaca	Kobetzt/Duraiswamy	North Anna and Surry License Renewal Application-Subcommittee Report	--		--
Powers	--	El-Zeftawy/ Cronenberg	Human Reliability Analysis Research Plan	B	To provide feedback	--
Ransom	--	Boehnert	Proposed Resolution of GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs".	A	To meet the CTM schedule	--
Sieber	--	Boehnert	Virginia Class Nuclear Propulsion Plant Submarine Design **[CLOSED]**	A	To meet the CTM schedule	--
Wallis	--	Boehnert/Weston	Draft Final Regulatory Guide (DG-1096) and SRP Section Associated with NRC Code Reviews	A	To meet the CTM schedule	--

ANTICIPATED WORKLOAD
October 10-12, 2002

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Cronenberg/ Duraiswamy	Draft Final ANS Standard on External Events PRA Methodology	A	To meet the CTM schedule	--
Bonaca	Leitch	Kobetz/Duraiswamy	License Renewal Application for Catawba Units 1 and 2 and McGuire Units 1 and 2 - **Subcommittee Report**	--	--	--
Kress	--	El-Zeftawy	Proposed Resolution of GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident"	A	To meet the CTM schedule	--
Powers	--	Kobetz/Duraiswamy	Proposed ANS Standard on Low-Power and Shutdown Operations PRA	A	To meet the CTM schedule	–
Ransom	Wallis	Boehnert	Beaver Valley Amendment Request to Convert to Atmospheric Containments **(Tentative)**	A	To support Licensing Review	--
Shack	--	Cronenberg/Larson	Risk-Informing 10CFR 50.46	A	To meet the CTM schedule	--
Wallis	--	Boehnert	Framatone ANP Richland, Inc. S-RELAP5 Realistic Large-Break LOCA Code	A	To support Licensing Review	--

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II. ITEMS REQUIRING COMMITTEE ACTION

1. Guidance for Performance-Based Regulation, (Open) (GEA/AWC) ESTIMATED TIME: 1 hour

Purpose: Determine a Course of Action

Review Requested by the NRC Staff [P. Kadambi, RES]. The staff has developed a draft NUREG/BR document entitled "Guidance For Performance-Based Regulation". This guidance document provides a process for developing a performance-based alternative for regulatory decision-making. Management Directive 6.3 (Rulemaking) calls for consideration of such a performance-based alternative to regulations. The process is set up to develop answers to questions which, in turn, provide the information to formulate a performance-based regulatory alternative that can be compared against others in a management review process. The five steps in the process are: (1) Defining the Regulatory Issue and its Context; (2) Identifying the Safety Functions; (3) Identifying Safety Margins; (4) Selecting Performance Parameters and Criteria; and (5) Formulating a Performance-Based Regulatory Alternative.

The staff requests a Subcommittee meeting (Reliability & PRA) in August and a full Committee review in September 2002.

The Planning and Procedures Subcommittee recommends that Dr. Apostolakis propose a course of action.

2. Draft Regulatory Guide DG-1119 (Proposed Revision 1 to Regulatory Guide 1.180), "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems" (Open) (GML/SD) ESTIMATED TIME: 1 hour

Purpose: Determine a Course of Action

Review Requested by the NRC Staff [C. Antonescu, RES]. The Staff has provided the ACRS with a copy of the DG-1119 (Proposed Revision 1 to Regulatory Guide 1.180.) The staff requests that the ACRS review the draft final version of this Guide after reconciliation of public comments.

Mr. Leitch recommends that the Committee review the draft final version of this Guide after reconciliation of public comments.

3. D.C. Cook Switchyard Fire (Open) (JDS/MWW)
ESTIMATED TIME: 1 hour

Purpose: Determine a Course of Action

On June 12, 2002, D. C. Cook Unit 1 was at about 68 percent power and increasing as a result of a refueling outage. Unit 2 was at 100 percent power.

At about 1:45 p.m., the Unit 1 345 kV switchyard breaker (L breaker) opened due to a fault. A catastrophic failure of the breaker and subsequent fire resulted in two additional breaker failures and a minor personnel injury. The licensee opened other breakers to isolate the fault. The fire was allowed to extinguish itself. The breaker fault resulted in a loss of all offsite power for both units. The units' electrical loads are currently being provided by the their respective unit auxiliary transformers. Both units remain at their initial power levels. The emergency diesel generators were not required to be started and remain in standby.

At 2:05 p.m., the licensee declared an "ALERT - Fire or Explosion Affecting Operability of Safety Equipment." Onsite and offsite emergency response facilities were activated. At 2:47 p.m. Region III and Headquarter Offices entered the Monitoring Phase for the ALERT.

As a result of the loss of the preferred reserve offsite power source for both units and in conjunction with an inoperable Unit 2 service water pump, which was undergoing planned maintenance, the licensee entered a Limiting Condition of Operation of its Technical Specifications which required the restoration of the service water pump or the shutdown of both units by 9:45 p.m. on June 12, 2002.

The licensee requested, and was granted a Notice of Enforcement Discretion (NOED) to allow the licensee to maintain operation of both units for an additional 10 hours while the licensee worked to restore the service water pump. At about 10:00 p.m., the licensee restored the service water pump to operable status. At 11:47 p.m., the licensee restored the preferred reserve offsite power source for both units, which satisfied the licensee's Technical Specifications requirements. The licensee subsequently terminated the ALERT classification at 11:54 p.m. and exited its Emergency Plan. At 11:55 p.m., the NRC exited the Monitoring Phase of Normal Mode. As of June 13, 2002, the licensee had restored four of six normal offsite power sources to the Unit 1 345-kV switchyard and was evaluating cause of the incident.

On June 13, 2002, the NRC Region III Office dispatched an inspector to the site to review the event with onsite inspectors. This information is current as of June 13, 2002.

The Subcommittee recommends that a briefing on this matter be scheduled and that Mr. Sieber provide his views.

PROPOSED CHANGES BY ACRS dtd. 6/11/02

RSK-Geschäftsstelle
beim Bundesamt für Strahlenschutz

16.05.2002

DRAFT

Quadripartite Meeting of the National Advisory Committees of France, Germany, Japan and USA

France: Groupe Permanent "Reacteurs" (GPR),
Groupe Permanent "Dechets" (GPD),
Groupe Permanent "Transports" (GPT),
Groupe Permanent "USINES" (GPU)

Germany: Reaktor-Sicherheitskommission (RSK)

Japan: Nuclear Safety Commission (NSC)

USA: Advisory Committee on Reactor Safeguards (ACRS),
Advisory Committee on Nuclear Waste (ACNW)

and invited representatives of Swiss and Swedish advisory committees

October 23 – 25, 2002, Berlin (Germany)
Maritim pro Arte Hotel Berlin

Wednesday, October 23

09:00 **Opening Remarks** — Sailer(RSK-Chairman), Renneberg(BMU)[1]

09:15 **Session 1**
Response to terrorist attack
Main topic: generic issues of commercial airplane **crash**
Chairman: N. N. (N. N.)
Co-Chairman: Schneider (RSK)
announced papers: Schneider
member GPR depending on briefing June 6, 2002

[1] Federal Ministry for the Environment, Nature Conservation and Nuclear Safety, Germany

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10.30 Coffee Break

11:00 Session 2
Safety Culture and Safety Management
Chairman: ~~N. N. (N. N.)~~ George Apostolakis (ACRS)
Co-Chairman: Wieland (RSK)
announced papers: Wieland (Safety Culture: Requirements, Indicators, Assessment and Reporting)
~~member~~ ACRS Mario Bonaca/Dana Powers, "Safty Culture and Safety Management"
member GPR depending on briefing June 6, 2002

12:30 Lunch

14:00 Session 3:
PSA/PSR/Risk informed Regulation
Chairman: ~~Apostolakis~~ Thomas Kress (ACRS)
Co-Chairman: Hahn (RSK)
announced papers: Hahn (Role of PSA, Periodic Safety Review, Risk Considerations in Regulatory Process, Potential and Limitations of Risk Informed Regulation)
Apostolakis/Kress (ACRS), "Risk-Informed Regulation"
member GPR depending on briefing June 6, 2002

16:00 Guided tour of Berlin or tour to Castle Sanssouci (Potsdam)

Thursday, October 24

09:00 Session 4
Thermal Hydraulic Analysis and Code Issues
Chairman: Wallis (ACRS)
Co-Chairman: Kersting (RSK)
announced papers: Kersting (Challenges to Development, Assessment and Application of Thermo-Hydraulic Codes in Germany)
Wallis/Ransom (ACRS), " Thermal- Hydraulic Analysis and Code"
member GPR depending on briefing June 6, 2002

10:30 Coffee Break

11:00 Session 5
Stress Corrosion Cracks in Pressure Retaining Components in NPP
Chairman: Speidel (RSK)
Co-Chairman: Schuler(GPR)
announced papers: Speidel
~~member~~(ACRS) Ford/Shack, " Stress Corrosion Cracks in Pressure Retaining Components in Nuclear Power Plants"
member GPR depending on briefing June 6, 2002

12:30 Lunch

14:00 Actual issues (incidents Brunsbuettel, Hamaoka, Davis Besse; fuel element damage at Cattenom; ...)

15:30 Coffee Break

16:00 Summary and Conclusion of sessions 1 – 5, preparation for plenary session ← Possibly ACNW Members will Join the meeting

19:00 Dinner on invitation of RSK

Friday, October 25

09:00 Session 6
Safety of Spent Fuel Storage
Chairman: Guillaumont(GPD)
Co-Chairman: Thomas (RSK)
announced papers: Guillaumont
Sailer (Dry Interim Storage of Spent Fuel Elements in Storage Casks)
~~member~~ (ACRS)(Risk Analysis of Spent Fuel Storage)
Kress/Powers

10:30 Coffee Break

11:00 Session 7
Waste Disposal Concepts; Performance Assessment for the Disposal; Safety Assessment of Final Repositories
Chairman: Thomas (RSK)
Co-Chairman: Devillers(GPD)
announced papers: Storck/RSK-subcommittee (Status of German Research for Final Disposal)
member GPD
~~probably member~~(ACNW), "Total System Performance Assessment
Garrick

12:30 Lunch

14:00 Session 8
Transport of Spent Fuel and Waste
Hornberger would like to chair Session 7, if possible.
Chairman: Hornberger(ACNW)
Co-Chairman: Sailer (RSK)
announced papers: Drotleff/RSK-subcommittee (amended regulations, experience)
~~probably member~~ ACNW (TBD)
Levenson

15:30 Coffee Break

16:00 Plenary Session
- **Summary and Conclusion**
- **Final Remarks**

Chairman: Sailer(RSK)
Co-Chairman: N. N. (N. N.)

17:00 **End of Meeting**

From: Zünd Marianne <Marianne.Zuend@hsk.psi.ch>
To: "'Larkins John'" <JTL@nrc.gov>
Date: 4/25/02 7:14AM
Subject: Invitation Workshop in Switzerland 15-18 October 2002

Dear Mr. Larkins

It is a great pleasure for us to invite you or a representative(s) of your organisation to the

Joint HSK-IAEA-NEA Workshop on Regulatory Decision Making Processes
Switzerland, 15 - 18 October 2002

HSK, the Swiss Federal Nuclear Safety Inspectorate will act as the host of the event, with the support of the International Atomic Energy Agency IAEA and the OECD Nuclear Energy Agency NEA. The Workshop will be held at the Grandhotel Giessbach in the beautiful Bernese Alps.

Please find enclosed the announcement, and the preliminary agenda of the Workshop as well as the registration form (Word-Doc).

May we please ask for your reply to this invitation by sending the registration form to the HSK Workshop Secretariat by 31 May 2002 at the latest.

Regarding the practical arrangements, please see the enclosed document. Further information will be provided to you together with the confirmation of your registration.

If you have any queries about the workshop, please do not hesitate to contact Mr. Sabyasachi Chakraborty, who is the chair of the organising committee (chakraborty@hsk.psi.ch, Tel. +41-56-310 39 36).

We look forward to welcoming you to a fruitful Workshop.

Sabyasachi Chakraborty
Marianne Zuend

Marianne Zuend
HSK - Swiss Federal Nuclear Safety Inspectorate
Safety Research and International Affairs
CH-5232 Villigen-HSK
marianne zuend@hsk.psi.ch
Tel +41-56-310 39 76 / Fax +41-56-310 49 76

Item 5
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HSK in cooperation with IAEA and NEA

Workshop on Nuclear Regulatory Decision Making Processes

Switzerland, 15 – 18 October 2002

Registration Form

Please complete each section of this Registration form and send by 31st May 2002 to:

Marianne Zuend
HSK
CH-5232 Villigen-HSK
E-mail: zuend@hsk.psi.ch
Phone: +41-56-310 39 76
Fax: +41-56-310 49 76

Title: .....
First Name: .....
Last Name: .....
Badge Name: .....
Organisation/Institution: .....
Address: .....
City: .....
Postal Code: .....
Country: .....
E-mail address: .....
Telephone: ..... *(incl. country code number)*
Fax Number: ..... *(incl. country code number)*

Any special dietary requirements? (e.g. vegetarian): .....

Accomodation: *(please underline the appropriate)*
Single room / Double room (please indicate name of 2nd person)

Your remarks / special requests: .....

The costs per person will be around 700 Swiss Francs or approx. 475 Euro to be paid individually and directly to the hotel. This amount includes accomodation in the Grandhotel Giessbach for 3 nights (Tuesday through Friday) as well as all meals (breakfast, morning & afternoon coffee, lunches and dinners), and conference materials. Travel costs and personal expenses (e.g. mini-bar consumptions) are not included.

Confirmation of your registration and accommodation booking as well as travel information in Switzerland will be sent to you via e-mail. Participants should await confirmation from the registration office before committing to other travel arrangements.

Federal Office of Energy (Switzerland)
Bundesamt für Energie (Schweiz)
Office fédéral de l'énergie (Suisse)
Ufficio federale dell'energia (Svizzera)

HSK

Swiss Federal Nuclear Safety Inspectorate
Hauptabteilung für die Sicherheit der Kernanlagen
Division principale de la Sécurité des Installations Nucléaires
Divisione principale della Sicurezza degli Impianti Nucleari

Preliminary Version

HSK in cooperation with IAEA and NEA

Workshop on Regulatory Decision Making Processes

Switzerland, 15 – 18 October 2002

Restructuring of the nuclear industry due to market deregulation, the increasing competition that forces nuclear power plant operators to enhance efficiency, the long-term extension of operating licenses and plant lifetimes, urgent decommissioning safety issues, and the debate on phasing out nuclear power in several countries are challenging nuclear regulation today. Moreover, the constantly increasing scientific and technical knowledge is leading to changes in regulatory requirements, and is at the same time providing improved and new methods and technologies for nuclear regulation.

In many countries, regulatory decision making processes are continuously being reviewed in response to the developments mentioned above. Following the latest US NRC developments some countries are embarking on implementation of "risk informed regulations". Optimizing regulatory body efficiency and effectiveness seems to be one of the objectives driving most of the regulatory bodies that have decided to apply risk informed regulation. Caution and fear of misinterpretation or overwhelming contribution of the PSA results in decision making are the main reasons why other regulatory bodies hesitate to consider or even reject the "risk informed regulation" concept.

IAEA and NEA have recognized the importance of revaluating nuclear regulatory activities in several working groups and publications. However, as of today, there is no comprehensive approach in terms of guidelines/best practices on how the regulatory decision making processes need to be refined and how they can be systemized.

This workshop aims at sharing information on approaches and methods for regulatory decision making (nuclear, aviation, space, food safety). Through presentations and panel discussions, attendees will illuminate issues, review the capacity of the current regulatory decision making process, discuss the implications of new technical and scientific knowledge for regulatory practices, and debate differences between countries. Thus, the workshop offers a good opportunity to assess in an open exchange the pending regulatory decision making issues, which have a direct impact on the safety of nuclear facilities and on the competitiveness of nuclear power plant operators.

The workshop results will be retained in the form of a written report which reflect the opinions expressed by the participants in the form of recommendations for the establishment of a regulatory decision making guidance document.

Telephone +41 56/310 38 11
Fax +41 56/310 39 07

Postadresse
adresse postale
indirizzo postale
postal address

Hauptabteilung für die Sicherheit der Kernanlagen
CH-5232 Villigen-HSK

Some of the issues that will be addressed in the workshop presentations, discussions and working groups will be:

- Regulatory decision making in circumstances of uncertainty – the precautionary approach
- Experience gained from new approaches from non-nuclear sectors which could be useful to the nuclear regulatory decision making processes
- Discussion of real decision making examples: basis and processes in practice
- Setting of priorities in regulatory decision-making
- Consideration of costs in regulatory decision-making
- Impact of risk informed decision making on regulatory inspection and assessment programmes
- Safety philosophy for adequate safety and protection, and risk-informed regulation, safety performance indicators
- How to create regulatory decision-making processes that are transparent, open, credible and accountable to the public
- Communication and dialogue with the public in regulatory decision making
- Engagement of stakeholders in decision-making processes
- Feedback and monitoring mechanism for regulatory decison making
- Clarification of the role of external experts in regulatory decision making processes

Workshop Goal and Objectives

This 3-day invitational workshop is designed to advance the discussion on nuclear regulatory decision making processes.

The goal of the workshop is to develop recommendations to the attention of the IAEA and the NEA to foster the identification and elaboration of effective guidance for nuclear regulatory decision making processes. The recommendations will address the legal, institutional, organizational, and financial factors that influence the regulatory decision making processes in the different participating countries.

Specific Objectives

- Refine the understanding of regulatory decision making processes by reviewing the current approaches, methods and tools
- Discuss concrete examples of regulatory decision making: what is the basis and which are the processes that lead to a decision?
- Collect information on national experience with "risk informed decisions" taken by regulatory bodies
- Identify criteria for activities and approaches that will have a positive impact on regulatory decision making processes
- Publish a post-workshop report that identifies recommendations for the improvement of nuclear regularory decision making processes.

Organizing committee

The organizing committee for the workshop will be chaired by S. Chakraborty, HSK, and will comprise representatives of the IAEA, the NRC and the NEA. Members of the organizing committee will help to complete the program, to create and disseminate the invitation and will also assist in drafting and reviewing of the workshop report.

Location, date, and organization

The workshop will be held in Switzerland in the Grandhotel Giessbach in the beautiful Bernese Oberland (http://www.giessbach.com).

Date: **Tuesday, 15 October, 2002 - Friday, 18 October, 2002**

The workshop will begin with the arrival of the participants and dinner on Tuesday evening 15 October, and will run from Wednesday morning through Friday noon.

Approximately 40 people will be invited to the workshop. Delegates from Eastern European countries are particularly welcomed.

The costs per person will be around 700 Swiss Francs or approx. 475 Euro (3 nights in a single room including all meals) to be paid individually and directly to the hotel at check-out on Friday 18th October 2002.

Workshop agenda

The organizing committee will devise the detailed and final workshop agenda on the basis of the preliminary programme as outlined on the next pages (as of 25th April 2002). Please note that the preliminary programme is subject to change.

Registration

Please register by filling in and returning the attached registration form by e-mail or fax to HSK:

Ms. Marianne Zuend
zuend@hsk.psi.ch
Tel. +41-56-310 39 76
Fax +41-56-310 49 76

Registration is required by 31st May 2002 at the latest.

HSK in cooperation with IAEA and NEA

Workshop on Nuclear Regulatory Decision Making Processes

Switzerland, 15 – 18 October 2002

Preliminary Programme

(version of 25th April 2002)

Tuesday, 15th October 2002

Afternoon	Individual arrival of participants by train or car Transfer by hotel bus from railway station Brienz to the Grandhotel Giessbach http://www.giessbach.com	
19:00	Dinner at the Grandhotel Giessbach Restaurant	

Wednesday, 16th October 2002

08.30 – 09.15	***Welcome address by HSK, IAEA and NEA*** Layout of workshop goals	presenters to be determined
09.20 – 10.20	***IAEA activities in the field of risk-informed decision making and risk-informed regulation*** Ms. Vesselina Ranguelova, IAEA	Working title
10.20 – 10:50	Coffee Break	
10:55 – 11:55	***U.S. NRC activities in the field of regulatory decision making*** Ashok C. Thadani, Director of Office of Nuclear Regulatory Research, Nuclear Regulatory Commission	
12:00	Lunch at the Grandhotel Giessbach Restaurant	
14:00 – 15:00	***U.S. ACRS perspectives on regulatory decision making*** ??, Advisory Committee on Reactor Safeguards	Presenter and title to be determined
15:05 – 15:40	***Practical Example: Decision making in regulatory licensing processes at HAEA (license renewal, major safety upgrading)*** Lajos Vöröss, Hungarian Atomic Energy Authority	Working title
15:40 – 16:10	Coffee Break	
16:15 – 16:50	***Practical Example: Communication and dialogue with the public in regulatory decision making in SNRCU*** Sergiy Peleshenko, State Nuclear Regulatory Committee of Ukraine	Working title
16:55 – 17:30	***Practical Example: Process of regulatory decision making at SNSA*** ??, Slovenian Nuclear Safety Administration	Presenter and title to be determined
17:35 – 18.30	***Panel discussion:*** Issues and opportunities for the evaluation and improvement of nuclear regulatory decision making processes Chairman: U. Schmocker (Director HSK). Panel participants: V. Ranguelova (IAEA), B. Kaufer (NEA), V. Gryschenko (Chairman SNRCU), L. Vöröss (Director HAEA), Representatives of US NRC and US ACRS	
19:00	Dinner at the Grandhotel Giessbach Restaurant	

continued on next page

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Thursday, 17th October 2002

Time	Activity	Remarks
08:30 – 09:00	***Long-term radioactive waste management: challenges and approaches to regulatory decision making*** Thomas Flüeler, Swiss Federal Nuclear Safety Commission KSA	Working title
09:05 – 09:45	***Decision making processes in ESA's flight programmes*** Christian Preyssl, European Space Agency	Working title
09:45 – 10:15	Coffee Break	
10:20 – 11:00	***Application of the system safety process within the FAA's aviation safety inspector workforce*** Geoffrey R. McIntyre, US Federal Aviation Administration FAA	Working title
11:05 – 11:45	***Approaches of the Food Standards Agency for involving consumers and other stakeholders in the regulatory processes on food safety*** Mr. Richard Burt, UK Food Standards Agency	Working title
11:50 – 12:30	***OECD-NEA activities in the field of regulatory decision making*** Mr. Barry Kaufer, OECD-NEA	Working title
12:40	Lunch at the Grandhotel Giessbach Restaurant	
14:00 – 16:00	***Take up of work in 2 working groups*** : identify issues, characteristics and opportunities, and formulate recommendations for the attention of the IAEA and the NEA to foster the identification and elaboration of effective guidance for regulatory decision making processes. Chairman of working group 1: Geoffrey R. McIntyre Chairwoman of working group 2: Vesselina Ranguelova	
16:00 – 16:30	Coffee Break	
16:35 – 17:30	***Continuation of work in 2 working groups***	
18:15	***Cocktail Reception*** offered by HSK	
19:15	Dinner with ***invited presentation*** at the Grandhotel Giessbach Restaurant	Invited presenter to be determined

Friday, 18th October 2002

Time	Activity	Remarks
08.30 – 09.10	***Presentation of results of working group 1***	
09:15 – 09:55	***Presentation of results of working group 2***	
10:00 – 10:30	Coffee Break	
10:35 – 11:30	***Closing plenary discussion***: wrap up of presentations, concrete examples and working group results, next steps. Chairman: S. Chakraborty, HSK	
11:30	***Hotel Check-Out*** of participants	
12:15	Lunch at the Grandhotel Giessbach Restaurant	
Afternoon	Individual departure of participants by train or car Transfer by hotel bus from the Grandhotel Giessbach to the railway station Brienz.	

From: "Mario V. Bonaca" <mvbonaca@snet.net>
To: <SXD1@nrc.gov>
Date: 7/8/02 7:45AM
Subject: Bonaca's notes for panel discussion at the Swiss meeting on decision-making

Sam,

Attached are my (preliminary) notes for the panel discussion at the Swiss workshop in October.
You told me that we will discuss at the P&P.
Please include in the agenda for that discussion.

Thank you,

Mario

CC: <TSKress@aol.com>, "John Larkins" <JTL@nrc.gov>, "Howard Larson" <HJL@nrc.gov>, <apostola@MIT.EDU>

Item 5

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Workshop on Regulatory Decision Making Processes.

Switzerland, 15-18 October 2002

Panel Discussion: **Issues and opportunities for the evaluation and improvement of nuclear regulatory decision making processes**
Notes of M. Bonaca, ACRS

Efficient and effective nuclear safety regulation would impose requirements that are recognized by all stakeholders as necessary and sufficient to assure that plants can be operated with an acceptably small risk to the health and safety of the public. The process of developing such regulation would require the following steps:

1- "Acceptably small risk to the health and safety of the public" (safety goals) would need to be defined in terms of risk components, i.e., probability and consequences, and found to be acceptable to the majority of stakeholders,
2- Plant performance criteria (risk metrics) would have to be established that would demonstrate how plants meet the "acceptably small risk" criterion,
3- Regulatory requirements necessary and sufficient to assure that plants meet the established performance criteria would have to be developed,
4- Plants specific analyses would be required to show that plants meeting regulatory requirements meet the established plant performance criteria, and
5- In order to facilitate acceptance by all stakeholders, the decision-making process would be systematic and would include explicit consideration of possible options as regulatory requirements are proposed and selected.

In other words, the regulatory framework would need to be risk-informed to demonstrate necessity and sufficiency of requirements, and supported by a formal decision analysis to enhance communication and stakeholder acceptance.

But the existing US regulatory framework is neither risk-informed nor supported by a transparent decision-making process. The regulatory framework was developed before safety goals were established, and before necessity and sufficiency of requirements could be assessed with the help of probabilistic safety analysis (PSA). The large uncertainties confronting decision-makers in the early times of the technology resulted in the imposition of layers of defense-in-depth (DID) requirements. Some of these requirements are now being recognized as unnecessary or excessive, thus imposing unnecessary regulatory burden.

The TMI-2 accident exposed areas where existing requirements proved insufficient. Regulatory response to the lessons learned from the accident was to expand regulatory requirements and to verify their sufficiency, with significant help from PSA. But in the rush to fill the gaps in the regulation exposed by TMI-2, the test of necessity was not

attempted consistently. This resulted in additional regulatory burden that, in some cases, has been shown to be unnecessary and, occasionally, to actually degrade plant safety. An example of unnecessary regulatory burden imposed by a TMI action item is the implementation of hydrogen recombiners that are not needed for design basis accidents and are ineffective in dealing with severe accidents.

Recent NRC initiatives to risk-inform the regulation, such as Reg. Guide 1.174 and Option 3, are showing how effectively the inclusion of risk criteria can help improve regulatory effectiveness and efficiency ***(brief description of Reg. Guide 1.174 and Option 3.)*** The inclusion of risk criteria allows the open evaluation (and dialog with stakeholders) of alternate options to meet quantitative objectives set forth for any specific regulation. These criteria also allow to determine when no further action is necessary, or when existing regulation is excessive. The ongoing review of existing regulation, such as the ones governing loss of coolant accident (LOCA) and pressurized thermal shock (PTS), allows the explicit review and evaluation of individual elements of the rules and how they contribute to the achievement of quantitative performance criteria. They also identify, based on current knowledge, the degree of conservatism built in the rules. In the approach being implemented by the NRC, PSA insights are still subordinate to defense-in-depth requirements. This approach where the decision-making process integrates risk insights with other considerations, such as DID requirements, should reduce the concern of those who fear that PSA results may be abused.

Although the effort to risk-inform the current body of regulation is improving regulatory effectiveness and efficiency, there are practical limitations to how far the existing deterministic framework can be improved. This consideration makes a compelling case for the development from scratch of a technology-independent, risk-informed regulatory framework for the new generation of reactor designs that would include the steps listed above. The effort to risk inform current regulation is developing experience that should prove valuable in developing this new risk-informed framework.

The expanded use of PSA to support regulatory decisions has encountered resistance from some reactor operators and regulators alike. Plant operators have been concerned that the expanded use of PSA could result in the imposition of additional requirements. Regulators have been concerned that PSA results could be abused, or would lead to unwarranted relaxation of requirements, undermining the deterministic foundations of the current regulatory structure. These concerns do not recognize that regulatory requirements were always risk-informed. Reactor protection and safeguards requirements stemming from the graded approach of the accident analysis evidence the early attempt to design safeguards with capabilities commensurate to the risk-significance of design basis accidents they were designed to control and contain. The probabilistic components of risk used in the early times were coarse at best, but, often, so were the deterministic inputs. As we have expected deterministic criteria and analyses to be refined and improved through the years, so we should expect of the probabilistic criteria and methods used to design and regulate nuclear reactors. Current PSA methodology has developed to the point of becoming indispensable in the design, operation and performance evaluation of power

plants. Further progress requires its inclusion in the foundations of the regulation.

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

May 8, 2002

Dr. George E. Apostolakis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: PROPOSED RULEMAKING AND ASSOCIATED GUIDANCE ON RISK-INFORMING THE SPECIAL TREATMENT REQUIREMENTS OF 10 CFR PART 50 (OPTION 2)

Dear Dr. Apostolakis:

In your letter to me dated March 19, 2002, you provided the views of the Advisory Committee on Reactor Safeguards (ACRS) on the proposed rulemaking and guidance on risk-informing special treatment requirements of 10 CFR Part 50 (Option 2). The Committee made three recommendations. Our responses provided below reflect both your letter and the discussion held during the ACRS meetings of February 22, 2002, and March 7, 2002.

1. Criteria for Integrated Decision-making Panel (IDP)

The Committee recommended that the criteria used by the IDP for categorizing structures, systems, and components (SSCs) be made explicit. The staff agrees with this recommendation and made some comments on this subject in a letter to the Nuclear Energy Institute (NEI) dated February 8, 2002 (ADAMS Accession No.: ML020430301) on NEI 00-04, "Option 2 Implementation Guidance." The staff will discuss this topic further with the Committee after receiving the revised version of NEI 00-04.

2. Consideration of Additional Metrics

The Committee recommended the use of additional risk metrics such as late containment failure and inadvertent release of radioactive material and noted that categorization of SSCs with a more complete set of metrics might allow eliminating additional treatment requirements for SSCs in risk-informed safety class (RISC) 3. The staff agrees that consideration needs to be given to the issues of long term containment integrity and inadvertent release of radioactive materials (not associated with severe accidents), but is suggesting a different approach. Instead of developing metrics for these aspects, we are expecting NEI to incorporate guidance into NEI 00-04 that would enable the IDP to consider these risks in categorizing the safety significance of SSCs. Licensees could elect to address the issues through a quantitative approach; however, the staff's expectation is that long term containment integrity be considered from a qualitative perspective within the element of defense-in-depth. The staff is still considering the issue of inadvertent radioactive release.

3. Simplified Methods for Treatment of Uncertainties

The Committee stated that the rigor in treatment of uncertainties should be made consistent with current capabilities of probabilistic risk assessment (PRA) software and data, and that

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when simplified methods are used, comparison with more rigorous analyses should be made available to demonstrate the adequacy of these methods. The discussion section of the letter elaborates on this and makes it clear that the Committee's interest in uncertainty analysis has to do with developing importance measures and evaluating changes in core damage frequency (ΔCDF) and changes in large early release frequency (ΔLERF). The staff agrees that rigorous treatment of uncertainties may be the best way to evaluate these measures, but believes that licensees do not necessarily need this rigor, as long as they recognize the impact of the uncertainties on the results. Accordingly, rigor in decision-making can be maintained without the use of such formal methods. For example, in treating parameter uncertainty in the evaluation of ΔCDF and ΔLERF, the licensee is expected to follow Regulatory Guide (RG) 1.174, which states that if true mean values obtained by propagating uncertainty are not provided, the licensee should examine cutsets to demonstrate that the impact on the mean of the results of simply multiplying means of basic events is acceptable. The impact of model uncertainty would also need to be addressed by sensitivity studies. Because the ΔCDF and ΔLERF tests are the deciding factor in the decisions on Option 2 SSC categorization, the exactness of the importance measures is not, in the staff's opinion, a significant issue. To the staff's knowledge, the use of uncertainty analysis in the determination of importance measures is not a standard feature of PRA software packages.

In the February 8, 2002, letter to NEI documenting the staff's review of draft revision B of NEI-00-04, the staff commented that the factor to be used in the sensitivity studies for estimating ΔCDF and ΔLERF should have a supporting engineering assessment. The staff is currently interacting with external stakeholders to identify and resolve any outstanding Option 2 implementation issues. The factor issue will be addressed in the final version of the proposed rule.

We look forward to further discussions with the ACRS on the proposed rule and associated guidance.

Sincerely,

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commission McGaffigan
Commission Merrifield
SECY

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

March 19, 2002

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED RULEMAKING AND ASSOCIATED GUIDANCE FOR RISK-INFORMING THE SPECIAL TREATMENT REQUIREMENTS OF 10 CFR PART 50 (OPTION 2)

During the 490th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 2002, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss the proposed rulemaking and associated guidance for risk-informing the special treatment requirements of 10 CFR Part 50 (Option 2). We discussed the staff's draft rule language for 10 CFR 50.69 and proposed industry guidance in NEI 00-04, Revision B, "Option 2 Implementation Guideline." Our Subcommittee on Reliability and Probabilistic Risk Assessment discussed these matters during meetings on December 4, 2001, and February 22, 2002. We also had the benefit of the documents referenced. This report focuses primarily on the proposed industry guidance in NEI 00-04, Revision B.

Conclusion and Recommendations

1. The criteria used by the Integrated Decision-making Panel (IDP) for categorizing structures, systems, and components (SSCs) should be made explicit and should include consideration of risk metrics that supplement core damage frequency (CDF) and large early release frequency (LERF), such as late containment failure and inadvertent release of radioactive material.

2. Categorization of SSCs performed with a more complete set of risk metrics may allow the elimination of additional treatment requirements for components in the risk-informed safety class 3 (RISC-3) category (safety related, low safety significant).

3. The rigor in the treatment of uncertainties in probabilistic risk assessment (PRA) results should be made consistent with the current capabilities of PRA software and data. When simplified methods are used, comparison with more rigorous analyses should be available to demonstrate the adequacy of these methods.

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Discussion

The overall categorization process described in NEI 00-04, Revision B, relies heavily on the judgments of the IDP. The Panel's decision concerning the assignment of an SSC to a risk-informed safety class is based on a variety of qualitative and quantitative inputs. The quantitative inputs are produced by a PRA, if available. A large majority of SSCs are categorized without the benefit of quantitative inputs from a PRA. Two major elements of the categorization process are the risk-informed decision criteria and the processes used by the IDP in making judgments.

In our report dated October 12, 1999, we commented extensively on the decision-making process and the need for guidance and training in conducting expert-panel sessions. Our comments on the processes described in the then-proposed Appendix T to 10 CFR Part 50 remain valid and are a continuing concern. This report focuses on additional issues that warrant attention in the revision of NEI 00-04 to support the proposed 10 CFR 50.69 rulemaking.

The traditional criteria for evaluating risk significance use the metrics CDF and LERF. The initial screening of SSCs for which PRA results are available is carried out by using importance measures that are based on these two metrics. We believe that the probability of late containment failure should be added to CDF and LERF to provide a more complete characterization of risk.

In categorizing SSCs for which PRA results are unavailable, qualitative considerations serve as the primary basis for decisionmaking. Even when PRA results are available, the risk-informed approach requires that the IDP consider qualitative inputs based on defense in depth and safety margins, as articulated by the principles in Regulatory Guide 1.174. NEI 00-04, Revision B, provides very little guidance to assist the Panel in making these qualitative assessments. Explicit criteria should be developed for the qualitative categorization of SSCs and the decision-making process needs to be scrutable with results that can be documented. Guidance to accomplish this should be included in NEI 00-04.

The qualitative considerations used by the IDP should include defense in depth and the traditional graded approach in which relatively frequent events are intended to not fail any of the barriers to the release of radioactivity, but relatively infrequent events are allowed some fuel damage provided that the resulting release is limited by the requirements of 10 CFR Part 100. Specific guidance to the IDP could include requirements for the Panel to determine whether (1) the SSC supports a system that acts as a barrier to fission product release during severe accidents; (2) the SSC is relied upon in the emergency operating procedures or the severe accident management guidelines; and (3) failure of the SSC will result in the inadvertent release of radioactive material even in the absence of severe accident conditions.

If any of the above conditions are true, the IDP should consider including such SSCs in RISC-1 (safety related, safety significant) or RISC-2 (non-safety related, safety significant) category. The IDP could justify its conclusions in the risk categorization by demonstrating that one of the following conditions are met:

- Relaxing the requirements will have minimal impact on the failure rate increase.
- Showing that adequate data are available to demonstrate that failure modes that prevent the SSC from fulfilling its function are unlikely to occur.
- Such failure modes can be detected in a timely manner.

The choice of appropriate treatment for RISC-3 has been a difficult issue for staff and industry. We believe that much of this difficulty has arisen because the staff recognizes that risk concerns cannot be completely addressed by CDF and LERF and is, therefore, reluctant to relax some special treatment requirements. By explicitly addressing all risk concerns in the categorization process, as discussed above, it may be easier to obtain agreement that components assigned to RISC-3 do not require any treatment beyond "commercial practice."

We note that materials degradation is not directly assessed in NEI 00-04, Revision B. We believe that aging phenomena and the management of degradation must be considered in the IDP deliberations concerning affected SSCs and passive system components.

The use of risk information in regulatory decisionmaking is relatively new. Some within the NRC, the industry, and the public view this evolution with skepticism. The NRC Strategic Plan has established increasing public confidence as a performance goal. The use of rigorous methods to produce risk information is essential to achieving this goal.[1] In many instances, simplified methods can yield satisfactory results. It should be demonstrated, however, that these simplified methods yield results that are consistent with those provided by more rigorous methods and that their limitations are well understood.

In our reports dated October 12, 1999 and February 11, 2000, we commented extensively on the limitations of importance measures. The requirement to use sensitivity studies to determine ΔCDF and ΔLERF provides evidence that NEI 00-04, Revision B, recognizes the major limitation of importance measures, namely, their inability to determine the change in risk associated with a group of components. We

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[1]In his speech to the Regulatory Information Conference on March 5, 2002, Commissioner Diaz stated: "This is the year 2002, almost 30 years after WASH-1400, and it is time that all licensees have a quality Level 2 PRA so they can effectively utilize our regulatory processes."

believe that the IDP would benefit from an explicit identification and discussion of this and other limitations that have been identified in the literature (References 8 and 9).

NEI 00-04, Revision B, shies away from providing guidance or encouragement for licensees to perform uncertainty analyses and relies heavily on sensitivity studies that are substitutes for uncertainty analyses. Modern PRA tools make it relatively routine to perform a genuine uncertainty analysis, i.e., one that propagates the uncertainties in failure rates, and such analysis should be performed where possible.

The argument has been made that using mean values for the failure rates in performing the PRA and the screening is "good enough." We agree that, in the majority of cases, this argument may be true provided that mean values are indeed used, although relatively few investigations are available in the literature (References 8 and 11) to substantiate this claim. We object to the practice of taking arbitrary "point" values of the parameters and declaring them as mean values. Such practices do not contribute to the credibility of the categorization process.

One of the most significant limitations of importance measures is that they measure the impact of individual SSCs on risk, and, consequently, they cannot be used directly to estimate changes in risk for a group of SSCs. This limitation is recognized in NEI 00-04, Revision B, and additional sensitivity studies are suggested to attempt to assess the impact of changing treatment requirements on a group of components. In NEI 00-04, Revision B, it is suggested that the failure rates of RISC-3 SSCs be increased by factors ranging from 2 to 5 to evaluate changes in CDF and LERF. The current justification for this choice of values is weak, and a better justification is needed, especially since these factors are smaller than the factor of 10 used in the South Texas Project multiple exemption request. A distinction between parameter and model uncertainties would be very useful in this case.

We look forward to reviewing the draft final rule language and associated guidance as more progress is made.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft Rule Language to amend Title 10 of the *Code of Federal Regulations* (10 CFR) by adding Section 50.69, "Risk-Informed Treatment of Structures, Systems, and Components," dated November 19, 2001.

2. Nuclear Energy Institute, NEI 00-04, Draft Revision B, "Option 2 Implementation Guideline," May 2001.
3. Memorandum dated January 24, 2002, from Michael T. Markley, ACRS staff, to Cynthia Carpenter, Office of Nuclear Reactor Regulation, NRC, Subject: Questions on NEI 00-04, "Option 2 Implementation Guideline."
4. Letter dated February 8, 2002, from Cynthia A. Carpenter, Office of Nuclear Reactor Regulation, NRC, to Anthony R. Pietrangelo, NEI, Subject: NRC Staff Review of Draft Revision B of NEI 00-04, "Option 2 Implementation Guideline."
5. Report dated October 12, 1999, from Dana A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, NRC, Subject: Proposed Plans for Developing Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
6. Report dated February 11, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Importance Measures Derived from Probabilistic Risk Assessments.
7. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
8. M.C. Cheok, G.W. Parry, and R.R. Sherry, "Use of Importance Measures in Risk-Informed Regulatory Applications," *Reliability Engineering and System Safety*, 60, 213-226, 1998.
9. W.E. Vesely, "Reservations on 'ASME Risk-Based Inservice Inspection and Testing: An Outlook to the Future,'" *Risk Analysis*, 18, 423-425, 1998.
10. U.S. Nuclear Regulatory Commission, NUREG-1150, Vols. 1-3, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990
11. M. Modarres and M. Agarwal, "Consideration of Probabilistic Uncertainty in Risk-Based Importance Ranking," Proceedings of the International Topical Meeting on Probabilistic Safety Assessment, PSA '96, *Moving Toward Risk-Based Regulation*, Park City, Utah, September 29-October 3, 1996, 230-236, American Nuclear Society.
12. N.J. Diaz, "When...Large is Small and Small is Large," Remarks at the U.S. Nuclear Regulatory Commission, 2002 Regulatory Information Conference, March 5-7, 2002.

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

May 6, 2002

Dr. George E. Apostolakis, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 2, EXTENDED POWER UPRATE

Dear Dr. Apostolakis:

On February 13, 2002, the staff presented its review of the Arkansas Nuclear One, Unit 2 (ANO-2), extended power uprate (EPU) application to the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Thermal-Hydraulic Phenomena. During the 490th meeting of the ACRS, on March 7, 2002, the staff discussed the EPU with the ACRS Full Committee. By letter dated March 14, 2002, the ACRS forwarded its conclusions and recommendations on the staff's review of the ANO-2 EPU application to Chairman Meserve. In that letter, the ACRS provided the following conclusions and recommendations:

1. The Entergy application for a power level increase from 2815 MWt to 3026 MWt for ANO-2 should be approved.

2. The process used by the staff and the Applicant was comprehensive enough to identify the important issues associated with pressurized water reactor (PWR) power uprates. The process would be greatly improved by the availability of a standard review plan to guide both staff and the Applicant.

3. The process used by the Applicant to perform the Reload Safety Analysis appears to be appropriate. Because this is the first large power uprate for a PWR, the staff should review the Reload Safety Analysis for the transitional core reloads to ensure that the plant will operate in compliance with the regulations.

In addition, the ACRS Chairman attached some comments on the human error probability (HEP) estimates in the staff's Draft Safety Evaluation (SE) (a copy of which was forwarded to the ACRS by a January 18, 2002, memorandum from John A. Zwolinski to John T. Larkins). Specifically, the ACRS Chairman believes that there is no credible HEP model that is sufficiently sensitive to the calculated reductions in available times for operator actions, to be able to yield believable HEP estimates, and that the staff should not accept results that are produced from methodologies that are neither approved by the NRC, nor widely accepted.

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The staff's responses to the ACRS conclusions, recommendations, and comments are enclosed.

Sincerely,

William D. Travers
Executive Director
for Operations

Enclosure: Staff Response to ACRS Comments

cc w/encl: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

THE STAFF'S RESPONSE TO
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
LETTER DATED MARCH 14, 2002
RELATED TO THE EXTENDED POWER UPRATE FOR
ARKANSAS NUCLEAR ONE, UNIT 2 (ANO-2)

1. **The Entergy application for a power level increase from 2815 MWt to 3026 MWt for ANO-2 should be approved.**

The staff has issued the approval for the extended power uprate (EPU).

2. **The process used by the staff and the Applicant was comprehensive enough to identify the important issues associated with pressurized water reactor (PWR) power uprates. The process would be greatly improved by the availability of a standard review plan to guide both staff and the Applicant.**

During its meeting with the Commission on December 5, 2001, the ACRS recommended to the Commission that the staff should develop a standard review plan for power uprates. As a result of that meeting, the Commission directed the staff, by a staff requirements memorandum dated December 20, 2001, to review the ACRS recommendation and inform the Commission of the results of this review. The staff has undertaken a comprehensive evaluation of the need for a standard review plan and has held a public workshop with stakeholders on March 19, 2002, to obtain feedback on this and other issues related to power uprates. The staff agrees with the ACRS conclusion and will develop standard criteria to be used to review licensee requests for power uprates.

3. **The process used by the Applicant to perform the Reload Safety Analysis appears to be appropriate. Because this is the first large power uprate for a PWR, the staff should review the Reload Safety Analysis for the transitional core reloads to ensure that the plant will operate in compliance with the regulations.**

In general, the staff agrees with the recommendation to perform reload safety analyses reviews associated with PWR uprates. The staff also believes that there is value in performing onsite reviews of reloads for power uprates that involve the use of mixed fuel types or new fuel designs, or when the full spectrum of transient and accident evaluations is not submitted with the power uprate amendment application. The staff will continue to evaluate the need to audit and review plant-specific reload evaluations on a case-by-case basis.

The ANO-2 power uprate, however, does not present unique issues. To support this power uprate, the licensee submitted and the staff reviewed the requisite reload safety analyses either as part of the steam generator replacement review or as part of the power uprate review. The ANO-2 power uprate core is not a transitional core because the specific fuel design remains unchanged from the fuel type previously used at ANO-2, and the use of fuel rods containing

ENCLOSURE

erbia as a neutron poison is not unique to ANO-2. The kilowatt (Kw) per foot heat generation rate is increasing to provide the additional power; however, the changes in the Kw per foot are well within in the values used by the industry and would present no unique evaluation concerns.

4. **There is no credible HEP model that is sufficiently sensitive to the calculated reductions in available time to be able to yield believable HEP estimates (i.e., the differences in HEP estimates resulting from minor changes in available time). Thus, the staff conclusion that the HEP values reasonably reflect the reductions in time available for operator action cannot be justified.**

The staff recognizes that the results of the application of any human reliability analysis (HRA) methodology may provide differences in the estimated HEP values for slight differences in available action times. This is an analytical result of the HRA methodology implemented by the licensee. The staff concluded that it was reasonable that slightly higher HEP values would be calculated for the same operator action with slightly less time available for the operator to perform the action. This is a qualitative finding (i.e., reasonableness check) of a quantitative result. The staff also recognizes that these differences in HEP estimates are insignificant in relation to the uncertainties in the HRA methodologies used in estimating them. As a result, the estimated absolute values for HEPs cannot be used as the sole basis for determining the acceptability of a license application. However, the evaluations can provide insights into the relative importance (or change in importance) of selected operator actions, can be used to focus the staff review of the license application on those aspects impacted by the EPU, and can be used to evaluate the overall relative change in risk from an application. A paragraph has been added to the ANO-2 EPU SE to discuss this issue.

5. **The HRA methodologies have not been approved by the NRC and are not widely accepted by the technical community. Thus, the staff should not accept results that are produced from methodologies that are neither approved by the NRC nor widely accepted.**

The staff recognizes that none of the numerous, different HRA methodologies used in probabilistic risk assessments (PRAs) throughout the industry have been formally reviewed and approved by the NRC. However, the HRA methodologies used in the ANO-2 EPU license application are among the methods that comprise the current state-of-the-art. The staff recognizes that the HRA methodologies are evolving and that no particular HRA method may have a full consensus within the technical community. However, the staff does not fully agree with the comment that these methodologies are not widely accepted by the technical community. Their use by a relatively large number of licensee PRA staff and by a number of PRA consultants, who use them to provide a means to estimate HEP values in a relatively coherent way that recognizes the influence of some situational characteristics (e.g., operator available response time), indicates their acceptance by these practitioners as the best available methods. Also, just because this is an area of active research does not invalidate the use of the current state-of-the-art methodologies to produce risk insights. As better and more refined methods evolve, the staff expects that licensees would incorporate them into their PRAs. The staff review of the ANO-2 HRA methodologies, which were used in their risk impact evaluation, was to ensure that the licensee properly used the methodologies that they cited. A paragraph

has been added to the ANO-2 EPU SE to clarify that the cited HRA methodologies have not been formally reviewed and approved by the NRC.

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

March 14, 2002

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

Dear Chairman Meserve:

SUBJECT: CORE POWER UPRATE FOR ARKANSAS NUCLEAR ONE, UNIT 2

During the 490th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 2002, we completed our review of the Entergy Operations, Inc. (Entergy) application for a power uprate of 7.5 percent for Arkansas Nuclear One – Unit 2 (ANO-2), and the related NRC staff's Safety Evaluation Report (SER). Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter on February 13, 2002. During our review, we had discussions with representatives of the Applicant and the NRC staff, and we also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The Entergy application for a power level increase from 2815 MWt to 3026 MWt for ANO-2 should be approved.

2. The process used by the staff and the Applicant was comprehensive enough to identify the important issues associated with pressurized water reactor (PWR) power uprates. The process would be greatly improved by the availability of a standard review plan to guide both staff and the Applicant.

3. The process used by the Applicant to perform the Reload Safety Analysis appears to be appropriate. Because this is the first large power uprate for a PWR, the staff should review the Reload Safety Analysis for the transitional core reloads to ensure that the plant will operate in compliance with the regulations.

Discussion

In 1997, the staff performed a comprehensive review of an application for a PWR power uprate involving the Joseph M. Farley nuclear power plant. The Farley plant Licensee used the guidance in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," to prepare its application. This guidance has not been formally reviewed and approved. ANO-2 is a Combustion Engineering reactor, not a Westinghouse reactor like Farley. We believe, however, that there is enough similarity

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between the ANO-2 plant and the Westinghouse plants to justify the use of WCAP-10263 and the Farley plant SER as templates and guidelines. The Applicant also used General Electric Topical Report NEDC-31897P-A, "Generic Guidelines for BWR Extended Power Uprates," and SECY 97-042, Section 3, "Power Uprate Review Process," to support and substantiate its analyses.

Although we believe that the approach used by Entergy and the staff is sufficiently comprehensive to identify the important PWR power uprate issues, the process would be greatly improved by the availability of a better template such as a standard review plan.

It is difficult to perform a major power uprate in a PWR unless significant modifications are made to the plant. In a PWR, the power is limited by the amount of heat exchange surface. ANO-2 installed larger replacement steam generators that can accommodate the higher thermal power, but, these larger steam generators impose greater accident loads on the containment. The increased energy release during a potential steamline break accident required an increase in the containment building design pressure rating from 54 psig to 59 psig. Instead of modifying the containment building, the Applicant reanalyzed the strength of the containment – considering additional tendons that had not been credited in the original analysis. The containment pressure capability was demonstrated by conducting a pressure test at 68 psig. We conclude that the Applicant's analyses of containment loads and demonstration of the design capability of the containment structure are adequate.

Entergy does not propose to alter the basic thermal-hydraulic design of the reactor core, but will change the neutronic design to provide more core power flattening.

For the uprated power plant, the licensee will use a different code for the analysis of the large-break LOCA. This code has previously been reviewed by the staff. It includes a revised reflood heat transfer coefficient correlation, derived from the FLECHT data, and other code improvements to the Appendix K ECCS evaluation model. The model predicts a peak cladding temperature approximately 150°F less than the previous evaluation model.

Because of the significant changes to the physical plant and to the analytical models used to analyze the plant under accident conditions, the staff should review the transition reload safety analyses for this plant to ensure that the Applicant properly incorporated plant design changes and parameters that describe the characteristics of the transition reload.

The Applicant has scheduled many modifications to the balance of plant to accommodate the increased power output and the additional component duty that will result from an increase in rated power. These involve changes to the Main Unit Turbine/Generator, the Main Unit Condenser, and accessories and associated supporting systems. We did not find significant safety issues associated with the planned modifications.

The uprated power level leads to an increase of reactor head temperature and thereby will increase the susceptibility of the Control Rod Drive Mechanism (CRDM) nozzles to cracking. ANO-2 is a "cold head" plant. There is some bypass flow directed to the reactor head region which lowers the reactor head temperature and reduces susceptibility to cracking of CRDM nozzles. This plant was ranked as an "intermediate plant" using Electric Power Research Institute Materials Reliability Program Reports 44 and 48 and will remain an "intermediate plant." Appropriate management of the issues involved in reactor vessel CRDM weld and nozzle cracking is under active consideration by the staff and the nuclear industry. The resolution of this problem will not be affected by the power uprate.

The ANO-2 reactor vessel has a very large margin to the pressurized thermal shock and upper-shelf energy limits and, thus, the neutron fluence and thermal conditions for the upgraded power level will have little effect.

The ANO-2 application for power uprate was not submitted as a "risk informed" application. However, the Applicant did supply risk information, which the staff examined. The Applicant's evaluation of the increase in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) indicates that these changes can be classified under the guidelines of Regulatory Guide 1.174 as a "small change" for CDF and as a "very small change" for LERF.

Based on our review of the ANO-2 power uprate application and the associated NRC staff's SER, we believe that the requested power level increase for ANO-2 should be approved.

Additional comments by ACRS Member George E. Apostolakis are provided below.

Sincerely,

George E. Apostolakis
Chairman

Additional Comments by ACRS Member George E. Apostolakis

I appreciate the fact that the power uprate requests are not risk informed. Even though estimates of ΔCDF and ΔLERF are provided, the decision of whether to approve the requested uprate is based primarily on conservative "deterministic" calculations.

An important input to the estimation of ΔCDF and ΔLERF is the change in human error probabilities (HEPs). This change is due to shorter available times for operator action that the power uprate generates.

The licensee and the staff did a commendable job in identifying operator actions that could be affected by the power uprate.

I do object, however, to the HEP quantitative estimates that are provided. I do not believe that there are any credible HEP models that are sufficiently sensitive to the calculated reductions in available time to be able to yield believable HEP estimates. For example, Table 8.1 of the SER lists the following human failure event: "Failure to re-energize 2A1/2A2 from ST2 (SBLOCA or SGTR)." The pre-uprate available time was 42 minutes and the estimated HEP was 0.19. The post-uprate available time was estimated to be 39 minutes and the new HEP was 0.29.

I do not believe these results. I do not think that the model that will discriminate between 42 and 39 minutes has been developed yet. The licensee states that these estimates are produced using several EPRI reports. These reports have not been approved by the NRC and are not widely accepted by the technical community. The staff is careful to state (Section 8.1.4) that "... the licensee's human reliability analysis application is consistent with the identified methodologies...." While this may be a true statement, it really does not say anything about the methodologies themselves.

I do not know whether the staff's conclusion that the HEP values reasonably reflect the reductions in times available for operator action is true. I suspect it is, but I do not have a credible model that will convince me that it is true.

I do not think that the staff should accept results that are produced from methodologies that are neither approved by the NRC, nor widely accepted.

References:

1. Memorandum dated December 19, 2000, from Entergy Operations, Inc., to U.S. NRC, Subject: Arkansas Nuclear One-Unit 2 Application for License Amendment to Increase Authorized Power Level.
2. Memorandum dated January 22, 2002, from Amarjit Singh, ACRS, to ACRS Members, transmitting memorandum dated January 18, 2002, from J. A. Zwolinski, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, transmitting Arkansas Nuclear One, Unit No. 2 - Draft Safety Evaluation for Extended Power Uprate (Predecisional).
3. Letter dated March 1, 2002, from Sherri R. Cotton, Entergy Operations, Inc., to Nuclear Regulatory Commission, Subject: ANO Unit 2, Follow-up Questions Resulting from the ACRS Subcommittee's Review of ANO-2's Proposed Power Uprate, dated March 1, 2002.
4. Memorandum dated February 7, 2002, from Paul Boehnert, ACRS, to ACRS Members, Subject: ACRS Review of ANO Unit 2 Core Power Uprate Request - Additional Background Material.

5. Letter dated February 7, 2002, from Glenn R. Ashley, Entergy Operations, Inc., to USNRC, Subject: ANO Unit 2, Response to Request for Additional Information on Vessel Head Penetration Nozzles Regarding the ANO-2 Power Uprate License Application.
6. Letter dated February 7, 2002, from Glenn R. Ashley, Entergy Operations, Inc., to USNRC, Subject: ANO Unit 2, Comments Regarding the Draft NRC Safety Evaluation for the Proposed ANO-2 Power Uprate.
7. Memorandums from Entergy Operations, Inc., Response to Requests for Additional Information Regarding the ANO-2 Power Uprate License Application, dated December 20 (contains proprietary material), November 16 (contains proprietary material), November 16, November 9, October 31(contains proprietary material), October 30, October 17, October 1, and September 14, 2001.
8. Memorandum dated January 31, 2002, from Entergy Operations, Inc., to Nuclear Regulatory Commission, Subject: Arkansas Nuclear One Unit 2 Response to Follow-up Request for Additional Information Concerning SGTR and MHA Dose Assessment Calculations Supporting ANO-2 Power Uprate.
9. Entergy Operations, Inc., Memorandums, Response to Requests for Additional Information Regarding the ANO-2 Power Uprate License Application, dated May 30, June 20, June 26, June 26, June 28, July 3 (contains proprietary material), July 24, July 24, August 7, August 13, August 21, August 23 (contains proprietary material), August 30, 2001.
10. Letter dated September 29, 2000, from Thomas W. Alexion, Office of Nuclear Reactor Regulation, NRC, to Craig G. Anderson, Entergy Operations, Inc., Subject: ANO Unit 2, Issuance of Amendment Re: Technical Specification Changes and Unreviewed Safety Question Resolution Related to Applicable Limits and Setpoints for Steam Generator Replacement.
11. Letter dated November 13, 2000, from T. Alexion, Office of Nuclear Reactor Regulation, NRC, to Craig G. Anderson, Entergy Operations, Inc., Subject: ANO Unit 2 Issuance of Amendment Re: Technical Specification Changes and Unreviewed Safety Question Resolution Related to Containment Building Design Pressure Increase to 59 PSIG.
12. GE Nuclear Energy Licensing Topical Report, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," NEDC-31897P-A, dated May 1992.
13. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

Janet/Patty Disk:Travel.Frm
9/9/94

ACRS SPECIAL TRAVEL ENDORSEMENT FORM

THIS FORM IS TO BE USED TO REQUEST ACRS ENDORSEMENT OF SPECIAL TRAVEL REQUESTS BY MEMBERS WHEN NRC SUPPORT FOR PARTIAL OR FULL REIMBURSEMENT OF EXPENSES AND/OR TIME IS DESIRED. THIS PROCEDURE IN NO WAY LIMITS THE FREEDOM OF A MEMBER TO PARTICIPATE IN A MEETING AS AN INDIVIDUAL AT PERSONAL EXPENSE. PLEASE SUBMIT THIS FORM TO THE PLANNING AND PROCEDURES SUBCOMMITTEE AT LEAST 60 DAYS PRIOR TO THE MEETING. IF POSSIBLE. SUPPLEMENTAL INFORMATION MAY BE ADDED AS DETAILS DEVELOP.

Member Name: Stephen L. Rosen Date Submitted: 7/8/02

Dates of Planned Trip: August 28-30, 2002 to

Destination: Hotel Monaco, seattle, Washington

Meeting or Facility to be Visited: NEI Fire Protection Information Forum

Purpose/Relevance to ACRS Business: Mr. Rosen is the Chairman of the ACRS' Subcommittee on Fire Protection.

Participation (Invited Speaker, paper presented, etc.): N/A

Justification (Foreign Travel Only):

NRC SUPPORT REQUESTED

Air Fare: Yes x No

Per Diem: Yes x No 3 Days 3

Registration: $ x

Compensation: Yes x No 3 Days 3

Item 10
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NEI Fire Protection Information Forum

August 28-30, 2002❖Hotel Monaco❖Seattle, WA

HOTEL INFORMATION

Hotel Monaco
1101 Fourth Avenue
Seattle, WA 98101
206.621.1770
800.945.2240 RESERVATIONS
206.621.7779 FAX

City Information: www.seattle.com
Hotel information: www.monaco-seattle.com

Make your hotel reservation directly with Hotel Monaco and identify yourself as an attendee of the NEI Fire Protection Forum to secure the room rate of $150 single/double. To guarantee a room and this rate the reservation must be made by **August 6**; after this date the price and room availability are up to the discretion of the hotel.
Check in is 3:00 p.m. / Check out is Noon.

CONFERENCE INFORMATION

- Registration will begin each day starting at 7:00 a.m. Continental breakfast will be provided.
- Conference sessions will be held at the following times:
  Wednesday 8:00 a.m. – 5:00 p.m.
  Thursday 8:00 a.m. – 5:00 p.m.
  Friday 8:00 a.m. – 1:00 p.m.
- NEI will host a Welcoming Reception Wednesday, August 28 5:30 p.m. – 6:30 p.m.
- Lunch will be provided Wednesday and Thursday.

TRANSPORTATION

Airline Reservations
NEI's Travel Department is available to assist you with your travel needs. For reservations and fare information, contact:
NEI Travel Office 202.739.8100
Monday - Friday - 8:30 a.m. - 5:30 p.m.

SEA-TAC Airport is located 20 miles from the hotel – about a 30-minute ride. Taxi's cost approximately $25-35. For shuttle information go to http://www.graylineofseattle.com/airport.cfm. NOTE: the shuttle does not go directly to the Monaco, but does stop at a hotel 1 block from the Monaco.

Car Rental Discount

Hertz Rent-A-Car is offering NEI a meeting discount. To make your reservation, call Hertz at 800.654.3131 and refer to discount number **189851**.

BUSINESS-CASUAL ATTIRE IS APPROPRIATE FOR THE FORUM.

I am attaching the registration forms and hotel/transportation information for the Fire Protection Information Forum August 28-30. Hotel reservations must be made by August 6 to get the NEI rate; please note that a government rate is available. When making reservations, please recall that the the EPRI Fire Modeling Workshop will be held in the same hotel August 26-27. Registration for that workshop is separate. When sending in the registration form, government employees should use the NEI Member rate of $375. I will provide a draft program in about a week. Currently the sessions will include: Fire protection benchmarking project results (this will be most of the first day) Plant fire protection operations and experience (2 sessions) Current regulatory issues (2 sessions) Circuit analysis and related issues Fire protection rulemaking Fire protection inspections and self-assessments Fire protection program change guidance Please let me know if you have difficulty with getting the NEI or government rate at the hotel on or before August 6, or have other questions. Fred Emerson 202-739-8086

NEI Fire Protection Information Forum

August 28-30, 2002 * Hotel Monaco * Seattle, Wa

Attendee Registration
Includes all sessions, session materials and catered events. Member rates apply to employees of member companies and all government agencies.

NEI Member $375 Non-Member $750

8/28/02 **Wednesday Lunch**	Will Attend	Will Not Attend
8/29/02 **Thursday Lunch**	Will Attend	Will Not Attend
8/28/02 **Reception**	Will Attend	Will Not Attend

All payments must accompany registration form. Conference fees will not be refunded after August 16, however; substitutions are accepted. In order to appear on the participants list, registration must be received by August 16. Return registration form to:

Nuclear Energy Institute
Department 9013
Attention: Alexandra Iwuchukwu
Washington, DC 20061-9013

Via FedEx
1776 I St., NW #400
Washington, DC 20006

Phone: 202.739.8039
Fax: 202.833.2282
Email: ani@nei.org

Please indicate any special needs:

Registrant Information:

Mr. Ms.

Name

Badge Name (informal)

Title

Organization

Mailing Address

City/State/Zip Code/Country

Phone Number Fax

E-mail Address

FedEx Address (do not use a P.O. Box)

Registration fee is paid by enclosed check
Charge my credit card for the registration fee
Wire Transfer: Please call 202.739.8161

Credit Card Information:

Visa MasterCard American Express

Card Number

Expiration Date

Required: Credit Card Billing Address (if different from above)

Name (as it appears on card)

Cardholder's Signature

NEI Fire Protection Information Forı

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