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

March 3, 2000

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

SUBJECT: SUMMARY REPORT - FOUR HUNDRED SIXTY-NINTH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, ON FEBRUARY 3-6, 2000, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

Dear Chairman Meserve:

During its 469th meeting, February 3-5, 2000, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports and letter:

REPORTS

- SECY-00-0011, "Evaluation of the Requirement for Licensees to Update Their Inservice Inspection and Inservice Testing Programs Every 120 Months" (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated February 8, 2000)

- Importance Measures Derived from Probabilistic Risk Assessments (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated February 11, 2000)

- Impediments to the Increased Use of Risk-Informed Regulation (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated February 14, 2000)

LETTER

- Revision of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, ACRS, dated February 11, 2000)

HIGHLIGHTS OF KEY ISSUES CONSIDERED BY THE COMMITTEE

1. Technical Aspects Associated With the Revised Reactor Oversight Process and Related Matters

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning technical aspects of the revised reactor oversight process, including the technical adequacy of current and proposed performance indicators, significance of the determination process, and initial implementation issues. The Committee and the staff extensively discussed the objectives of the performance indicators in identifying adverse changes in performance; technical bases, sensitivity, and adequacy of thresholds; use of a 95th percentile criterion; sufficiency of generic values and design- and plant-specific considerations affecting the application of thresholds; collective risk from approaching thresholds in multiple areas rather than crossing a single threshold; and research that might be needed for Phase 3 decisionmaking to compensate for inadequacies in individual plant examinations and probabilistic risk assessments (PRAs). The Committee also discussed the staff's plans for initial program implementation in April 2000 and plans to reassess the program for full implementation after about a year. The Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Plant Operations met on January 20, 2000, to discuss these issues.

Conclusion

The Committee decided to continue its review during the ACRS meeting on March 2-4, 2000, when the proposed Commission paper would be available for review.

2. Proposed Final Amendment to 10 CFR 50.72 and 50.73

The Committee heard presentations by and held discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) concerning the proposed final amendment to 10 CFR 50.72 and 50.73 regarding reporting requirements. The Committee members, the staff, and NEI discussed the added requirement for reporting degraded components and the relationship between the added list of reportable system actuations and the engineered safety feature systems identified in the licensees' final safety analysis reports. The staff committed to hold a workshop concerning the added requirement for reporting degraded components.

Conclusion

The Committee decided to review this issue after the staff holds a public workshop.

3. Proposed Regulatory Guide and Associated Document NEI 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations"

The Committee heard presentations by and held discussions with representatives of the NRC and NEI concerning NEI 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluation," and the proposed regulatory guide.

NRC regulation 10 CFR 50.59 defines the conditions under which reactor licensees may make changes to their facilities or procedures or may conduct tests or experiments without prior NRC approval. Generally, these changes, tests, or experiments may be carried out unless they involve a change to the technical specifications, or an unreviewed safety question. In 1999, the NRC revised 10 CFR 50.59 to provide for more flexibility, primarily by allowing changes that have a minimal safety impact to be made without prior NRC approval. The final rule was approved on June 22, 1999, and was published on October 4, 1999.

NEI prepared NEI 96-07 (final draft dated January 18, 2000), which provides guidance for implementing the revised rule, and requested NRC endorsement in the regulatory guide. The rule revisions will become effective 90 days after approval of the guidance.

Currently, the staff is still reviewing NEI 96-07, and several open issues that need to be resolved. These open issues include fire protection plan changes; methods and guidance on plant-specific approvals; design basis limits for fission product barriers; and screening of design function, numerical values, and their relationship to maintenance assessments.

Conclusion

The Committee believes that the staff has revised "Guidelines for 10 CFR 50.59 Safety Evaluations" to the point that further review of this material by ACRS would not add value.

4. Proposed Revision of the Commission's Safety Goal Policy Statement for Reactors

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning the proposed revision of the Commission's Safety Goal Policy Statement (SGPS) for nuclear power plants. This revision was driven by both Commission direction and recommendations from the ACRS. Changes to the SGPS are recommended in the following areas:

- Reflection of Current Agency Policy
- Treatment of Core Damage Frequency as a Fundamental Goal
- Treatment of Uncertainty
- Defense in Depth
- Safety Goal Structure and Adequate Protection
- Frequency of a Large Release of Radioactive Material
- Land Contamination and Overall Societal Impact
- Temporary Changes in Risk

The SGPS is still being reviewed by internal management, and the staff is scheduled to submit its recommendations to the Commission by March 30, 2000.

Conclusion

The Committee will continue its discussion of this issue during the March 2000 ACRS meeting.

5. Impediments to the Increased Use of Risk-Informed Regulation and the Use of Importance Measures in the Risk-Informing of 10 CFR Part 50

The Committee heard presentations by and held discussions with representatives of the NRC staff, NEI, and the industry-invited experts regarding the impediments to the increased use of risk-informed regulation and the use of importance measures in the risk informing of 10 CFR Part 50.

The staff stated that there are six key elements of the risk-informed regulation: (1) policy, (2) strategy , (3) knowledgeable staff, (4) decisionmaking, (5) tools, and (6) communication. The staff from NRR is working on the reactor safety policy, and the Commission has directed the staff in NMSS to develop safety goals and an approach for risk-informing in the NMSS activities. There will certainly be key challenges in the development of safety goals for the non-reactor activities because they have a number of different areas that are afflicted and have different levels of risk. In the area of strategy, the staff has a different

PRA implementation plan because the staff was criticized by the General Accounting Office for not having a real strategy for risk-informing of agency activities. The staff is developing such a strategy to cover the risk-informing PRA implementation plan. The third item is staffing, which also includes licensee staff. NRC has training programs in place and continue to review the training needs and the staffing level in this area. The fourth area is decisionmaking, which basically provides guidance documents, both for the industry and the staff to utilize. In the area of communication, the pilot programs provide an effective way of communicating with the industry and the stakeholders, and the NRC staff will also hold internal panel sessions to educate the staff.

The representatives of industry-invited experts made the following significant points regarding the impediments to the importance measures for risk-informed regulation:

- Difficulty in quantifying costs and benefits.
- Variations in PRA quality and scope.
- Duration of the regulatory review process.
- Lack of PRA standards to establish quality.
- No differentiation between design basis events and events that are likely to occur during plant life.
- No mechanism available or path by which to change safety-related component classification based on risk information.
- Lack of clarity or criteria in the degree to which risk-informed applications could be approved using only qualitative approaches versus those applications that use quantitative methods or both.
- Lack of training that demonstrates the complementary effect of blending deterministic and probabilistic approaches in decisionmaking.
- Misconception that PRA is too expensive relative to its benefits.
- Risk-ranking methods and uncertainty analyses that need further development.

- Importance measures that can identify what is important but do not necessarily identify what is not important.

- Performance of sensitivity studies regardless of the methods used, following classification of components into risk-significance categories to confirm the classification.

Conclusion

The Committee issued reports to Chairman Meserve on importance measures and impediments dated February 11 and 15, 2000, respectively.

6. Proposed Final Revision of Appendix K to 10 CFR Part 50

The Committee heard presentations by and held discussions with representatives of the NRC staff and the nuclear industry concerning the proposed final revision of Appendix K to 10 CFR Part 50. This change would relax the requirement that a licensee assume 1.02 times licensed power for the Appendix K emergency core cooling system analysis and allow licensees to seek credit in safety analyses for installation of highly accurate flow measurement systems. The ACRS had reviewed the proposed version of this rule revision during the July 1999 meeting. The staff issued the proposed version of this rule revision for public comment on October 1, 1999; the public comment period ended on December 15, 1999. Six respondents provided comments. All responses were positive, and the staff will make some minor clarifications in the Federal Register notice in response to these comments. The language of the proposed rule itself has not been modified.

During the discussion, members of the Committee indicated that the staff needs to provide guidance to licensees regarding appropriate accounting for measurement uncertainty in their safety analyses associated with the use of highly accurate flow instrumentation.

Conclusion

The Committee issued a report to the EDO, dated February 11, 2000, on this issue.

7. NRC Safety Research Program Report to the Commission

The Committee discussed its final draft of the ACRS Year 2000 report to the Commission on the NRC Safety Research Program.

Conclusion

The Committee finalized its final draft report on this issue and sent an advance copy to the Commission on February 7, 2000.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee discussed the response from the NRC Executive Director for Operations (EDO) dated January 13, 2000, to the ACRS comments and recommendations included in the ACRS report dated December 8, 1999, concerning a draft Commission paper regarding elimination of the 120-month update requirement from 10 CFR 50.55a, "Codes and standards."

  The Committee was not satisfied with the EDO response. The Committee decided to prepare a reply to the EDO that would restate the Committee's recommendation that the 120-month update be retained.

- The Committee discussed the response from the NRC Executive Director for Operations (EDO) dated January 24, 2000, to the ACRS comments and recommendations included in the ACRS report dated December 10, 1999, concerning the safety aspects of the license renewal application for Calvert Cliffs Nuclear Power Plant.

  The Committee was satisfied with the EDO response.

- The Committee discussed the response from the NRC Executive Director for Operations dated January 20, 2000, to comments and recommendations of the joint ACRS/ACNW report dated November 17, 1999, concerning implementing a framework for risk-informed regulation in the Office of Nuclear Material Safety and Safeguards.

  The Committee referred this item to the ACRS/ACNW Joint Subcommittee for evaluation. The Committee plans to review the ACRS/ACNW Joint Subcommittee's recommendations during a future meeting.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from December 2, 1999, through February 2, 2000, the following

Subcommittee meetings were held:

- Reliability and Probabilistic Risk Assessment - December 15-16, 1999

  The Subcommittee discussed the staff's programs for risk-based analysis of reactor operating experience, including special studies for common-cause failure analyses, system and component analyses, accident sequence precursor analyses. In addition, the staff's efforts in the area of risk-informed technical specifications and associated industry initiatives proposed by the Risk-Informed Technical Specification Task Force was discussed.

- Joint Subcommittee - January 13-14, 2000

  The Advisory Committee on Reactor Safeguards and Advisory Committee on Nuclear Waste Joint Subcommittee discussed the defense-in-depth philosophy in the regulatory process, including its role in the licensing of a high-level waste repository, its role in revising the regulatory structure for nuclear reactors, and how the two applications should be related to each other. The discussion also included the role of defense in depth in the regulation of nuclear materials applications.

- Plant Operations - January 20, 2000

  The Subcommittee discussed selected technical components of the revised reactor oversight process, including the updated significant determination process and plant performance indicators.

- Planning and Procedures - January 27-29, 2000

  The Subcommittee discussed issues related to PRA quality, including development of industrial standards; use of importance measures in risk-informing 10 CFR Part 50; impediments to the increased use of risk-informed regulation; technical aspects of the revised reactor oversight process, including technical adequacy of the current and proposed performance indicators; and safety culture. In addition, the Subcommittee will discuss best estimate computer codes, technical quality of codes, and how they are used at the NRC. It will also discuss industry views of ACRS activities, self-assessment of ACRS performance in CY 1999, potential operational areas for improved effectiveness, positions on PRA issues, technical adequacy of the current and proposed performance indicators for the revised reactor oversight process.

- Planning and Procedures - February 2, 2000

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

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

- The Committee decided to review the proposed final amendment to 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors." and 50.73, "licensee event reporting system," after the staff meets with the Nuclear Energy Institute to discuss the added reporting requirement concerning degraded components.

PROPOSED SCHEDULE FOR THE 470TH ACRS MEETING

The Committee agreed to consider the following during the 470th ACRS Meeting, March 1-4, 2000:

- Development of Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"

  Briefing by and discussions with representatives of the NRC staff regarding the status of developing risk-informed revisions to 10 CFR Part 50.

- Discussion of Topics for Meeting with the NRC Commissioners

  Impediments to the increased use of risk-informed regulation; use of importance measures in regulatory applications, impact of the scope and quality of the PRA on importance measures, and threshold values for importance measures; technical adequacy of performance indicators.

- Technical Components Associated with Revised Reactor Oversight Process

  Briefing by and discussions with representatives of the NRC staff regarding the technical components associated with the revised reactor oversight process, including the updated significant determination process, technical adequacy of the current and proposed plant performance indicators.

- Oconee Nuclear Power Plant License Renewal Application

  Briefing by and discussions with representatives of the NRC staff and Duke Energy Corporation regarding the license renewal application for the Oconee

Nuclear Power Station and the associated NRC staff's Safety Evaluation Report.

- Proposed Final Amendment to 10 CFR 50.72 and 50.73

  Discussions with representatives of the NRC staff regarding issues raised by the ACRS members during the February ACRS meeting, including the intent of the 10 CFR 50.73 requirement for reporting degraded components.

- Proposed Final Revision 3 to Regulatory Guide 1.160, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants"

  Discussions with representatives of the NRC staff, as needed, regarding the proposed final revision 3 to Regulatory Guide 1.160.

- Phenomena Identification and Ranking Table (PIRT) for High Burnup Fuel

  Briefing by and discussions with representatives of the NRC staff regarding the use of PIRT process for high burnup fuel.

- Proposed Resolution of Generic Safety Issue B-17, "Criteria for Safety Related Operator Actions"

  Briefing by and discussions with representatives of the NRC staff regarding the proposed resolution of Generic Safety Issue B-17.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

CERTIFIED

Date Issued 3/14/2000
Date Signed 3/21/2000

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APPENDICES

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MINUTES OF THE 469TH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS FEBRUARY 3-5, 2000 ROCKVILLE, MARYLAND

The 469th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on February 3-5, 2000. The purpose of this meeting was to discuss and take appropriate action on the items listed in the following meeting minutes. The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

A transcript of selected portions of the meeting was kept and is available in the NRC Public Document Room at the Gelman Building, 2120 L Street, N.W., Washington, D.C. [Copies of the transcript are available for purchase from Ann Riley & Associates, Ltd., 1025 Connecticut Avenue, N.W., Suite 1014, Washington, D.C. 20036, and on the ACRS/ACNW Web page at (www.NRC.gov/ACRS/ACNW).]

ATTENDEES

ACRS Members: Dr. Dana A. Powers (Chairman), Dr. George Apostolakis (Vice-Chairman), Mr. John Barton, Dr. Mario V. Bonaca, Dr. Thomas S. Kress, Dr. William J. Shack, Dr. Robert L. Seale, Mr. John D. Sieber, Dr. Robert E. Uhrig, and Dr. Graham B. Wallis. [For a list of other attendees, see Appendix III.]

I. Chairman's Report (Open)

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

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

II. Technical Aspects Associated with the Revised Reactor Oversight Process and Related Matters (Open)

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

Mr. John Barton, Chairman of the ACRS Subcommittee on Plant Operations, introduced the topic to the Committee. He stated that the Subcommittee had met on January 20, 2000, to discuss the technical components of the reactor oversight process, including the Performance Indicators (PIs) and Significance Determination Process (SDP). During that meeting, the staff informed the Subcommittee that the proposed Commission paper associated with the reactor oversight process will not be available for ACRS review until mid-February 2000. The Subcommittee identified a number of issues for the staff to address during the ACRS meeting ON February 3-5, 2000. These issues and questions were provided to the NRC staff in a memorandum dated January 27, 2000.

Mr. Barton also noted that the Director of the Office of Nuclear Reactor Regulation (NRR) requested the Committee to review selected technical components of the reactor oversight process. In a staff requirements memorandum (SRM) dated December 17, 1999, the Commission requested the ACRS to review the technical adequacy of the PIs (current and proposed) for the new reactor oversight process, which includes an assessment of the extent to which the PIs, collectively, provide meaningful insights into those areas of plant operations that are most important to safety. Mr. Barton noted that the ACRS response to the Commission is due March 15, 2000.

NRC Staff Presentation

Mr. Michael Johnson, NRR, led the discussion for the NRC staff. Messrs. Frank Gillespie, William Dean, Alan Madison, and Gareth Parry, NRR, provided supporting discussion. Significant points raised during the staff presentation include the following:

- The PIs and baseline inspection program provide a sound technical framework to ensure that reactor safety is maintained. The process is more objective and focuses on risk-significant issues. The revised reactor oversight process is ready for initial implementation at all plants.

- The current reactor oversight process relies heavily on inspection in which PIs have a minor role, and assessments are completed every 18-24 months. The revised reactor oversight process continuously integrates PIs with inspection and assessment.

- PI thresholds are used to identify performance levels above which increased NRC attention is warranted. PI results are not ranked or trended.

- The SDP evaluates risk on a plant-specific basis, using the individual plant examination (IPE) and/or probabilistic risk assessment (PRA).

- The revised reactor oversight process continues to be a work in progress. The staff expects to complete a containment SDP, screening tools for shutdown operations and external events in April 2000. The staff plans to continue to evaluate and modify the program, as appropriate.

Committee members raised the following significant points:

- Dr. Apostolakis questioned the objectives of the PIs in identifying adverse changes in performance. In particular, he questioned whether the objective was to ensure safety or to verify the plant's operation as licensed. Dr. Apostolakis stated that PIs should be plant-specific. The staff stated that the objectives are to verify that licensee performance is below certain thresholds. The staff stated that licensee performance relative to these thresholds would be used to determine inspection allocation relative to the baseline inspection program that all plants receive.

- Dr. Bonaca and Mr. Barton questioned the technical bases, sensitivity, and adequacy of thresholds. The staff stated that the technical bases were demonstrated in the feasibility study conducted for plants as noted in SECY-99-007A. The staff stated that the PIs use thresholds for regulatory action below which licensees have flexibility in managing activities using the corrective action program. The staff stated that the PIs serve as triggers for a diagnostic mode for further evaluation. Dr. Bonaca expressed the view that the PIs are not sensitive to change and will not provide early warning of declining performance.

- Drs. Apostolakis and Kress questioned the use of a 95th percentile criterion. Dr. Apostolakis stated that this criterion allows a plant to increase risk and

still maintain "green" PI status. Dr. Kress noted that the value is arbitrary and suggested that it could have been 25 or 50 percent. The staff stated that the intent is to identify plants that are extreme outliers in performance relative to the overall population of plants.

- Dr. Apostolakis questioned the sufficiency of using generic PIs and noted that design- and plant-specific considerations affect the application of thresholds. He reiterated his concern regarding collective risk from approaching thresholds in multiple areas rather than crossing a single threshold. He noted that a plant's performance could degrade and not be detected by the NRC PIs. The staff stated that most licensees use lower thresholds to manage their activities in order to maintain sufficient margin from NRC thresholds. The staff stated that it is likely that the NRC would consider increased inspection for plants that approach thresholds. The staff also stated that inspection is an integral element in addition to PIs and would weigh heavily in the final assessment (i.e., color coding).

- Drs. Powers and Apostolakis questioned what research might be needed for Phase 3 decisionmaking to compensate for inadequacies in IPEs and PRAs. The staff stated that the oversight process was sufficient to support decisionmaking and that no immediate research was needed before initial implementation. The staff reiterated that the oversight was sufficient to identify adverse or declining licensee performance and areas needing additional inspection.

At the conclusion of the meeting, the staff reemphasized that the revised reactor oversight program is a work in progress and that additional changes would likely be made as more experience is gained. The staff stated that it would be requesting Commission approval for initial implementation, with a possible reexamination in about a year.

Conclusion

The Committee decided to continue its review during the ACRS meeting of March 2-4, 2000, when the proposed Commission paper would be available for review.

III. Proposed Final Amendment to 10 CFR 50.72 and 50.73 (Open)

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

Dr. Mario V. Bonaca, Chairman of the Plant License Renewal Subcommittee, reviewed the objectives of the proposed final amendment and noted the number of public meetings and the workshop that the staff had conducted. He stated that the Committee reviewed an earlier draft of the proposed amendment and commented on it in a letter issued on March 23, 1999. Dr. Bonaca requested that the staff address the significance of the added reporting requirement concerning degraded components and its relationship to 10 CFR Part 21 reporting requirements.

Mr. Dennis Allison, NRR, presented the objectives of and the principle changes made by the amendment. He explained that the requirement to report conditions outside the design basis of a plant would be deleted. However, a new requirement would be added for reporting degraded components that could be precursors to common-cause failures. Mr. Scott Newberry, NRR, explained that the staff planned to explain the new requirement to stakeholders during a public workshop. Other changes presented by Mr. Allison included the following:

- inclusion of a list for reportable system actuations
- eliminating report requirements for invalid actuations
- changes to required reporting times
- placing a time limit on reporting historical problems
- eliminating the requirement for reporting a late surveillance test

The Committee members and the staff discussed the relationship between the new requirement for reporting degraded components and the 10 CFR Part 21 reporting requirements. They also discussed the list of system actuations and its relationship to engineered safety feature systems contained in final safety analysis reports.

Mr. James Davis, Nuclear Energy Institute (NEI), noted that the staff had used a well-developed process to discuss the proposed rule changes with the industry. The process focused on the operability of the safety functions of systems and components. He objected to the new reporting requirement for degraded components. With the exception of the new reporting requirement, Mr. Davis stressed that the industry supported the proposed final amendment.

The Committee members, Mr. Davis, and the staff discussed the process used to develop the proposed amendment and when the staff would reach a final position concerning the wording of the requirement to report degraded components.

Conclusion

The Committee decided to review this matter after the staff holds a public workshop.

IV. Proposed Regulatory Guide and Associated NEI Document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations" (Open)

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

Mr. J. D. Sieber, ACRS Member, stated that the Committee would hear presentations by and hold discussions with representatives of the NRC and NEI concerning NEI document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations," and the proposed regulatory guide.

Mr. Sieber noted that the NRC regulation 10 CFR 50.59 defines the conditions under which reactor licensees may make changes to their facilities or procedures, or may conduct tests or experiments without prior NRC approval. Generally, these changes, tests, or experiments may be carried out unless they would involve a change to the technical specifications, or an unreviewed safety question. In 1999, the NRC revised 10 CFR 50.59 to provide for more flexibility, primarily by allowing changes that have minimal safety impact to be made without prior NRC approval. The final rule was approved on June 22, 1999, and was published on October 4, 1999.

Ms. Eileen McKenna, NRR, and Mr. R. Bell, NEI, stated that NEI prepared the NEI 96-07 document (final draft, dated January 18, 2000) that provides guidance for implementing the revised rule and is requesting NRC endorsement in a regulatory guide. The rule revisions become effective 90 days after approval of the guidance.

Currently, the staff is still in the process of reviewing the NEI 96-07 document, and several open issues need to be resolved. The open issues include the following:

- Fire protection plan changes - Generic Letter 86-10 regarding a license condition uses 10 CFR 50.59. The proposal is to use a license condition on its own without 10 CFR 50.59.

- Methods - Clarifications are still needed on "essentially the same," and guidance is needed on plant-specific approvals.

- Design basis limits for fission product barriers - The "subordinate" limits concept is not accepted by the staff.

- Screening on design function needs to be defined.

- Numerical values - Additional clarification is still needed.

- Relationship to maintenance assessments - NEI is proposing that "changes associated with maintenance" be covered by the maintenance rule (a)(4) assessments and not by 10 CFR 50.59. Details of the proposal are still being reviewed by the staff.

Conclusion

The Committee believes that the staff has revised "Guidelines for 10 CFR 50.59 Safety Evaluations" to the point that further review of this material by ACRS would not add value.

V. Proposed Revision of the Commission's Safety Goal Statement for Reactors

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

Dr. T. Kress, Chairman, Severe Accident Management Subcommittee, introduced this topic to the Committee. He noted that the Committee has met on this matter previously and that the staff had identified a set of nine issues that were to be evaluated with regard to revising the Safety Goal Policy Statement (SGPS). Further, the Committee had agreed that these issues appeared to be an appropriate set for this purpose. Dr. Kress said that the Committee will attempt to draft a letter on this matter during the meeting in March 2000.

Mr. J. Murphy, Office of Nuclear Regulatory Research (RES), discussed the proposed revisions to the SGPS. These revisions were categorized into eight topic areas. These topic areas and a summary of the proposed changes are as follows:

- Reflection of Current Agency Policy

  (1) Add the following "Five Principles" to clarify and reflect current practice:

  — Plants are expected to meet current regulations and any applicable exemptions
  — Maintain defense-in-depth philosophy
  — Maintain sufficient safety margins
  — Increases in risk of core damage frequency should be small relative to the safety goals
  — Plant performance should be monitored using performance measurement strategies

  (2) Incorporate positions taken in on a SRM dated June 15, 1999, that safety goals establish a level of safety considered safe enough and they represent a risk level to strive for by utilizing the provisions of the backfit rule.

- Treatment of Core Damage Frequency as a Fundamental Goal

  (1) Elevate the qualitative statement in Section 2 of the policy statement that the Commission has as its objective that a severe core-damage accident will not occur at a U.S. nuclear power plant to the status of a qualitative goal

  (2) Retain a quantitative value of 10^{-4} per reactor-year as a useful subsidiary performance objective.

- Treatment of Uncertainty: Reference in the policy statement (Section IV) the appropriate section of Regulatory Guide 1.174 (Section 2.2.5) and incorporate the more general portions, as appropriate.

- Defense in Depth: Incorporate the relative portion of the Commission's White Paper dealing with the role of defense in depth in a risk-informed regulatory framework. Take note that "the ACRS and ACNW are developing

additional recommendations to the Commission in the area of defense in depth."

- Safety Goal Structure and Adequate Protection: The staff recommends that no change be made to the SGPS relative to the Committee's recommendations in this regard. (The argument made is that a structure similar to that recommended by the Committee already exists in the regulations and other implementing documents.) Consistent with Commission guidance in the SECY-99-191 SRM, the staff will consider revising the "reasonable assurance of adequate protection" guidance, as well as regulatory and backfit analyses after experience is gained with the use of risk information in regulatory practices.

- Frequency of Large Release of Radioactive Material: Delete reference to the "General Performance Guideline" (a housekeeping chore because this effort was terminated by SECY-93-138) and incorporate a large early release frequency (LERF) subsidiary goal of 10^{-5} per reactor-year (consistent with the ACRS recommendation in its report of May 11, 1998).

- Societal Risk: Two issues were considered:

  (1) Should the policy statement or the Regulatory Analysis Guidelines be modified?

  (2) Should these two documents be made consistent relative to treatment of societal dose?

  The staff argued that both documents serve different purposes and the use of a 10-mile zone in the SGPS and a 50-mile zone in the Regulatory Analysis Guidelines should not be changed.

- Land Contamination and Overall Societal Impact: The staff recommends that a qualitative goal be added for protection of the environment. The current tools (PRAs and IPEs) are not sufficiently robust for determination of the extent of land contamination and the resulting societal impact. Development of the needed tools will be considered pursuant to the agency's normal planning, budgeting, and performance management process.

- Temporary Changes in Risk: Temporary changes in risk are already covered in principle. To make it clear, however, the staff suggests the SGPS be clarified as it applies to temporary changes, as well as to average annual risk.

The SGPS is still undergoing internal management review, and the staff is scheduled to provide its recommendations to the Commission by March 30, 2000.

During the discussion, the following key points were noted:

- In response to questions from Drs. Kress and Wallis, Dr. Murphy said that the safety goal has influenced the development of regulations by way of the Regulatory Analysis Guidelines. Mr. King, RES, said that on the basis of evaluation of the IPEs, most plants meet the safety goals.

- The term "adequate protection" was extensively discussed, as referenced in the SGPS, as well as development of a more quantitative definition. Drs. Kress and Apostolakis proposed exploring the use of core damage frequency (CDF) and LERF as surrogates to the high-level goals in the policy statement to achieve this end. Dr. Murphy indicated that the staff will consider this matter in the context of possible future revisions to the SGPS for the issue of adequate protection as experience is gained with use of risk-informed regulation.

- Dr. Kress recommended revisions to the "Five Generalized Principles" to properly reflect their status as high-level goals. Dr. Murphy indicated his agreement with these suggested revisions.

- Dr. Powers suggested that NRC explore setting some cap on the amount of acceptable dose to the public with regard to limiting the dose dispersion calculations relative to land contamination considerations.

Conclusion

The Committee will continue its discussion of this matter during the ACRS meeting in March 2000.

VI. Impediments to the Increased Use of Risk-Informed Regulation and Use of Importance Measures in Risk-Informing 10 CFR Part 50

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

Dr. George Apostolakis, Chairman of the Subcommittee on Reliability and Probabilistic Risk Assessment, introduced the topic to the Committee. He stated that the purpose of this session was to discuss with representatives of the NRC staff, NEI, and the industry invited experts the impediments to the increased use of risk-informed regulation and the use of importance measures in the risk informing of 10 CFR Part 50.

NRC Staff Presentation

Mr. Thomas L. King led the discussions for the staff. He stated that there are six key elements of the risk-informed regulation: (1) policy, (2) strategy , (3) knowledgeable staff, (4) decisionmaking, (5) tools, and (6) communication. The staff has been involved in working on the reactor safety policy and the Commission has directed the Office of Nuclear Material Safety and Safeguards (NMSS) staff to develop safety goals and an approach for risk informing in NMSS activities. There will certainly be key challenges in the development of safety goals for the non-reactor activities because they have a number of different areas that affect risk and have different levels of risk. In the area of strategy, the staff has a different PRA implementation plan because the staff was criticized by the General Accounting Office for not having a real strategy for the risk informing of agency activities. The staff is in the process of developing a strategy to cover the risk-informing PRA implementation plan. The third item is staffing, which also includes the licensee staffing. We have training programs and continue to review the training needs and the staffing level in this area. The fourth area is decisionmaking, which basically provides guidance documents both for the industry and the staff to utilize. In the area of improving tools in PRA methods, this could be classified as one of the impediments in the use of risk-informing regulation. Finally, in the area of communication, the pilot programs are a very effective way of communicating with the industry and the stakeholders, and they also have internal panel sessions that will educate the staff.

Industry Presentations

Mr. Thomas G. Hook, San Onfre Nuclear Generating Station, presented the following significant points regarding the impediments to risk-informed regulation:

- Difficulty in quantifying costs and benefits
- Variations in PRA quality and scope
- Duration of the regulatory review process
- Lack of PRA standards to establish quality

He also stated that the importance measures are acceptable only when augmented by the sensitivity analyses, and uncertainty analysis is underutilized by most licensees.

Mr. Rick C. Grantom, South Texas Project, presented the following significant issues in the areas of regulatory impediments, cultural impediments, and PRA institutional impediments.

Regulatory Impediments

- No regulatory limits exist for establishing the importance or the non-importance of components.

- There is no differentiation between design basis events and events that are likely to occur during plant life.

- There is no mechanism available or path to change safety-related component classification on the basis of risk information.

- Where is a lack of clarity or criteria in the degree to which risk-informed applications could be approved using only qualitative approaches versus those that use quantitative methods or both.

Cultural Impediments

- There is a lack of training that demonstrates the complementary effect of blending deterministic and probabilistic approaches in decisionmaking.

- There is a misconception that PRA analyses are too expensive relative to the benefits.

PRA Institutional Impediments:

- Where is limited availability of PRA practitioners for both NRC and utilities.
- Risk-ranking methods uncertainty analyses need further development.

Mr. Robert A. White, Palisades Nuclear Plant, briefly presented the following issues and alternatives in the area of importance measures:

- Importance measures can identify what is important but do not necessarily identify what is not important.
- Regardless of the methods used, sensitivity studies should be performed following classification of components into risk significance categories to confirm the classification.

Mr. White stated that there is a significant uncertainty regarding what it costs and how long it takes to obtain approval for a risk-informed submittal; even post-pilot plant submittals have taken significant resources.

Conclusion

The Committee issued reports to Chairman Meserve on importance measures and impediments dated February 11 and 15, 2000, respectively.

VII. Proposed Final Revision of Appendix K to 10 CFR Part 50

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

Dr. Uhrig noted that he is performing research that complements the work of the Caldon Corporation. Mr. Boehnert noted that on the basis of an e-mail from Mr. J. Szabo (Office of the Counsel) dated July 16, 1999, no conflict exists for Dr. Uhrig on this matter.

Dr. Wallis, Chairman of the Thermal-Hydraulic Phenomena Subcommittee, introduced this topic to the Committee. He noted that this issue involves revision of Appendix K to 10 CFR Part 50 to eliminate the requirement that licensees assume a core power level of 1.02 for emergency core cooling system analysis.

Licensees will now be able to propose a reduced value on the basis of the use of highly accurate flow instrumentation. Dr. Wallis noted that in a letter on the review of the proposed version of this rule, the Committee recommended that the staff ensure that this revision of Appendix K did not conflict with other requirements in the regulations and that the staff consider the impact of the reduction in uncertainties relative to regulatory margins for cases other than this rule version.

Mr. J. Donoghue, NRR, discussed the proposed final version of the revised Appendix K rulemaking. He recounted the key points of the Committee's review of the proposed version of the rulemaking. The proposed rule version was issued for public comment on October 1, 1999. The public comment period ended on December 15, 1999.

Six public comments were received. All responses were positive; clarifications were added to the Federal Register notice for the final rule version in response to these comments. No changes were made to the language of the rule, however.

During the discussion, Dr. Wallis opined that the staff needs to issue written guidance to licensees to account appropriately for power measurement uncertainty in their safety analyses for use of the new highly accurate flow measurement instrumentation. In response to questions from Dr. Kress, Mr. Wermiel, NRR, noted that RES has an effort underway to evaluate Appendix K for additional revisions. One of these areas is the requirement pertaining to decay heat analysis.

In response to Dr. Wallis, Mr. Wermiel said that the staff performed a search to ensure that this rule revision does not conflict with any other requirements of 10 CFR Part 50. Dr. Bonaca noted that his initial concern with this rule revision is that the staff should ensure consistent treatment of uncertainties for both pressurized water reactors and boiling water reactors.

Mr. H. Estrada, Caldon Corporation, provided remarks on the rule revision. He urged the staff to issue written guidance for implementation of this rule with regard to proper accounting of measurement uncertainties for use of the new flow instrumentation, noting that few engineers are skilled in measurement science. He also noted that Caldon had provided written comments to the staff and the ACRS during last year's review of this rule that included some suggested methodologies for combining uncertainties and for ensuring a rigorous demonstration of power measurement, including the bounding of modeling uncertainties.

Conclusion

The Committee provided a report on this matter to the Executive Director for Operations, dated February 11, 2000.

VIII. Subcommittee Report

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

Dr. George Apostolakis, Chairman of the Subcommittee on Reliability and Probabilistic Risk Assessment, provided a report regarding matters discussed during its meeting on December 15, 1999. He stated that the purpose of the meeting was to discuss the staff's programs for risk-based analysis of reactor operating experience, including special studies for common-cause failure analysis, system and component analyses, accident sequence precursor analyses, and related matters. On December 16, 1999, the Subcommittee discussed NRC staff efforts in the area of risk-informed technical specifications and associated industry initiatives proposed by the Risk-Informed Technical Specification Task Force.

Dr. Apostolakis recommended that the Committee consider preparing a letter on the staff's programs for risk-based analysis during future meetings. He also recommended that the Committee schedule briefings on risk-informed technical specification initiatives as the submittals become available. The Committee agreed to both recommendations.

IX. NRC Safety Research Program Report to the Commission

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

The Committee discussed its final draft of the ACRS Year 2000 report to the Commission on the NRC Safety Research Program.

Conclusion

The Committee finalized its final draft report on this matter and sent an advance copy to the Commission on February 7, 2000.

X. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations

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

- The Committee discussed the response from the NRC Executive Director for Operations (EDO) dated January 13, 2000, to the ACRS comments and recommendations included in the ACRS report dated December 8, 1999, concerning a draft Commission paper regarding elimination of the 120-month update requirement from 10 CFR 50.55a, "Codes and standards."

  The Committee was not satisfied with the EDO response. The Committee decided to prepare a reply to the EDO that would restate the Committee's recommendation that the 120-month update be retained.

- The Committee discussed the response from the NRC Executive Director for Operations (EDO) dated January 24, 2000, to the ACRS comments and recommendations included in the ACRS report dated December 10, 1999, concerning the safety aspects of the license renewal application for Calvert Cliffs Nuclear Power Plant.

  The Committee was satisfied with the EDO response.

- The Committee discussed the response from the NRC Executive Director for Operations dated January 20, 2000, to comments and recommendations of the joint ACRS/ACNW report dated November 17, 1999, concerning implementing a framework for risk-informed regulation in the Office of Nuclear Material Safety and Safeguards.

  The Committee referred this item to the ACRS/ACNW Joint Subcommittee for evaluation. The Committee plans to review the ACRS/ACNW Joint Subcommittee's recommendations during a future meeting.

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

The Committee heard a report from Dr. Powers and the Executive Director, ACRS, on the Planning and Procedures Subcommittee meeting held on February 2, 2000. The following items were discussed:

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

  Member assignments and priorities for ACRS reports and letters for the February ACRS meeting were discussed. Reports and letters that would benefit from additional consideration at a future ACRS meeting were also discussed. The Committee completed its response to the issues raised by the Commission in the SRM dated December 17, 1999, regarding: Impediments to the Increased Use of Risk-Informed Regulation; Use of Importance Measures in Risk-Informing 10 CFR Part 50; and Technical Components of the Revised Reactor Oversight Process, including the Technical Adequacy of the current and proposed performance indicators. The Committee completed its report regarding the 120-month ISI/IST update requirement in response to EDO comments. In addition, the Committee completed the annual report to the Commission on the NRC Safety Research Program and response to questions raised by individual Commissioners following the ACRS meeting with the Commission on November 4, 1999.

- Anticipated Workload for ACRS Members

  The anticipated workload of the ACRS members through April 2000 was discussed. The objectives were: (1) to review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate, (2) to manage the members' workload for these meetings, and (3) to plan and schedule items for ACRS discussion of topical and emerging issues.

- Meeting with the Commission

The ACRS will meet with the NRC Commissioners on March 2, 2000, to discuss risk-informing 10 CFR Part 50 and related matters.

- Schedule for the March ACRS Meeting

  The March ACRS meeting is scheduled for March 2-4, 2000. Several letters scheduled for the February meeting were deferred to the March meeting. There are other matters that will be reviewed by the Committee during the March meeting. In view of the heavy workload for the March meeting, the Committee will consider extending this meeting.

- Follow-up Items Resulting from the January 27-29, 2000 ACRS Retreat

  The positions agreed to during the ACRS retreat and the follow-up items resulting from the retreat were discussed.

- Status of Selecting Candidates for Potential ACRS Membership

  In response to solicitation of candidates for ACRS membership, we received approximately 20 applications. The ACRS Member Candidate Screening Panel reviewed all of the applications. The Panel selected four best-qualified candidates for interview by the Panel and the ACRS members. A schedule for the members and the Panel to interview three of these candidates during the March ACRS meeting was discussed.

- ACRS Self-Assessment Matrix

  In an SRM dated August 6, 1999, the Commission stated that "the periodic self-assessment report and the ACRS and ACNW Operating Plans can be combined into one annual report to the Commission that should include self-assessment summary matrices." In order to prepare the matrix, the ACRS staff needs to summarize the comments and recommendations included in the ACRS reports, which may result in interpreting the Committee positions. The Committee will approve the matrix and self-assessment report to preclude any

ambiguities and delegate the ACRS Executive Director to authorize or interpret Committee comments and recommendations.

- Change in Travel Requirements for Federal Employees

  Effective March 1, 2000, all Federal employees (including members) will be required to use their government issued credit card for all government travel expenses exceeding $75.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 470th ACRS Meeting, March 1-4, 2000.

The 469th ACRS meeting was adjourned at 1:30 p.m. on February 5, 2000.

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

March 21, 2000

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

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

SUBJECT: CERTIFIED MINUTES OF THE 469th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), FEBRUARY 3-5, 2000

I certify that based on my review of the minutes from the 469th ACRS full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

Dana A. Powers

Dana A. Powers, Chairman

March 21, 2000
Date

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

March 14, 2000

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 469th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - FEBRUARY 3-5, 2000

Enclosed are the proposed minutes of the 469th 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.

Please note that these minutes are being issued in two parts: (1) main body (working paper form), and (2) appendices. The appendices are being sent only to those members who have requested them.

Attachment:
As stated

Environmental Impacts of the Proposed Action

The Commission has completed its evaluation of the proposed action and concludes that the modifications to TSs are administrative in nature.

The proposed action will not increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action does not involve any historic sites. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, there are no significant nonradiological environmental impacts associated with the proposed action.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

As an alternative to the proposed action, the staff considered denial of the proposed action (*i.e.*, the "no-action" alternative). Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the Vermont Yankee Nuclear Power Station.

Agencies and Persons Consulted

In accordance with its stated policy, on December 13, 1999, the staff consulted with the Vermont State official, William Sherman, of the Vermont Department of Public Service regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated October 21, 1999, which is available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC. Publically available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

Dated at Rockville, Maryland, this 5th day of January 2000.

For the Nuclear Regulatory Commission.

Richard P. Croteau,

Project Manager, Section 2, Project Directorate I, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 00–610 Filed 1–10–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards will hold a meeting on **February 3–5, 2000, in** Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Thursday, October 14, 1999 (64 FR 55787).

Thursday, February 3, 2000

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

8:45 a.m.–10:45 a.m.: Technical Aspects Associated with the Revised Reactor Oversight Process and Related Matters (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the technical aspects associated with the revised reactor oversight process, including the updated significance determination process, plant performance indicators, and related matters.

11 a.m.–12 Noon: Proposed Final Amendment to 10 CFR 50.72 and 50.73 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding the proposed final amendment to 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," and 50.73, "Licensee Event Report System."

1 p.m.–2:30 p.m.: Proposed Regulatory Guide and Associated NEI Document 96–07, "Guidelines for 10 CFR 50.59 Safety Evaluations" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and NEI regarding the proposed Regulatory Guide, which endorses guidance in NEI 96–07, associated with the implementation of the revised 10 CFR 50.59 process.

2:45 p.m.–4:15 p.m.: Proposed Revision of the Commission's Safety Goal Policy Statement for Reactors (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding proposed revision of the Commission's Safety Goal Policy Statement for reactors and related matters, including industry views.

4:15 p.m.–5:15 p.m.: Break and Preparation of Draft ACRS Reports (Open)—Cognizant ACRS members will prepare draft reports for consideration by the full Committee.

5:15 p.m.–7:00 p.m.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting. In addition, the Committee will discuss proposed ACRS reports on: Low-Power and Shutdown Operations Risk Insights Report; License Renewal Process; and Response to Follow-up Questions Resulting from the ACRS Meeting with the Commission on November 4, 1999.

Friday, February 4, 2000

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

8:35 a.m.–10:30 a.m.: Impediments to the Increased Use of Risk-Informed Regulation and Use of Importance Measures in Risk-Informing 10 CFR Part 50 (Open)—The Committee will hear presentations by and hold discussions with representatives of NEI, the NRC staff as needed, and invited experts regarding impediments associated with the increased use of risk-informed regulation and use of importance measures in risk-informing 10 CFR Part 50, and related matters.

10:45 a.m.–11:30 a.m.: Proposed Final Revision of Appendix K to 10 CFR Part 50 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed

final revision of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

11:30 a.m.–11:45 a.m.: Subcommittee Report (Open)—The Committee will hear a report by the Chairman of the Reliability and Probabilistic Risk Assessment Subcommittee regarding matters discussed during the December 15–16, 1999 meeting.

11:45 a.m.–12:00 Noon: Report of the Joint ACRS/ACNW Subcommittee—The Committee will hear a report on matters discussed during the January 13–14, 2000 meeting of the Joint ACRS/ACNW Subcommittee.

1:00 p.m.–3:00 p.m.: NRC Safety Research Program Report to the Commission (Open)—The Committee will discuss the proposed final report to the Commission on the NRC Safety Research Program and related matters.

3:15 p.m.–3:30 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.

3:30 p.m.–3:45 p.m.: Future ACRS Activities (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.

3:45 p.m.–4:30 p.m.: Report of the Planning and Procedures Subcommittee (Open)—The Committee will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business.

4:30 p.m.–5:30 p.m.: Break and Preparation of Draft ACRS Reports (Open)—Cognizant ACRS members will prepare draft reports for consideration by the full Committee.

5:30 p.m.–7:15 p.m.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, February 5, 2000

8:30 a.m.–2 p.m.: Discussion of Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

2 p.m.–2:30 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 28, 1999 (64 FR 52353). In accordance with these procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify Mr. Sam Duraiswamy, ACRS, five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during this meeting may be limited to selected portions of the meeting as determined by the Chairman.

Information regarding the time to be set aside for this purpose may be obtained by contacting Mr. Sam Duraiswamy prior to the meeting. In view of the possibility that the schedule for ACRS 5 meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with Mr. Sam Duraiswamy if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting Mr. Sam Duraiswamy (telephone 301/415–7364), between 7:30 a.m. and 4:15 p.m., EST.

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

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

Dated: January 5, 2000.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 00–608 Filed 1–10–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Planning and Procedures

The ACRS Subcommittee on Planning and Procedures will hold a meeting on January 27–29, 2000, Radisson Suite Resort, Cedarwood #2 Room, 1201 Gulf Boulevard, Clearwater, Florida.

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

The agenda for the subject meeting shall be as follows:

Thursday, January 27, 2000—8:30 a.m. Until the Conclusion of Business

The Subcommittee will discuss issues related to PRA quality, including development of industrial standards; use of importance measures in risk-informing 10 CFR Part 50; impediments to the increased use of risk-informed regulation; technical aspects of the revised reactor oversight process, including technical adequacy of the current and proposed performance indicators; and safety culture.

Friday, January 28, 2000—8:30 a.m. Until the Conclusion of Business

The Subcommittee will discuss best estimate computer codes, technical quality of codes, and how they are used at the NRC. It will also discuss industry views of ACRS activities, self-assessment of ACRS performance in CY 1999, potential operational areas for improved effectiveness, other activities related to the conduct of ACRS business, and proposed response to follow-up questions resulting from the ACRS meeting with the Commission on November 4, 1999.

Saturday, January 29, 2000—8:30 a.m. Until 12:00 Noon

The Subcommittee will discuss ACRS positions on PRA issues, technical adequacy of the current and proposed performance indicators for the revised reactor oversight process, and potential future ACRS review activities.

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

January 6, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
469TH ACRS MEETING
FEBRUARY 3-5, 2000

THURSDAY, FEBRUARY 3, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - ~~8:45~~ 35 A.M. Opening Remarks by the ACRS Chairman (Open)
   - 1.1) Opening statement (DAP/JTL/SD)
   - 1.2) Items of current interest (DAP/NFD/SD)
   - 1.3) Priorities for preparation of ACRS reports (DAP/JTL/SD)

2) ~~8:45~~ 35 - ~~10:45~~ 11:00 A.M. Technical Aspects Associated with the Revised Reactor Oversight Process and Related Matters (Open) (JJB/MVB/MTM)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding the technical aspects associated with the revised reactor oversight process, including the updated significance determination process, technical adequacy of the current and proposed plant performance indicators, and related matters.

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

~~**10:45 - 11:00 A.M.**~~ 11:00 - 11:12 *****BREAK*****

3) ~~11:00~~ 11:12 - ~~12:00~~ 12:10 Noon Proposed Final Amendment to 10 CFR 50.72 and 50.73 (Open) (MVB/NFD)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding the proposed final amendment to 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," and 50.73, "Licensee Event Report System."

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

4) ~~1:00~~ 1:12 - ~~2:30~~ 37 P.M. Proposed Regulatory Guide and Associated NEI Document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations" (Open) (JDS/JJB/MME)
   - 4.1) Remarks by the Cognizant ACRS member
   - 4.2) Briefing by and discussions with representatives of the NRC staff and NEI regarding the proposed Regulatory Guide, which endorses guidance in NEI 96-07 associated with the implementation of the revised 10 CFR 50.59 process.

40 - 55
2:~~30~~ - 2:~~45~~ P.M. *****BREAK*****

55 - 35
5) 2:~~45~~ - 4:~~15~~ P.M. Proposed Revision of the Commission's Safety Goal Policy Statement for Reactors (Open) (GA/PAB)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding proposed revision of the Commission's Safety Goal Policy Statement for reactors and related matters, including industry views.

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

4:35-5:15pm Power Uprate Applications & Potential Synergistic Safety Issues (Cronenberg)

~~6) 4:15 - 5:15 P.M. Break and Preparation of Draft ACRS Reports~~ Cognizant ACRS members will prepare draft reports for consideration by the full Committee.

5:30-5:42 Discussion of Upcoming Commission Meeting in March

7) ~~5:15~~ 5:42 - 7:15 P.M. Discussion of Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

- Impediment Letter
- Importance Measures
- Follow On Questions
- Research Report

7.1) Low-Power and Shutdown Operations Risk Insights Report (GA/MTM)

7.2) Technical Aspects Associated with the Revised Reactor Oversight Process (JJB/MVB/MTM)

7.3) Proposed Final Amendment to 10 CFR 50.72 and 50.73 (MVB/NFD)

7.4) Proposed Regulatory Guide and Associated NEI Document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations" (JDS/JJB/MME)

7.5) License Renewal Process (MVB/RLS/NFD)

7.6) Response to Follow-up Questions Resulting from the ACRS Meeting with the Commission on November 4, 1999 (DAP/NFD/SD)

FRIDAY, FEBRUARY 4, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 - 10:~~30~~ 45 A.M. Impediments to the Increased Use of Risk-Informed Regulation and Use of Importance Measures in Risk-Informing 10 CFR Part 50 (Open) (TSK/GA/AS)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of NEI as well as invited experts regarding impediments associated with the increased use of risk-informed regulation and use of importance measures in risk-informing 10 CFR Part 50, and related matters.

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

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

11:00 - 37
10) ~~10:45~~ - 11:~~30~~ A.M. Proposed Final Revision of Appendix K to 10 CFR Part 50 (Open) (GBW/PAB)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussion with representatives of the NRC staff regarding the proposed final revision of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

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

49 - 12:05
11) 11:~~30~~ - ~~11:45 A.M.~~ P.M. Subcommittee Report (Open) (GA/MTM)

Report by the Chairman of the Reliability and Probabilistic Risk Assessment Subcommittee regarding matters discussed during the December 15-16, 1999 meeting.

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

37 - 11:49 AM
12) 11:~~45~~ - ~~12:00 Noon~~ Report of the Joint ACRS/ACNW Subcommittee (Open) (TSK/GA/MTM)

Report on matters discussed during the January 13-14, 2000 meeting of the Joint ACRS/ACNW Subcommittee.

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

05 20
13) 1:~~00~~ - 3:~~00~~ P.M. NRC Safety Research Program Report to the Commission (Open) (GBW/MME)

13.1) Remarks by the Subcommittee Chairman

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

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

20 40
3:~~00~~ - 3:~~15~~ P.M. *****BREAK*****

40 4:00
14) 3:~~15~~ - ~~3:30~~ P.M. Reconciliation of ACRS Comments and Recommendations (Open) (DAP, et al./SD, et al.)

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

15) ~~3:30~~ 4:00 - ~~3:45~~ 4:25 P.M. — Future ACRS Activities (Open) (DAP/JTL/SD)
Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee.

16) ~~3:45~~ 4:25 - ~~4:30~~ 35 P.M. — Report of the Planning and Procedures Subcommittee (Open) (DAP/JTL)
Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business.

17) ~~4:30~~ 35 - ~~5:30~~ 35 P.M. — Break and Preparation of Draft ACRS Reports
Cognizant ACRS members will prepare draft reports for consideration by the full Committee.

18) ~~5:30~~ 40 - ~~7:15~~ 05 P.M. — Discussion of Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:

2/4/00
5:45-6:16 120-Month letter (F)
2/5/00
9:50-10:00
8:30-8:40 & Importance (F)
10:02-10:18 & Measures
11:10-11:25
11[illegible]-12:30 Impediments (F)
10:26-11:10 PIs

18.1) Impediments to the Increased Use of Risk-Informed Regulation and Use of Importance Measures in Risk-Informing 10 CFR Part 50 (GA/TSK/AS)
18.2) Technical Aspects Associated with the Revised Reactor Oversight Process (JJB/MVB/MTM)
18.3) NRC Safety Research Program (GBW/MME)
18.4) Response to Follow-up Questions Resulting from the ACRS Meeting with the Commission on November 4, 1999 (DAP/NFD/SD) — ① Dana-gram ② Vote-agreed
18.5) Low-Power and Shutdown Operations Risk Insights Report (GA/MTM)
18.6) Proposed Final Amendment to 10 CFR 50.72 and 50.73 (MVB/NFD)
18.7) Proposed Regulatory Guide and Associated NEI Document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations" (JDS/JJB/MME)
18.8) Proposed Revision of the Commission's Safety Goal Policy Statement for Reactors (GA/PAB)
18.9) License Renewal Process (MVB/RLS/NFD)
6:33-6:40 18.10) Proposed Final Revision of Appendix K to 10 CFR Part 50 (GBW/PAB) (F)

SATURDAY, FEBRUARY 5, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

19) 8:30 - ~~2:00~~ 1:30 P.M. — Discussion of Proposed ACRS Reports (Open)
(12:00-1:00 P.M. - LUNCH) Continue discussion of proposed ACRS reports listed under Item 18.

~~20) 2:00 - 2:30 P.M.~~ — ~~Miscellaneous (Open) (DAP/JTL)~~
~~Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.~~

NOTE:

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

- **Number of copies of the presentation materials to be provided to the ACRS - 35.**

APPENDIX III: MEETING ATTENDEES

469TH ACRS MEETING
FEBRUARY 3-5, 2000

NRC STAFF (February 3, 2000)
A. Levin, OCM/RAM
B. Ott, OEDO
J. Shea, OEDO
A. Madison, NRR
S. Stein, NRR
B. Dean, NRR
G. Parry, NRR
D. Hickman, NRR
C. Holden, NRR
S. Sanders, NRR
S. Dinsmore, NRR
T. Boyce, NRR
D. Allsopp, NRR
M. Pohida, NRR
T. Frye, NRR
S. Magruder, NRR
M. Malloy, NRR
D. Fischer, NRR
D. Allison, NRR
A. Spector, NRR
S. Long, NRR
F. Akstulewicz, NRR
S. Wong, NRR
E. McKenna, NRR
S. West, NRR
R. Aulude, NRR
J. Andersen, NRR
T. Wof, RES
D. Yielding, RES
B. Brady, RES
R. Spence, RES
J. Ibarra, RES
J. Mitchell, RES
T. King, RES
J. Murphy, RES
P. Brockman, NMSS

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
F. Mashburn, TVA
R. Huston, Licensing Support Services
D. Raleigh, SERCH, Bechtel Power
T. Houston, NEI
B. Post, NEI
J. Davis, NEI
R. Bell, NEI
C. Amoruso, NUS
N. Chapman, SERCH/Bechtel
H. Fonticella, VA Power
P. Negus, GE
B. Bradley, NEI

NRC STAFF (February 4, 2000)
M. Drouin, RES
J. Flack, RES
J. Costello, RES
T. Jackson, RES
J. Mitchell, RES
A. Thadani, RES
F. Eltawila, RES
M. Virgilio, NMSS
R. Boyle, NMSS
S. Wong, NRR
G. Parry, NRR
M. Cheok, NRR
S. Dinsmore, NRR
S. West, NRR
R. Aulude, NRR
F. Akstulewicz, NRR
J. Williams, NRR
J. Donoghue, NRR
R. Caruso, NRR
J. Wermiel, NRR
N. Gilles, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
B. Bogan, CMS Energy, Palisades
R. Huston, Licensing Support Services
B. Youngblood, ISL, Inc.

H. Fonticella, VA Power
B. Horin, Winston & Strawn
J. Regan, Key Technologies, Inc.
P. Negus, GE
S. Rosen, STPNOC

Appendix IV

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

February 11, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
470TH ACRS MEETING
MARCH 1-4, 2000

WEDNESDAY, MARCH 1, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 1:00 - 1:15 P.M. Opening Remarks by the ACRS Chairman (Open)
   - 1.1) Opening statement (DAP/JTL/SD)
   - 1.2) Items of current interest (DAP/NFD/SD)
   - 1.3) Priorities for preparation of ACRS reports (DAP/JTL/SD)

2) 1:15 - 3:15 P.M. Development of Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Open) (GA/MTM)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding the status of developing risk-informed revisions to 10 CFR Part 50 and related matters.

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

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

3) 3:30 - 6:00 P.M. Discussion of Proposed ACRS Reports (Open)
   Discussion of proposed ACRS reports on:
   - 3.1) Low-power and Shutdown Operations Risk Insights Report (GA/MTM)
   - 3.2) Proposed Revision of the Commission's Safety Goal Policy Statement for Reactors (TSK/GA/PAB)

6:00 - 6:15 P.M. *****BREAK*****

4) 6:15 - 7:15 P.M. Discussion of Topics for Meeting with the NRC Commissioners (Open)
   Discussion of issues associated with risk-informed regulation, including:
   - 4.1) Impediments to the increased use of risk-informed regulation (TSK/MTM)
   - 4.2) Use of importance measures in regulatory applications, impact of the scope and quality of the PRA on importance measures, and threshold values for importance measures (GA/AS)
   - 4.3) Technical Adequacy of Performance Indicators (JJB/NFD)

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

5) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (DAP/SD)

6) 8:35 - 9:15 A.M. Discussion of Topics for Meeting with the NRC Commissioners (Open)
Discussion of topics listed under Item 4.

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

7) 9:30 - 11:30 A.M. Meeting with the NRC Commissioners (Open) (DAP, et al./JTL, et al.)
Meeting with the NRC Commissioners, Commissioners' Conference Room, One White Flint North, to discuss topics listed under Item 4 and other items of mutual interest.

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

8) 1:00 - 2:30 P.M. Technical Components Associated with the Revised Reactor Oversight Process (Open) (JJB/MTM)
8.1) Remarks by the Subcommittee Chairman
8.2) Briefing by and discussions with representatives of the NRC staff regarding the technical components associated with the revised reactor oversight process, including the updated significant determination process, technical adequacy of the current and proposed plant performance indicators, and related matters.

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

9) 2:45 - 4:00 P.M. Oconee Nuclear Power Plant License Renewal Application (Open) (MVB/RLS/NFD)
9.1) Remarks by the Subcommittee Chairman
9.2) Briefing by and discussions with representatives of the NRC staff and Duke Energy Corporation regarding the license renewal application for the Oconee Nuclear Power Station and the associated NRC staff's Safety Evaluation Report.

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

10) 4:15 - 4:45 P.M. Proposed Final Amendment to 10 CFR 50.72 and 50.73 (Open) (MVB/NFD)
10.1) Remarks by the Subcommittee Chairman
10.2) Discussions with representatives of the NRC staff regarding issues raised by the ACRS members during the February ACRS meeting, including the intent of the 10 CFR 50.73 requirement for reporting degraded components.

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

11) 4:45 - 5:15 P.M. Proposed Final Revision 3 to Regulatory Guide 1.160, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants" (Open) (JJB/JDS/AS)

11.1) Remarks by the Subcommittee Chairman

11.2) Discussions with representatives of the NRC staff, as needed, regarding the proposed final revision 3 to Regulatory Guide 1.160.

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

12) 5:15 - 6:15 P.M. Break and Preparation of Draft ACRS Reports
Cognizant ACRS members will prepare draft reports for consideration by the full Committee.

13) 6:15 - 7:15 P.M. Discussion of Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:

13.1) Technical Components Associated with the Revised Reactor Oversight Process/Technical Adequacy of the Current and Proposed Performance Indicators (JJB/MVB/MTM)

13.2) Proposed Final Amendment to 10 CFR 50.72 and 50.73 (MVB/NFD)

13.3) Proposed Final Revision 3 to Regulatory Guide 1.160 (JJB/JDS/AS)

13.4) Oconee License Renewal Application (MVB/RLS/NFD)

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

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

15) 8:35 - 10:15 A.M. Phenomena Identification and Ranking Table (PIRT) for High Burnup Fuel (Open) (DAP/MME)

15.1) Remarks by the Subcommittee Chairman

15.2) Briefing by and discussions with representatives of the NRC staff regarding the use of PIRT process for high burnup fuel.

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

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

16) 10:30 - 11:30 A.M. Proposed Resolution of Generic Safety Issue B-17, "Criteria for Safety Related Operator Actions" (Open) (RLS/PAB)

16.1) Remarks by the Subcommittee Chairman

16.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed resolution of Generic Safety Issue B-17.

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

17) 11:30 - 12:00 Noon Report of the Planning and Procedures Subcommittee (Open) (DAP/JTL)

Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business.

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

18) 1:00 - 1:15 P.M. Future ACRS Activities (Open) (DAP/JTL/SD)

Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee.

19) 1:15 - 1:30 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (DAP, et al./SD, et al.)

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

20) 1:30 - 2:30 P.M. Break and Preparation of Draft ACRS Reports

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

21) 2:30 - 7:00 P.M. Discussion of Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

21.1) Oconee License Renewal Application (MVB/RLS/NFD)

21.2) Proposed Resolution of Generic Safety Issue B-17 (RLS/PAB)

21.3) Low-power and Shutdown Operations Risk Insights Report (GA/MTM)

21.4) Proposed Revision of the Commission's Safety Goal Policy Statement for Reactors (TSK/GA/PAB)

21.5) Technical Components Associated with the Revised Reactor Oversight Process/Technical Adequacy of the Current and Proposed Performance Indicators (JJB/MVB/MTM)

21.6) Proposed Final Amendment to 10 CFR 50.72 and 50.73 (MVB/NFD)

21.7) Proposed Final Revision 3 to Regulatory Guide 1.160 (JJB/JDS/AS)

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

22) 8:30 - 1:30 P.M. **(12:00-1:00 P.M. - LUNCH)**	Discussion of Proposed ACRS Reports (Open) Continue discussion of proposed ACRS reports listed under Item 21.
23) 1:30 - 2:00 P.M.	Miscellaneous (Open) (DAP/JTL) Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

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

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
469th ACRS MEETING
FEBRUARY 3-5, 2000

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

MEETING HANDOUTS

AGENDA ITEM NO.	DOCUMENTS
1	Opening Remarks by the ACRS Chairman
	1. Items of Interest, dated February 3-5, 2000
2	Technical Aspects Associated with the Revised Reactor Oversight Process and Related Matters
	2. Revised Reactor Oversight Process Pilot Program Results and Lessons Learned, presentation by W. Dean, A. Madison, M. Johnson, G. Parry, NRR [Viewgraphs]
	3. Letter to Tennessee Valley Authority, Subject: Inspection Plan for Sequoyah, dated December 21, 1999
	4. Letter to Commonwealth Edison Company, Subject: Inspection Plan-Quad Cities Nuclear Power Station, dated December 22, 1999
	5. Letter to Nebraska Public Power District, Subject: Inspection Plan-Cooper Nuclear Station, dated December 27, 1999
	6. Letter to New York Power Authority, Subject: Mid-Cycle Performance Review and Inspection Plan-James A. Fitzpatrick Nuclear Power Plant, dated January 3, 2000
3	Proposed Final Amendment to 10 CFR 50.72 and 50.73
	7. Draft Final Rule-Modification of Event Reporting Requirements 10 CFR 50.72 and 50.73, presentation by NRR [Viewgraphs]
	8. Proposed Final Amendment to 10 CFR 50.72 and 50.73, e-mail from the NRC staff dated January 28, 2000, Subject: "Corrected Copy of Noteworthy Issues" [Handout 3.1]
	9. Licensee Event Reporting System, presentation by J. Davis, NEI
4	Proposed Regulatory Guide and Associated NEI Document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations"
	10. Status of 10 CFR 50.59 Guidance, presentation by E. McKenna, NRR [Viewgraphs]
	11. NEI 96-07, Revision 1, Guidelines for 10 CFR 50.59 Evaluations,

presentation by R. Bell, NEI [Viewgraphs]

12. Memorandum dated January 31, 2000, Subject: NEI Document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations [Handout 4)

5 Proposed Revision of the Commission's Safety Goal Policy Statement for Reactors

13. Modifications to the Reactor Safety Goal Policy Statement, presentation by J. Murphy, RES [Viewgraphs]
14. ACRS Proposed Review of the Commission's Safety Goal Policy Statement for Reactors, P. Boehnert [Handout 5-1]

5a Review of Power Uprate Applications and Potential Synergistic Safety Issues

15. Presentation by G. Cronenberg, Senior Fellow [Viewgraphs]

9 Impediments to the Increased Use of Risk-Informed Regulation and Use of Importance Measures in Risk-Informing 10 CFR Part 50

16. Risk-Informed Regulation - Challenges and Importance Measures, presentation by T. King, RES, G. Holahan, NRR, M. Virgilio, NMSS [Viewgraphs]
17. Risk Informing 10 CFR 50, Top Event Prevention (TEP) A Deterministic Application of PSA, presentation by CMS Energy [Viewgraphs]
18. Importance Measures (Issues/Alternatives), presentation by CMS Energy [Viewgraphs]
19. Impediments to Risk-Informed Regulation and Risk Importance Measures, presentation by T. Hook, Manager, Nuclear Safety Oversight, San Onofre Nuclear Generating Station [Viewgraphs]
20. Impediments to Risk Informed Regulation, presentation by STP Nuclear Operating Company [Viewgraphs]
21. Risk Ranking of All PSA Basic Event, presentation by South Texas Project [Viewgraphs]

10 Proposed Final Revision of Appendix K to 10 CFR Part 50

22. Appendix K Rulemaking, Final Rule Change Revising the 102% Power Level Requirement, presentation by J. Donoghue, NRR [Viewgraphs]

14 Reconciliation of ACRS Comments and Recommendations

23. Reconciliation of ACRS Comments and Recommendations [Handout #14.1]

15 Future ACRS Activities

24. Future ACRS Activities - 470th ACRS Meeting, March 2-4, 2000 [Handout #15-1]

16 Report of the Planning and Procedures Subcommittee

25. Final Draft Minutes of Planning and Procedures Subcommittee Meeting - February 2, 2000 [Handout #16.1]

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Technical Aspects Associated with the Revised Reactor Oversight Process and Related Matters
	1. Table of Contents
	2. Proposed Schedule
	3. Status Report, dated February 3, 2000
	4. Note dated January 27, 2000, from John J. Barton, ACRS, to Michael Johnson, NRR, Subject: Issues and questions for February 3 ACRS meeting
	5. E-mail message dated January 23, 2000, from Jack D. Sieber, ACRS, Subject: Use of Pis (predecisional)
	6. E-mail dated January 22, 2000 from T. S. Kress (predecisional)
	7. Facsimile dated January 22, 2000, from D. A. Powers (predecisional)
	8. SRM dated December 17, 1999, Subject: Meeting with the ACRS
	9. Letter dated November 23, 1999, from Samuel J. Collins, Director, NRR, Subject: Request for review of revised reactor oversight program
	10. SRM dated June 18, 1999, Subject: SECY 99-007 and SECY 99-007A
	11. Letter dated June 10, 1999, from Dana A. Powers, Chairman, ACRS, Subject: Inspection/assessment programs, PIs & performance-based initiatives
	12. Letter dated August 9, 1999, from William D. Travers, EDO, NRC, to Dana A. Powers, Chairman, ACRS, Subject: EDO response to ACRS letter
3	Proposed Final Amendment to 10 CFR 50.72 and 50.73
	13. Table of Contents
	14. Proposed Schedule
	15. Status Report dated February 3, 2000
	16. Letter dated March 23, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director, NRC, Subject: Proposed Amendment to 10 CFR 50.72, Immediate Notification and 50.73, Licensee Event Reporting System
	17. Letter dated April 19, 1999, from William D. Travers, Executive Director, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Proposed Rulemaking to Modify the Reactor Event Reporting Requirements in 10 CFR 50.72 and 50.73
	18. Memorandum dated June 15, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY 99-119, Rulemaking to Modify the Event Reporting Requirements for Power Reactors in 10 CFR 50.72 and 50.73

19. E-mail dated January 24, 2000, Noteworthy Issues Associated with the Proposed Final Revision to 10 CFR 50.72 and 50.73

4 Proposed Regulatory Guide and Associated NEI Document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations"

20. Table of Contents
21. Proposed Schedule
22. Status Report dated February 3, 2000
23. ACRS Report dated May 17, 1999
24. Staff Requirements Memorandum, dated June 22, 1999
25. NEI 96-07 (Draft Rev. 1C), dated December 30, 1999 (predecisional)
26. Table-Resolution Status of NRC Nov. 3 comments

5 Proposed Revision of the Commission's Safety Goal Policy Statement

27. Table of Contents
28. Proposed Schedule
29. Status Report dated February 3, 2000
30. Memorandum dated January 27, 2000, from Joseph A. Murphy, RES, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Draft Commission Paper on Proposed Modifications to the Reactor Safety Goal Policy Statement, and attachments (predecisional-for internal ACRS use only)
31. SRM dated October 28, 1999, on SECY 99-191 re: Safety Goals
32. SRM dated October 16, 1997, on SECY 97-208 re: Elevation of CDF
33. Report dated April 19, 1999, from Dana A. Powers, Chairman, ACRS to Shirley Ann Jackson, Chairman, NRC, Subject: Status of Efforts on Revising the Commission's Safety Goal Policy Statement
34. Letter dated May 24, 1999, from William D. Travers, EDO, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Status of Efforts on Revising the Commission's Safety Goal Policy Statement
35. Report dated May 11, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Elevation of CDF to a Fundamental Safety Goal and Possible Revision of the Commission's Safety Goal Policy Statement

9 Impediments to the Increased Use of Risk-Informed Regulation and Use of Importance Measures in Risk Informing 10 CFR Part 50

36. Memorandum dated December 17, 1999, to John T. Larkins, Executive Director, ACRS, from Annette L. Vietti-Cook, Secretary, NRC, Subject: Staff Requirements - Meeting with ACRS on Thursday, November 4, 1999
37. Report to Greta Joy Dicus, Chairman, NRC, from Dana A. Powers,

Chairman, ACRS, Subject: Proposed Plans for Developing Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" dated October 12, 1999

38. Letter dated November 8, 1999, from William d. Travers, Executive Director for Operations, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Proposed Plans for Developing Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing Production and Utilization Facilities"
39. Statement of James P. Ricco, Staff Attorney, Public Citizen's Critical Mass Energy Project, to ACRS dated September 30, 1999
40. Report to Shirley Ann Jackson, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, Subject: The Role of Defense in Depth in a Risk-Informed Regulatory System dated May 19, 1999

10 <u>Proposed Final Revision of Appendix K to 10 CFR Part 50</u>

41. Table of Contents
42. Presentation Schedule
43. Project Status Report dated February 4, 2000
44. Note to P. Boehnert from J. Donoghue, "Final Rule Package for Appendix K Revision," dated January 19, 2000
45. Public Comments Received on Proposed Revision to the Appendix K Revision
46. Letter to W. D. Travers, EDO, from D. A. Powers, ACRS, Subject: Revision of Appendix K, "ECCS Evaluation Models: to 10 CFR Part 50, dated July 22, 1999
47. Letter to D. A. Powers, ACRS, from W. D. Travers, EDO, Subject: Staff Response to ACRS Letter of July 22, 1999, on Revision of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, dated August 18, 1999
48. Excerpt from Minutes of 464th ACRS Meeting: Proposed Revision to Appendix K of 10 CFR Part 50

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

469TH FULL COMMITTEE MEETING

FEBRUARY 3-5, 2000
Date(s)

FEBRUARY 3, 2000
Today's Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	BADGE #	AFFILIATION
ALAN MADISON	B8558	NRR
Steven Stein	B8064	NRR
Bill Dean	B6175	NRR
Garett Parry	B8060	NRR - DSSA
DON HICKMAN	B6244	NRR
C. Holden	B 7813	NRR
Serita Sanders	B-7451	NRR
Stephen Dinsmore	B-7898	NRR
TOM BOYCE	B-8610	NRR
DAVID ALLSOPP	B-7812	NRR
Marie POHIDA	B-7908	NRR
Alan Levin	B-6675	OCM/RAM
Tim Frye	B-8578	NRR
TOM WOLF	B-7299	RES
Stan Ornstein	B-8390	RES
Phil Brochman	A-7353	NMSS
DALE YEIGOING	B-7450	RES
STEWART MAGRUDER	B-6721	NRR

Melinda Malloy	B-8526	NRR
DAVID FISCHER	B-6112	NRR
Dennis Allison	A-6104	NRR
Kenneth Brady	B-6363	RES
August Spector		NRR
Robert Spence	B-8421	RES
Steve Long	A 7141	NRR
Bill Olx	B6886	EDO
FRANCIS AKSTULEWICZ	B6013	NRR
See-Meng Wong	B-8181	NRR
Eileen McKenna	B2226	NRR
Cindi Carpenter	B6464	NRR
Jose Ibarra	B8398	RES
Jocelyn Mitchell	B6685	RES
Steven West	B7258	NRR
T. King	B7941	RES
J Murphy	B6558	RES
C J Paperiello	A-7065	EDO
Raj Anand	B-6030	NRR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

469TH FULL COMMITTEE MEETING

FEBRUARY 3-5, 2000
Date(s)

FEBRUARY 3, 2000
Today's Date

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Roger Huston	Licensing Support Services
Deann Raleigh	SERCH, Bechtel Power
Tom Houghton	NEI
Bob Post	NEI
James Davis	NEI
James Andersen	NRR
Joe Shea	OGC
Russ Bell	NEI
Gilles Amaruso	NUS Info Services
Nancy Chapman	SERCH / Bechtel
H Fontecilla	VA Power
~~Jose~~ T	
Paige Negus	GE
Biff Bradley	NEI

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

469TH FULL COMMITTEE MEETING

FEBRUARY 3-5, 2000
Date(s)

FEBRUARY 4, 2000
Today's Date

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Mary Drouin	B7583	RES/DRAA
Matt Vaughan	A6798	NRC
See-Meng Wong	B-8181	NRR
Garth W. Parry	B-8060	NRC-NRR
Mike Cheok	B-7917	NRC-NRR
John FLACK	B-6114	NRC/RES
Stephen Dinsmore	B-7898	NRR
Steve West	B-7258	NRR
Raj Anand	B-6030	NRR
Regis Boyle	A-6277	NMSS
F. Akstulewicz	B6013	NRR
JF Costello	B6395	RES
JOE WILLIAMS	B7257	NRR/DLPM
Terry Jackson	B7777	RES
J Donoghue	B6478	NRR
R. CARUSO	B7988	NRR
Jared Wermiel	A6824	NRR/DSSA
Nanette Gilles	B8580	NRR/DRIP

Jocelyn Mitchell	B6685	RES
Ashok Thadani	A6747	RES
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

469TH FULL COMMITTEE MEETING

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Jennifer Regan	Key Technologies Inc.
Kaye Nelson	GE
S. ROSEN	STPNOC
~~FAROUK ELTA~~	

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ITEMS OF INTEREST

469TH ACRS MEETING

FEBRUARY 3-5, 2000

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
469th MEETING
FEBRUARY 3-5, 2000

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

January 29, 2000

CHAIRMAN

Dr. Dana A. Powers, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear ~~Dr. Powers~~ Dana:

My fellow Commissioners and I want to express our appreciation for your efforts and those of the Advisory Committee on Reactor Safeguards associated with the Calvert Cliffs license renewal application. Your expeditious and thorough review of the staff's evaluation and licensing activities was of great benefit to us.

As you are aware, numerous licensees have indicated an intention to submit similar applications in the near future, in large part, I believe, because of the capacity we have demonstrated to evaluate renewal applications in a timely fashion. We look forward to your continued support in evaluating appropriate aspects of the staff's future initiatives in this area.

Best regards,

Very truly yours,

Richard A. Meserve

Nuclear Regulatory Commission

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S-99-37

Chairman Richard A. Meserve

Keynote Address for 1999 State Liaison Officers' National Meeting

Wednesday, December 1, 1999

Good morning and welcome to the 1999 State Liaison Officers' meeting. As you know, I have served as Chairman of the Nuclear Regulatory Commission for only one month. I suspect that many of you may have forgotten things that I have yet to learn. Nonetheless, I welcome the opportunity to share some perspectives with you.

In my prior life as a practicing attorney, I had many interactions both with Agreement States and other states on licensing and regulatory issues. I was thus aware of the role of the states in assuring public health and safety. My short tenure as Chairman has served only to reinforce the importance of the NRC's linkage with you if the NRC is to achieve its fundamental mission. I thus view our relationship as essential to the NRC's success and I am hopeful that it will continue to be cooperative and productive. As a result, welcome the opportunity to meet with you early in my term.

We are in a period of remarkable innovation at the NRC. Let me discuss some of the broad areas of change at the NRC and then turn to some particular issues of significance that bear on our relationship with the states.

I. NRC Issues

As I am sure you are fully aware, we are in the middle of a significant restructuring of the utility industry. In a growing number of states, the competitive market determines the price of electricity and thus profitability for all forms of electricity generation is dependent on achieving economically efficient operations. This has important implications for the NRC's work.

On the one hand, the changed financial environment reinforces the need for us to be vigilant in demanding safe operations by licensees. The NRC must assure that the pressures to reduce costs do not become incentives to cut corners on safety. Protection of the public health and safety will always remain our transcendent mission and responsibility.

On the other hand, this changed environment reinforces the need for the NRC to regulate efficiently -- to regulate in a fashion that imposes the minimum degree of burden consistent with getting the job done. This implies a careful approach both in crafting new regulatory initiatives -- making sure that the benefits outweigh the costs -- and a willingness to cast a critical eye on existing policies and practices. In this connection, the Commission is engaged in a comprehensive effort to reevaluate the entire foundation for the regulation of reactors by seeking to apply the insights arising from probabilistic assessments in modifying regulatory requirements. This is termed "risk-informed" regulation and it will consume the efforts of many of our staff over the next several years.

In addition to the application of risk insights in regulations, the Commission is pursuing reactor license renewal. As I am sure you know, the Atomic Energy Act authorizes the NRC to issue a reactor license for a 40-year term, while providing the possibility of license renewal. In light of the fact that many reactors can safely be operated for an additional period of time beyond 40 years, the NRC has established a process to allow license renewals for up to an additional 20 years in appropriate cases. The first two applications for renewal are now in process and others are expected.

The NRC has recently issued its safety evaluation report on the proposed renewal of the operating license for the two Calvert Cliffs nuclear power plants, concluding that there are no safety concerns that would preclude renewal of the license. The renewal of the license for the three Oconee nuclear units remains on schedule. As I am sure you know, the Commission received an adverse ruling by the U.S. Court of Appeals for the D.C. Circuit concerning the request by the National Whistleblowers Center to intervene in the Calvert Cliffs proceeding. But, in a striking development, the court recently vacated that decision and ordered further briefing and argument. I do not believe that this case, regardless of the outcome, will constitute a significant setback in our efforts to assure the timely processing of renewal applications.

In the materials and waste area, many challenges are also looming. In the years ahead, the agency will have to grapple with the problems associated with the geologic disposal of high-level waste -- a task that will present very thorny technical, legal, social, and political problems. The consideration of issues associated with Yucca Mountain, if the Department of Energy decides to pursue licensing, are sure to be trying and difficult. Moreover, the associated transportation issues will certainly have implications for the states. In the interim, we must continue to address the issues associated with dry cask storage of spent nuclear fuel. In addition, more nuclear utilities are beginning to decommission, requiring a more effective regulatory framework. In this area, utilities are seeking new ways to satisfy the License Termination Rule while reducing decommissioning costs. The staff will be challenged to consider new concepts under a performance-oriented approach while ensuring that radiological criteria are met.

It will also be necessary and appropriate to apply in the materials context some of the lessons learned from the development of a risk-informed and performance-based approach to the regulation of reactors. There are four major categories of regulated materials activities that would benefit from greater application of risk- assessment methods : 1) the long-term commitment of a site or facility to the presence of nuclear material (e.g., high-level waste disposal); 2) use of engineered casks to isolate nuclear material under a variety of normal and off-normal conditions; 3) physical and chemical processing and possession of nuclear material at a large-scale facility (e.g., fuel fabrication); and 4) use of either sealed or unsealed byproduct material in industrial and medical applications. However, the characteristics of nuclear materials regulation differ in important respects from those relating to reactor regulation -- materials regulations are driven by exposure standards, as opposed to measures of facility damage; there are a far wider diversity of activities undertaken by materials licensees than by reactor licensees; materials activities are not dominated by a clear-cut risk feature, such as core damage; and operational risk, as opposed to accident risk, may be the central feature of the regulation of materials. Nonetheless, despite these differences, we believe the application of risk insights can and should be applied to materials regulation in the years ahead. This, of

course, will have implications for counterpart state regulation, as I will mention in a moment.

Finally, I must also mention the need for the NRC to maintain its connection to the broader international community. As the recent incident in Japan has revealed, an event anywhere in the world can cause heightened concern about nuclear-related enterprises everywhere. As a result, the NRC needs to continue to work with its counterparts abroad to advance nuclear safety throughout the globe. Moreover, in a world that is awash with plutonium and highly enriched uranium, we need to work internationally to find ways to reduce the risks that these materials present.

II. State Issues

Let me turn now to a variety of issues or concerns that have arisen over my few weeks at the NRC that bear directly on the work of the states. I welcome your perspectives on these matters because I understand the need for the NRC to work cooperatively with the states to achieve appropriate resolution of these matters. I will discuss five issue areas. These include the need for consistency, the allocation of lead responsibility, consideration of a clearance standard, rulemakings affecting states, and assuring public confidence.

1. Consistency. There is clearly a need for consistency in regulatory approaches across international, national and State boundaries, particularly with respect to regulations that affect international trade and interstate commerce. There is an ongoing need for the NRC to work with international and other federal agencies on the development of guidelines for controlling the import or export of contaminated materials and products, as well as with the states on the identification and recovery of orphan sources and the control of licensed devices to minimize loss or inadvertent disposal. In light of the increasing concerns that are being expressed by metal recyclers, I expect that concerted action will be required in these areas at the international, national and state levels in the years ahead.

Moreover, there is a need to make greater efforts to assure consistency with respect to regulatory approaches, particularly for those actions that have impacts across state boundaries. For example, manufacturers and distributors of products that are shipped throughout the United States now frequently must deal with different licensing practices and regulations in as many as 31 different Agreement States, in addition to the NRC requirements for the rest of the country. It is important that we take steps to minimize the costs and confusion that can arise from the diversity of regulatory approaches.

As one minor step that might facilitate greater consistency, let me mention a suggestion that was made by my fellow Commissioner, Jeff Merrifield, at a recent nuclear materials stakeholders' workshop on November 9. He suggested that the NRC's website might provide a link to and a common source of information about proposed changes to state regulations, thereby enhancing awareness of changes by potentially impacted groups or individuals and helping assure that states receive comments during rulemakings concerning problems of inconsistency. We are considering this suggestion, but no doubt other efforts to encourage consistency are also in order.

2. Lead Responsibility. The Commission will need to examine the implications of the fact that we now have 31 Agreement States -- perhaps soon 35 Agreement States -- and that the states now regulate the majority of materials licensees. Traditionally, although the NRC has a well defined program of consultation with Agreement States, the NRC has been the initiator of regulatory changes. It may now be appropriate for the states to assume a greater responsibility for undertaking regulatory revision. The Organization of Agreement States and the NRC have effectively used staff working groups to address a number of regulatory issues. In addition, the Conference for Radiation Control Program Directors (the "CRCPD") has worked effectively in the past with the NRC and other Federal agencies to develop model

state regulations for radiation protection. Perhaps there is the potential for it to take a leadership role in developing regulatory initiatives in the materials area. In any event, the NRC will establish a working group with Agreement State and CRCPD participation to examine the potential framework through which the regulation of nuclear materials can be accomplished in the future.

3. Clearance Standard. One issue that is likely to attract significant attention of the coming year relates to the NRC's exploratory efforts to determine whether to develop a national standard for clearance of material with small amounts of residual radioactivity. In June the NRC published an issues paper in the Federal Register to start the process of obtaining the views of stakeholders on this issue. The matter is attracting increased attention as a result of the recent public and Congressional interest in the decision by the State of Tennessee to authorize the release of volumetrically contaminated nickel that is recycled from a DOE facility and would contain trace amounts of fission products. There is an argument for a consistent national approach to such matters, particularly since material that is released by one state may well be used in another state. All states should have a keen interest in this subject since it goes beyond material regulated under the Atomic Energy Act to include Technologically Enhanced Natural Radioactive Material (TENORM), which is an area of exclusive state jurisdiction.

4. Rulemakings. I should also mention a number of ongoing NRC initiatives that will have an impact on the Agreement State programs. The proposed final rule for Part 35, "Medical Uses of Byproduct Material," is now before the Commission for final action. That rule uses risk insights, together with other factors, to establish requirements that better focus licensee and regulatory attention on design and operational issues that have importance to health and safety. The staff worked closely with the CRCPD SR-6 committee on a parallel rulemaking approach to develop a draft "Suggested State Regulation" for Part G, "Medical Uses of Radionuclides." Let me add that in addition to undertaking the modification of Part 35, the Commission is considering a related initiative to risk-inform 10 CFR Part 40, "Domestic Licensing of Source Material," which could also have implications for state regulations applicable to TENORM.

5. Public Confidence. We all need to remember that there is an overarching obligation on the part of the NRC and the states to meet all these challenges in a fashion that justifiably enhances public confidence. We work in a field in which public concern can easily arise. Our decisions must be fair, and must be perceived as being fair. Moreover, providing a fair opportunity for all to speak their views is essential; the public simply will not accept decisions from which it has been excluded. Thus, we must all approach our work in a way that includes the affected public in ways that are meaningful and that contribute to sound decisions.

* * * *

In sum, we have an abundance of challenges to confront together. I look forward to working with you as we address them and appreciate the opportunity to meet with you this morning.

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BENEFITS OF SAFETY-FOCUSED REGULATION

Commissioner Nils J. Diaz
United States Nuclear Regulatory Commission

1999 ANS Winter Meeting, Long Beach, CA

Good Morning, Ladies and Gentlemen. I'm very happy to have this opportunity to discuss a regulator's view of the road ahead for nuclear technology as we enter a new millennium. The accelerating pace of industrialization and the expansion of market economies throughout the world continue to create opportunities and challenges for all areas of endeavor, including the generation of electric power by nuclear energy. Change is here, and everywhere; change is here to stay.

Although I am speaking to you as a member of the Nuclear Regulatory Commission, I will be offering my individual views today.

[*Figure 1*] Economic deregulation is a reality in the United States and in many places abroad. 16 of the 31 states in the United States of America with operating nuclear power plants have already decided to deregulate electricity supply. Sustained performance improvements at nuclear power plants, license renewals, sales of existing plants, and mergers are making headlines. And yes, there are regulatory changes. The question, therefore, is not whether to change or not to change, but how to make change serve this country, and serve other countries also.

[*Figure 2*] "Rio revuelto, ganancia de pescadores" (translation: in a murky, turbulent river, fishermen profit).

In a Wall Street Journal article of June 18, 1997, two key problems facing nuclear power plants were raised in the context of forced early shutdowns: the safety and the cost competitiveness of nuclear power plants. Those were the times of Millstone and design-bases compliance, and of the doomsday predictions of the effects of de-regulation and stranded costs. Two dozen early shutdowns of plants with "marginal safety" and/or cost were forecast by many; up to 50% of the fleet by some. The Wall Street article stated: "more conservatively, NRC Commissioner Nils Diaz estimates only one dozen early shutdowns." There have been 6.

In another Wall Street Journal article, this one on October 28, 1999, a different perspective is presented. [*Figure 3*] The first sentence, however, recalled the same old theme. ***"Put aside for a moment all conventional wisdom about the poor economics and high risks of nuclear power."*** The article then attempts to describe

the present merger-buyout financial picture. Yet, I see other, more significant changes: the decommissioning gloom of 1997 is being replaced by the license renewal boom, and the compliance orientation has been replaced by safety-focused regulation. Most stranded costs are not stranded anymore. Moody's Investors Services changed its estimates of nuclear stranded costs from $130 billion 3 years ago to $10 billion presently. The amortization rate has been expedited, offering the prospect for many power plants' costs to approach production costs in the next few years. And, independent of financial considerations, the Nuclear Regulatory Commission has been changing its regulatory regime, improving predictability and accountability for all stakeholders.

I submit that two independent yet related variables -- safety and cost competitiveness -- determine the viability, indeed the survivability, of nuclear power and nuclear technologies. They are both integral quantities and embody most of the determinant issues. Safety and cost competitiveness [*Figure 4*] are both dynamic variables and easily tailored for use in decision-making. They have been, and could be, at odds with each other but should not be. In fact, it is imperative that they work together and not against each other. How this is done is an industry prerogative.

I suggest that in the United States of America, the marketplace and regulatory reform are coupling nuclear safety and cost in the right manner. Safety is the priority that enables cost competitiveness while safety-conscious cost considerations strengthen safety. This coupling is obvious when looking at averaged safety and cost performance indicators, and it is dramatic for "top performers." There is no doubt that the safest nuclear power plants in this country are generating electricity at very competitive production costs, lower than coal and approaching hydroelectric power. The United States Nuclear Regulatory Commission is no longer portrayed as being a dominant restraint on marketplace forces. The industry is now able to focus more sharply on real safety, licensing and regulatory requirements. It was the industry that first enabled the NRC's shift by lowering the number and significance of events and improving overall performance. The result is a regulatory agency that is far less event-driven and far more risk-informed, and an industry whose operating priority is safety.

[*Figure 5*] Safety and cost are also determinants of the credibility of the industry, a factor that cannot be overstated. Safety and cost should work in a synergistic relationship since for the industry -- still clouded in a mantle of adverse publicity and fear -- having credible benefits to society, including both safety and cost, is a must. And, for regulators, having credible processes to ensure adequate protection of public health and safety and the environment is fundamental.

[*Figure 6*] A reality check reveals that there can be no credible regulator without a credible industry, nor can there be a credible industry without a credible regulator.

Do these changes mean that a "level playing field" is around the corner? No. Nuclear power and other industrial users of radiation are not going to get a level playing field any time soon. Public perception of risk and of economics, influenced by the conflicting opinions presented by the media and by local, state, and federal entities, and the feedback effects among the influential parties, will keep the field uneven. Therefore, let me state the obvious: nuclear power and industrial users of radiation have to be better, much better. "Better" means keeping a very sharp focus on safety, on minimizing radiological events, and on meeting the safety-focused regulatory requirements. Effective and demanding self-regulation is a key element of being "better." I assume, for the industry, "better" also means to be economically competitive. The public will not settle for less.

Before I discuss the NRC's regulatory shift, I would like to show you a graph that depicts a cause and effect relation between safety and cost for some years back. [*Figure 7*] This slide shows the U.S. nuclear power plants O&M (no fuel) cost. Notice the effects of TMI and Chernobyl, the effects of costly lessons learned, some not so good lessons. Notice the industry recovery afterwards. Notice the 1996 "Millstone effect," when compliance equaled safety: notice the spike! A good portion of the cost increase was the result of a regulatory regime not safety-focused. And by the way, the American people paid. The present NRC initiative toward more effective, efficient, safety-focused regulation was started in the summer of '97 and fully launched in the summer of '98. In the interim, the cost of a lengthy regulatory shutdown became unacceptable in a cost competitive marketplace; Millstone and D.C. Cook probably are the last exceptions. Safety is a pre-requisite to economic performance.

Nuclear safety improvements that are necessary for adequate protection should be required without considering cost and the NRC will continue to do so. The NRC's ongoing regulatory changes are based on a more thorough and objective determination of what are real safety issues. I remember my first Millstone Commission Meeting when we were briefed on the 5,778-odd licensee issues to be resolved; of those, about 190 were "important" issues for NRC. I asked how many of them were safety issues. Silence ensued; neither NRC nor the licensee could then answer. Only afterwards was a real safety focus brought to the forefront of the Millstone recovery. Of course, Millstone had some serious administrative problems that clouded the entire issue and needed major cures. In retrospect, though few safety issues surfaced, safety had not permeated the Millstone organization, and they paid the price. Ultimately, I believe standards that are unnecessarily tight will have negative economic impacts, while standards that are not sufficiently exacting will have negative safety impacts -- eventually leading to negative economic impacts.

The Regulatory Shift

It is my strongly held belief that improving the quality of life of the American people is the foundation, the balance and the measure of success for our regulatory agency. Democracy and the marketplace are two key elements working for a level playing field. For me as a regulator, I see as an obligation and as an opportunity the use of our regulatory mandate to enhance the quality of life of the people we serve. The NRC must be forceful and credible in changing its regulations, carefully choosing its way in the rapid current (rio revuelto) so there is no question that safety is paramount; that, in fact, safety is improving. The NRC must be vigilant and rely on the strength of the safety fabric it is weaving.

[*Figure 8*] The NRC "change process" is calibrated by four objectives or, as we call them, outcomes:

- Maintain and/or improve safety
- Improve regulatory efficiency and effectiveness
- Reduce unnecessary burden
- Increase public confidence

The enabling factors for these outcomes are objectivity and due process, accountability and definition, working from a solid technical and legal basis. To accommodate the combined requirements of these objectives will be very difficult without the systematic use of the tool we call risk-informed regulation. In fact, there is no doubt that a significant driver of the regulatory shift has been the promise and the capability to risk-inform the regulatory framework so licensees can make risk-informed decisions.

Risk-informed regulation is a set of deterministic criteria, operating experience, defense-in-depth, engineering judgments and probabilistic risk assessments that qualitatively and quantitatively increases the knowledge base and is conducive to safety-focused decisionmaking. Risk-informed regulation is not a panacea. It will not replace what most people do now, but it is efficient and effective in focusing on safety. Operators will continue to operate, mechanics and electricians will continue to maintain, engineers will engineer, but there is going to be a new edge, a new dimension that provides decisionmaking capabilities. And if your implementation matches a risk-informed regulatory regime, then the NRC will have the information it needs, licensees will know what is required and what to do about it, and the public will be informed.

[*Figure 9*] Now I am going to quickly review areas where improvements have been made, emphasizing the philosophical shift in the regulatory framework. I should point out that this regulatory shift is being accomplished through open, participatory processes.

10 CFR 50.59

The fundamental process under which an operating reactor license holder can make changes to its facility was suffocated by the "zero factor." Any change or variation not clearly improving safety fell prey to the compliance mentality of 1996, i.e., zero increase in risk was the "law of the land." The term "Unreviewed Safety Question" reigned supreme: darned if you do and darned if you don't.

[*Figure 10*] Almost 2 years ago, I raised the "zero factor" issue in mathematical terms, zero = $10^{-\infty}$. Zero risk is not of this world, nor is infinity. I am pleased to tell you that the Commission eliminated the "zero factor," and that the Commission allowed for minimal changes that do not truly affect safety. A new, functional rule has been constructed and is ready to be implemented. I believe this rule increases safety by focusing resources on what is important to safety.

Maintenance Rule

10 CFR 50.65 was promulgated in 1991 as a risk-informed, performance-based rule. In practice, it was neither. The NRC and licensees were not prepared for such a rule, and in fact, the scope of the rule itself was and is contrary to the essential premise of risk-informed regulation and risk-informed maintenance. A truly risk-informed rule must be based on determining what are risk-significant structures, systems and components (SSCs) and on how to make the decisions affecting them accordingly. In 1999, while not changing the main scope of the rule, a new paragraph 50.65(a)(4) was finalized, permitting the configuration assessment to be limited to an optional scope determined by a risk-informed evaluation process. The reason I am mentioning this last point is because of the importance of the maintenance rule and the importance that the quality of PRAs will soon have in meeting the upcoming guidance. It is in the PRA quality where the basis for scope reduction will lie, where the additional confidence on safety-focused decisionmaking will be found and where the benefits of quantitative determinations will be based. Low safety significance SSCs or combinations thereof will be accounted for in a state-of-the-art PRA and will, therefore, not be challenged in regulatory space. I have one recommendation to make to the industry: if you have not done it yet, complete a functional, quality PRA and train your people in how to use it. One small regulatory or preventable shutdown will pay for this type of PRA many times over.

Reactor Oversight

In addition to establishing the body of safety regulations for this industry, the NRC needs an effective method to verify that the regulations are fulfilling their functions. The fundamental way we do this is through our oversight process, which, in effect, buttresses adequate protection.

To be consistent with the performance improvements at nuclear power plants, the NRC is changing its oversight process to a more risk-informed method of assessing plant performance. These changes will result in streamlined inspection, assessment, and enforcement by focusing inspections on activities where the potential risks are greater and by using objective measurements of plant performance whenever possible. They should also add clarity and predictability to NRC performance assessments, as well as permit more efficient use of NRC resources.

The existing assessment processes were analyzed in relation to their impact on the NRC's mission to provide adequate protection of public health and safety. The NRC task force on inspection and assessment, working closely with the agency's public and industrial stakeholders, identified seven "cornerstones" that provide the foundation for safe performance at nuclear power plants. Information to support assessments of licensee performance in each of the cornerstones will be derived from plant performance indicators and NRC inspections. The baseline inspection program will review areas that are not covered by performance indicators, will verify the accuracy of the licensee's performance indicators, and will also provide a comprehensive review of the licensee's ability to find and correct problems. The major objective of these assessments is to determine where the NRC should focus its attention and resources.

Two points of fact

— While the performance indicators attract all the attention, the beef is in the data gathering and processing. There lies the strength of the assessment process because it has to be open, it is periodic, and it will be assessed by multiple stakeholders. The fact that deficiencies will go to the Corrective Action Program and be tracked, without pity, provides the backbone for this major shift.

— This process is new, will be in force in April 2000 and is unforgiving. The Commission has committed to take action, whenever warranted, and we will.

Enforcement

In parallel with the changes to the inspection and assessment process, the NRC's enforcement policy continues to be revised to respond to violations in a safety-focused, predictable, and consistent manner. Enforcement has been properly restructured to be an outcome, when warranted, of the inspection and assessment processes. This is in contrast to the previous situation, in which enforcement very often drove inspection and assessment, regardless of the safety significance of the issues being addressed. Indeed, as I stated in one of my votes on this process in the Spring of 1998, "informed enforcement is one of several regulatory tools, not a driving force of assessment activities."

Risk-Informing Part 50

In November of 1997, I stated that the often patched regulatory fabric of the NRC was no longer "patchable." I proposed that the entire Part 50 be made risk-informed. Skeptical at first, both the NRC and the industry have become convinced. The work on risk informing Part 50 is accelerating and holds rich promise for a more efficient and safer way to regulate.

I should point out that there is one field of nuclear endeavor that is ripe for work: regulatory technology. [*Figure 11*] I define regulatory technology as the science and practices that combine scientific, engineering and technological knowledge with regulatory requirements, as well as socio-political constraints, to effect changes in regulation and technology for the benefit of society. Truly a challenging global technology and one that should not be the exclusive domain of the regulator.

In summary, the United States Nuclear Regulatory Commission (NRC) has recognized that the use of risk insights can be a catalyst for reconciling the beneficial and radiological protection aspects of the peaceful application of nuclear technology. It is safety-focused and a valuable decisionmaking tool. The NRC has been employing risk information in developing rules and policies for regulating nuclear power plants, and is continuing to widen the application of risk-informed techniques. Building on this experience, the NRC is also expanding the use of risk information in its regulation of nuclear materials, including medical uses of radioisotopes, high level waste, licensing of domestic uses of special nuclear material, clearance of radioactively contaminated materials, and the regulatory controls for generally licensed devices. There are regulations in various stages of development to risk inform these areas. I urge those of you from abroad to consider these initiatives and how they could fit the national interests of your countries.

And speaking about global opportunities and challenges, there is a pervasive opinion that safety and economic deficiencies in a few nuclear programs could force the demise of all other nuclear programs. This truism is known all over the world as "a nuclear accident anywhere is a nuclear accident everywhere." This is true but is not a complete picture. I believe that the world-wide quality of nuclear technology is, and will be, its greatest asset. There is strength in quality and in numbers, and this is particularly true for nuclear power. [*Figure 12*] So I leave you with these thoughts:

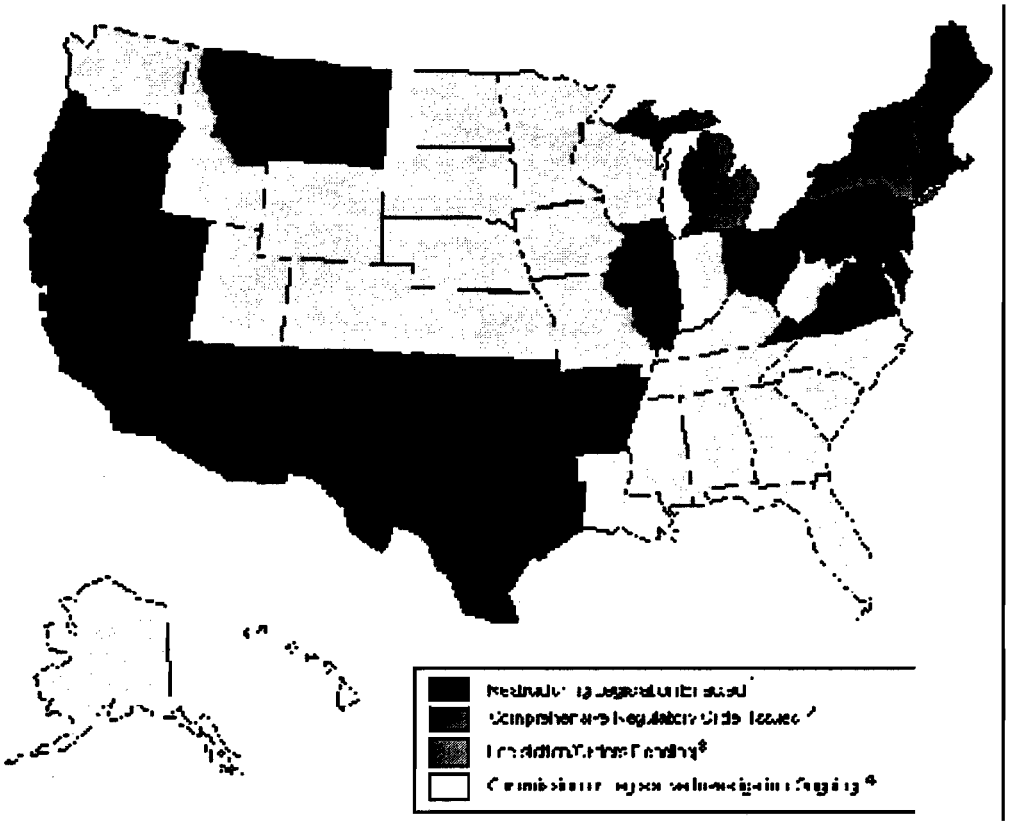
> "a nuclear safety improvement anywhere is a nuclear safety improvement everywhere"
>
> "a nuclear regulatory improvement anywhere is a nuclear regulatory improvement everywhere"

En rio revuelto, ganancia de pescadores.

[*Figure 13*] A recent USA Today article discussed how a 20th Century visionary, President Ronald Reagan, changed the way people thought about business, how he "celebrated the vitality and magic of the marketplace" and how he urged people in government to think like entrepreneurs, "seeing possibilities where others see only problems." I urge you to think of the possibilities to benefit from safety-focused regulations and from risk-informed decisionmaking as they interact with the marketplace.

It has been my privilege to address you this morning and I wish you well.

Status of State Electric Industry Restructuring Activity as of November 1, 1999



1. Arizona, Arkansas, California, Connecticut, Delaware, Illinois, Maine, Maryland, Massachusetts, Montana, Nevada, New Hampshire, New Jersey, New Mexico, Ohio, Oklahoma, Oregon, Pennsylvania, Rhode Island, Texas, and Virginia.
2. Michigan, New York, and Vermont.
3. None
4. Alabama, Alaska, Colorado, District of Columbia, Florida, Georgia, Hawaii, Idaho, Indiana, Iowa, Kansas, Kentucky, Louisiana, Minnesota, Mississippi, Missouri, Nebraska, North Carolina, North Dakota, South Carolina, South Dakota, Tennessee, Utah, Washington, West Virginia, Wisconsin, and Wyoming.

Source: Energy Information Administration

Figure 1

Rio revuelto,

ganancia de pescadores

(Translation: Turbulent river; fishermen profit)

Figure 2

"Put aside for a moment

all conventional wisdom

about the poor economics

and high risks of

nuclear power."

Wall Street Journal
October 28, 1999

Figure 3

Safety and Cost: Friends or Foes

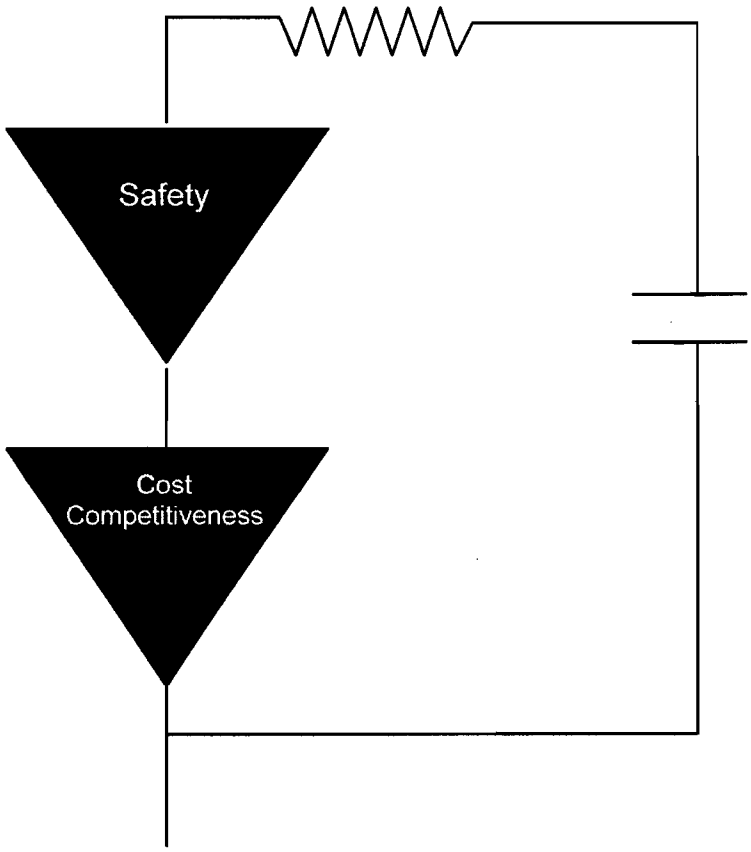


Figure 4

Credibility and Acceptability

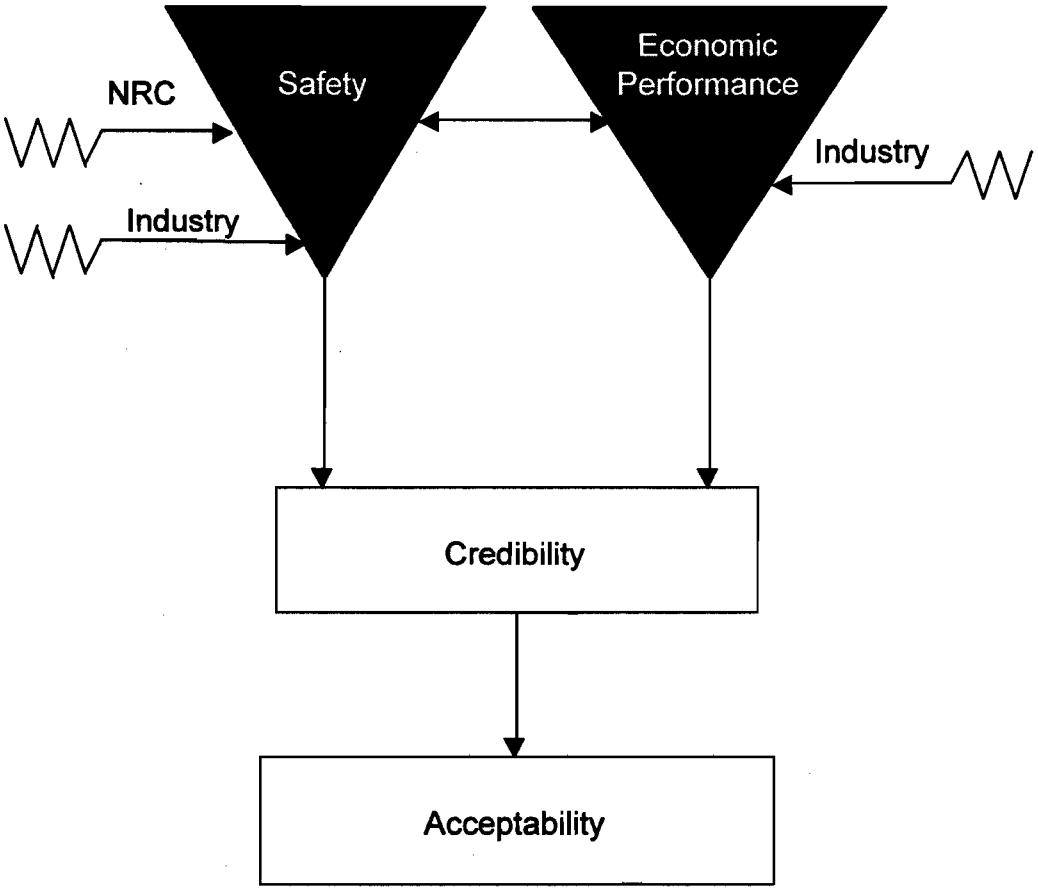


Figure 5

"There can be no credible regulator

without a credible industry,

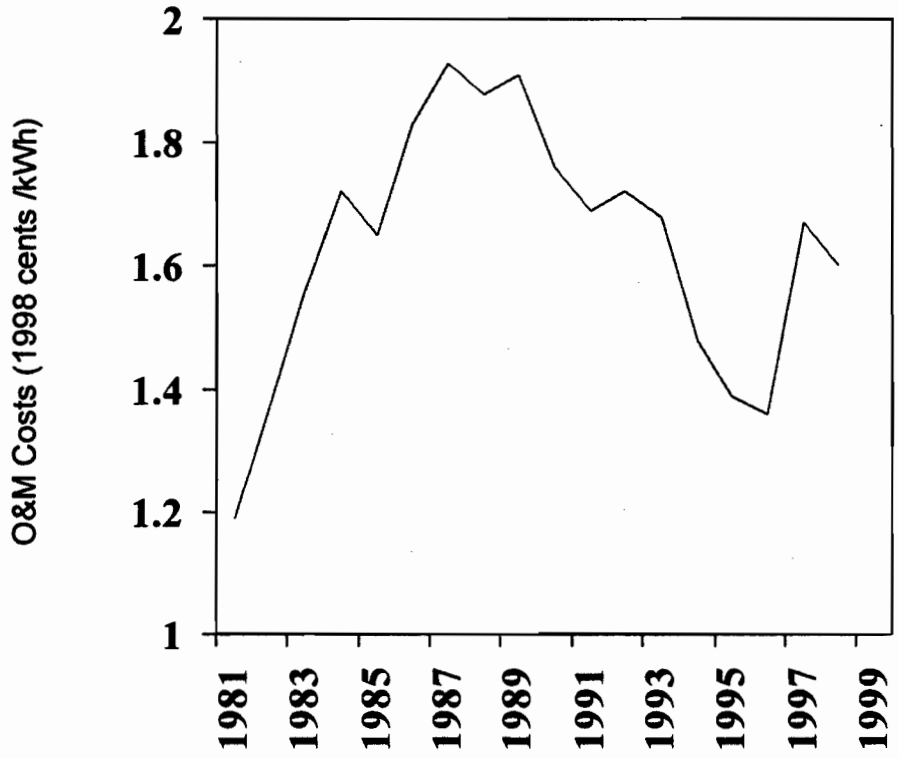
nor

can there be a credible industry

without a credible regulator."

Figure 6

U.S. Nuclear Plant O&M Costs



Source: Utility Data Institute

Figure 7

Regulatory Objectives / Outcomes

- Maintain / Improve Safety
- Improve Regulatory Efficiency and Effectiveness
- Reduce Unnecessary Burden
- Increase Public Confidence

Enablers

Technical and Legal Basis

- Objectivity and Due Process
- Accountability
- Definition

Figure 8

Current Regulatory Initiatives:

- 50.59
- Maintenance Rule
- Reactor Oversight
- Enforcement
- Risk-Informing Part 50

Figure 9

$$\text{Zero} = 10^{-\infty}$$

Zero risk is not of this world.

Figure 10

Regulatory Technology

The science and practice that

combine scientific, engineering,

and technological knowledge

with regulatory requirements,

as well as socio-political constraints.

Figure 11

"A nuclear safety improvement anywhere
is
a nuclear safety improvement everywhere."

"A nuclear regulatory improvement anywhere
is
a nuclear regulatory improvement everywhere."

Figure 12

President Ronald Reagan, 1985

"see(ing) possibilities

where others

see only problems"

Figure 13

Nuclear Regulatory Commission

Office of Public Affairs -- Region I

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Fax: 610/337-5241

Diane Screnci (Phone: 610/337-5330) (e-mail: dps@nrc.gov)

Neil Sheehan (Phone: 610/337-5331) (e-mail: nas@nrc.gov)

I-00-02

January 7, 2000

NRC Completes Agency Action on Letters of Reprimand Issued to Millstone Managers in Employee Discrimination Case

The Nuclear Regulatory Commission has completed its administrative review of a notice of violation and four letters of reprimand issued in April to five Northeast Nuclear Energy Company managers for discrimination against three employees who raised safety concerns in the 1993-to-1995 time frame at the Millstone nuclear power plant in Waterford, Conn.

As a result of that review, the notice of violation and three letters of reprimand remain as issued; one letter of reprimand has been withdrawn.

The NRC took this action after reviewing responses of the five managers. That review determined that one of the individuals provided new and significant information not previously available which indicated he was not responsible for the discriminatory action. The letters and the names of all four recipients of the letters of reprimand have been withheld from public disclosure while the review was under way.

Copies of the three letters that were issued, the notice of violation, and the responses to them are available from the NRC Public Document Room and the Office of Public Affairs.

EDITORS: NRC issued a press release April 6 about the initial action in this case.

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

United States Nuclear Regulatory Commission

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No. 99-261

December 9, 1999

Thermal Science Agrees to $300,000 Settlement With Nuclear Regulatory Commission

The Nuclear Regulatory Commission today announced that Thermal Science, Inc. (TSI) of St. Louis has agreed to pay a $300,000 settlement of an NRC enforcement action regarding the company's Thermo-Lag fire barrier products.

The settlement follows a series of measures taken by the NRC since it began considering enforcement action against TSI regarding the testing of the company's Thermo-Lag fire barrier.

NRC investigators found inconsistencies in the testing of Thermo-Lag products in test reports and other documents submitted by TSI.

Results of the NRC investigation were forwarded to the U.S. Department of Justice, which presented the matter to a grand jury that indicted TSI and its president, Mr. Rubin Feldman. But both the firm and its president were acquitted following a criminal trial that ended in August 1995.

In 1996, the NRC proceeded with a civil action, a proposed civil monetary penalty, alleging several violations of its regulations by TSI. TSI challenged the NRC's action in U.S. District Court on the grounds that (1) it constituted double jeopardy, and (2) NRC lacked authority to fine non-licensees. The District Court dismissed TSI's suit last year. TSI appealed, but the U.S. Court of Appeals denied a motion for a stay of NRC's administrative process.

TSI subsequently filed an answer to the NRC's enforcement action, denying any violations. Although the NRC believed TSI's arguments to be without merit, the parties agreed to settle the matter for $300,000 in order to bring this longstanding issue to closure without further litigation.

A copy of the settlement is posted on the Internet at this address: http://www.nrc.gov/OE/rpr/ea95009.htm
.# # # # #

Nuclear Regulatory Commission

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801 Warrenville Road, Lisle IL 60532

Jan Strasma (Phone: 630/829-9663)

Angela Greenman (Phone: 630/829-9662)

E-mail: opa3@nrc.gov

RIII-99-050

December 21, 1999

FOR IMMEDIATE RELEASE

NRC Proposes $88,000 Fine Against U.S. Enrichment Corp. For Employment Discrimination Violation at Paducah Plant

The Nuclear Regulatory Commission staff has proposed an $88,000 fine against U. S. Enrichment Corporation of Bethesda, Maryland, for discriminating against a manager who raised safety issues at the Paducah Gaseous Diffusion Plant in Paducah, Kentucky.

NRC regulations prohibit employers at nuclear facilities from discriminating against employees who raise safety issues.

The NRC Office of Investigations determined that the Manager of Quality Systems at the Paducah plant was transferred in August 1998 to a non-managerial position in the Training Department after he expressed concerns about the quality assurance program.

He told his supervisor that the plant was not fulfilling all the requirements of an industry standard for quality assurance programs. He also stated his concern that the plant's Quality Assurance Program would be adversely affected by the Quality Systems staff having to perform other activities at the plant.

The investigation was completed in March of this year, and the NRC staff met with company officials on June 30 for a predecisional enforcement conference to review the case.

The company contended that the manager was transferred because of performance considerations. Based on the investigation findings and subsequent information, however, the NRC staff determined that the decision to transfer the manager was based, in part, on his raising of concerns about the Quality Assurance Program.

U. S. Enrichment Corporation informed the NRC in October that it was taking extensive corrective actions to help its managers address the safety concerns of its employees and to encourage a "nuclear safety conscious" work environment.

The company has until January 19 to pay the fine or to challenge it. If U.S. Enrichment Corporation challenges the fine, and it is subsequently imposed by the NRC staff, the company may request a hearing.

# # # # #

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No. 99-270

December 23, 1999

New NRC Regulation to Permit Nuclear Power Plants to Change Accident Analyses of Public Radiation Dose

The Nuclear Regulatory Commission has amended its regulations to permit nuclear power plant licensees to take advantage of updated research findings on estimated public radiation doses from reactor accidents.

The new rule will permit these licensees to use what is known as an alternative "source term" for the accident analysis on which plant design and operations are based, replacing a source term that has been in effect for the past 37 years. Experience from the 1979 Three Mile Island accident and research that followed it have made this change possible.

"Source term" is the technical name for the calculation of the rate, magnitude and chemical form in which the radioactive material produced by the atom-splitting process in a nuclear reactor would be released from the reactor to the containment if an accident occurred.

Nuclear power plants use the source term for analyzing possible accident consequences -- including potential radiation dose to the public from leakage out of the containment into the environment -- and factor that analysis into plant design and operation.

All currently operating nuclear power plants were licensed on the basis of a source term published in 1962 by the Atomic Energy Commission, NRC's predecessor agency. That procedure assumed an immediate release of radioactive materials to the containment during a severe accident, including a substantial amount of radioactive iodine.

But what occurred in the Three Mile Island accident, in addition to extensive research which followed it, suggests that a release into the containment would be phased, rather than immediate, and that radioactive iodine would be predominantly in the form of cesium iodide, an aerosol that is more amenable to mitigation mechanisms. Revised source terms published by NRC in 1995 reflected those findings.

The rule now being adopted will permit utilities with nuclear power plant operating licenses to replace the 1962-era source term in their licenses with a revised one. NRC believes this change can reduce an unnecessary burden on many licensees without compromising public health and safety, reduce worker radiation exposure, and improve overall safety. This regulation, however, is not intended to provide licensees with relief from NRC's emergency planning requirements.

Specifically, it is expected that such a change could cut down on occupational radiation exposures in activities such as the frequency of installation of charcoal filters, maintenance of certain containment isolation valves, and repairs to systems to maintain leak-rate limits that are overly restrictive in the light

UNITED STATES
NUCLEAR REGULATORY COMMISSION

Announcement No. 103

Date: December 15, 1999

To: **All NRC Employees**

SUBJECT: IN MEMORIAM: NUNZIO J. PALLADINO, 1916-1999

On Sunday evening, December 12, 1999, former Nuclear Regulatory Commission Chairman Nunzio J. Palladino passed away at the age of 83 after a long struggle with Parkinson's disease. At the time of his death, Dr. Palladino was being treated at Centre Community Hospital in State College, Pennsylvania.

Dr. Palladino had a long and distinguished career in public service, in academia, and in the nuclear industry. Born on November 10, 1916, in Allentown, Pennsylvania, he earned his Bachelor and Masters degrees in Mechanical Engineering at Lehigh University in 1938 and 1939. Subsequently, he performed graduate work at the University of Tennessee and the University of Pittsburgh. From 1939 to 1959, he worked for the Westinghouse Electric Corporation. During this period, he served four years as an engineer on loan to the Oak Ridge and Argonne National Laboratories and led the Westinghouse team that designed the reactor cores for the submarine Nautilus and the first full-scale nuclear electric generating plant at Shippingport, Pennsylvania. In July 1959, he came to The Pennsylvania State University at the request of the then Dean of the College of Engineering, who wanted him to start a nuclear engineering department at Penn State. Dr. Palladino served as Professor of Nuclear Engineering and was appointed Dean of the College of Engineering in 1966. At about the same time, Dr. Palladino was selected by the Atomic Energy Commission as a member of its Advisory Committee on Reactor Safeguards. He served as a Committee member from 1964 to 1974, and as ACRS Chairman in 1967. Dr. Palladino was also active in the public service in his home state, serving on the Governor's Energy Council and the Science Advisory Committee, the Pennsylvania Advisory Committee on Atomic Energy Development and Radiation Control, and as member of the Governor's Commission on Three Mile Island. He also participated on a Nuclear Regulatory Commission Special Task Force on the TMI cleanup. In July 1981, President Reagan named him the NRC Chairman, a position he held from 1981-1986.

With Dr. Palladino's passing, the NRC has lost one of its most distinguished elder statesmen, and the Nation has lost one of its true nuclear pioneers. To those who knew him personally, the loss is far greater. By all accounts and in spite of his many accomplishments, Dr. Palladino remained a warm, soft-spoken man who exhibited the unusual combination of great learning, common sense, and respect for those with whom he worked. At the NRC's 25th Anniversary observance next month, one place of honor will now be vacant, and the agency as a whole will miss his counsel and wisdom long after the celebration has faded in memory.

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No. 99-263

Wednesday, December 15, 1999

FOR IMMEDIATE RELEASE

NRC Certifies Westinghouse Electric Company's AP600 Reactor Design

The Nuclear Regulatory Commission has amended its regulations to certify the AP600 standard plant design developed by Westinghouse Electric Company. The certification is the third issued under the agency's new licensing process for standard design certification and is valid for 15 years.

The AP600 design is for a nuclear power plant that is capable of producing 600 megawatts of electricity. The plant, which can be assembled from modular components, features enhanced safety systems that rely on gravity and pressure differentials to safely shut down the reactor or mitigate the effects of an accident. It is designed for a 60-year operating life.

With the certification, a utility that wishes to build and operate a new nuclear power plant can choose to use the design and reference it in a license application. Safety issues within the scope of the certified design are not subject to litigation, although site-specific environmental impacts associated with building and operating the plant at a particular location are.

Applicants for a license can make plant-specific changes to portions of the AP600 standard design by following the procedures set out in the design certification rule. The applicant is required to maintain records of all such changes.

In the fall of 1998, the agency issued a final design approval for the Westinghouse AP600 plant, completing the staff's technical review of the application for design certification received in 1992. This step permitted the staff to begin the administrative, or rulemaking, phase.

No application for a license using the AP600 standard design has been filed with the agency.

This rule will become effective 30 days after publication in the Federal Register.

The public had been invited to submit comments on the proposed design certification rule, the AP600 design control document submitted by Westinghouse that was incorporated into the agency rule, and the environmental assessment of the AP600 design. Interested parties also had the opportunity to request an informal hearing. No requests for hearings were received and the submitted comment was of a general nature.

For additional information on the design's certification, contact Jerry N. Wilson, at 301-415-3145, or e-mail: jnw@nrc.gov .

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United States Nuclear Regulatory Commission

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No. 00-013

Friday, January 14, 2000

NRC Staff Schedules Three-Day Conference to Discuss Regulatory Issues With Nuclear Industry

Advanced registration is now available for the Nuclear Regulatory Commission's 12th Annual Regulatory Information Conference, March 27, 28 and 29, at the Capital Hilton Hotel, 16th and K Streets, NW, Washington, D.C.

The conference will focus on major issues and initiatives associated with the regulation of commercial nuclear power plants. The NRC's revised oversight process, risk-informing regulations, license renewal and decommissioning are among the topics that conferees will discuss during a variety of issue-specific sessions. For additional information and to register for the conference, interested parties should access the NRC's conference Web site at: http://www.nrc.gov/NRC/REACTOR/RIC/2000/index.html , or contact IQ Solutions, Inc. 11300 Rockville Pike, Suite 801, Rockville, Maryland 20852, (phone) 301-984-1471, (fax) 301-984-1333.

The conference is open to the public, but advanced registration is suggested. Hotel reservations should be made through the Capital Hilton at 202-393-1000, or 1-800-HILTONS. After William D. Travers, Executive Director for Operations, opens the conference on Monday, March 27, at 1 p.m., Chairman Richard A. Meserve will deliver the keynote speech. Samuel J. Collins, Director, Office of Nuclear Regulatory Regulation, and Ralph Beedle, Senior Vice President and Chief Nuclear Officer of the Nuclear Energy Institute (NEI), will then co-chair a plenary session on conference objectives and expectations. Regional breakout sessions will be held at 4:30 p.m.

On Tuesday, March 28, the conference will begin at 8 a.m. with a plenary session on regulatory trends and eight issue-specific breakout sessions, including those on the revised oversight process, decommissioning, technical specifications and licensing improvements, effective/efficient regulation, waste issues and public confidence. Commissioner Greta Joy Dicus will address a plenary session at noon followed by lunch, sponsored by the NEI, which will feature Joe Colvin, the organization's President and Chief Executive Officer, as the guest speaker. Luncheon tickets can be obtained at a cost of $35 per person by calling Nicki Rocco at 202-739-8048, or via e-mail at lec@nei.org .

The Tuesday post-luncheon schedule will include Commissioner Edward J. McGaffigan, Jr. addressing the conferees in a plenary session at 2:15 p.m. Afternoon breakout sessions will follow on risk-informing regulations, revised oversight processes, information access/sharing and international regulatory experience. Commissioner Nils J. Diaz then will speak to the attendees at 4:45 p.m.

The Wednesday session will start at 8 a.m. with a plenary session featuring Commissioner Jeffrey S. Merrifield. Four breakout sessions will follow on license renewal, work-environment issues, industry initiatives and ongoing rulemakings. A feedback plenary session will be held at 11a.m., with adjournment at noon.

ACRS PRESENTATION

REVISED REACTOR OVERSIGHT PROCESS PILOT PROGRAM RESULTS AND LESSONS LEARNED

WILLIAM DEAN
ALAN MADISON
MICHAEL JOHNSON
GARETH PARRY

FEBRUARY 3, 2000

AGENDA

- INTRODUCTION
- PILOT PROGRAM RESULTS - READINESS FOR START OF IMPLEMENTATION
- DEFINING PRINCIPLES AND ASSUMPTIONS
- PERFORMANCE INDICATORS
- SIGNIFICANCE DETERMINATION PROCESS
- ASSESSMENT PROCESS

PILOT PROGRAM RESULTS - READINESS FOR IMPLEMENTATION

- PIs and Baseline Inspection provide a sound framework for providing oversight of licensee performance to assure that reactor safety is maintained

- NRC assessments and actions more objective, understandable, and predictable to industry and public

No reduction in the Inspection Force, stability + predictable program!

- Focus on risk significant issues has reduced unnecessary regulatory burden

- Revised oversight process adequate to support initial implementation at all plants

Made improvement, will continue to remain - still skeptical internal & external!

- The staff will implement an ongoing self-assessment process

next phase implementation initial

4 outcome measures should be used!
Thresholds should be consistent with objectives!

FRAMEWORK DEFINING PRINCIPLES AND ASSUMPTIONS

- Thresholds can be set, beyond which only minimal NRC interaction is warranted.

Thresholds need to be plant specific!

  - Revised Oversight Process
    - Defined, objective threshold
    - PI "Green" - acceptable with the inspection program!
    - SDP "Green"

Licensee Response Broad!

  - Current Process
    - Subjective threshold
    - Minor violation

- Adequate assurance of performance needs both PIs and inspection results.
  - Revised Oversight Process
    - Integrates PIs with inspection findings
    - Continual assessment
  - Current Oversight Process
    - Relies on inspection findings
    - PIs have minor role and used broadly
    - Assessments every 18-24 months

- Performance in crosscutting areas will be inspected or inferred through both PIs and inspection findings.

  - Revised Oversight Process
    - Assesses performance <u>in</u> cornerstones
    - Considers cross cutting issues causes of problems in cornerstones
    - Directly inspects PI&R, certain aspects of human performance, reviews SCWE
    - Recent changes

  - Current Oversight Process
    - Assesses performance <u>of</u> functional areas
    - Looks for issues crossing functional areas
    - Addresses PI&R in SALP letter

- The oversight process will be indicative within the licensee response band.

  - Revised Oversight Process
    - Risk-informed Baseline Inspection Program (indicative)
    - Supplemental (diagnostic)
    - Increased oversight based on Action Matrix

  - Current Oversight Process
    - Core Inspections and regional initiative inspections diagnostic
    - Initiative inspections loosely based on SALP score

PERFORMANCE INDICATORS

THRESHOLDS

- Used to identify performance levels below which increased NRC interaction is warranted; no ranking or trending of performance

- Green-white threshold identifies outliers/nominal performance
  - --- Based on data from 1995 to 1997
  - --- Identified about 5% of plants per year
  - --- Will be reevaluated using historical data

- IE and MS Yellow and red thresholds based on increase in risk
  - --- Yellow corresponds to ΔCDF of about 10^{-5}
  - --- Red corresponds to ΔCDF of about 10^{-4}

PERFORMANCE INDICATORS

SET OF PIS

- Based upon framework: cornerstones and attributes of licensee performance
- Selected from those currently in use or readily available
  --- Minor modifications to simplify, clarify, or customize
- Improvements made continuously
- Benchmarking showed indicators identified poor performers
- Benchmarking showed SSAs provided no new information

PERFORMANCE INDICATORS

ONGOING WORK

- Consistency of PI Definitions
- Guidance on Programmatic Issues
- Definitions and Guidance for Some Indicators
- Impact of Multi-Unit Sites or Indicators on Site-wide Indicators
- Continued Review of Indicators In Self-Assessment Program
- Risk-Based Indicators/Industry-Wide Performance

SDP Principal Objectives

Significance Characterization

- To characterize the significance of inspection findings arising from deficient licensee performance, using risk metrics where appropriate

Communication

- To clearly communicate the staff's bases for its characterization of the significance of deficient licensee performance

SDP Development/Refinement

Plant Specific Reactor Safety SDP

- Plant-specific worksheets are developed from information directly available to the staff (e.g., IPEs)
- Site visits to be conducted with each licensee to obtain comments and any recommended worksheet changes
- Each reactor safety SDP should be tested against the licensee's PRA for general consistency of results

All SDPs

- A feasibility study using actual issues is performed on all SDPs prior to initial implementation

SDP Ongoing Work

- Site-visits and consistency testing for reactor safety SDP are expected to continue through April 2000
- Containment SDP expected to be developed and ready in April 2000
- Shutdown issues screening tool expected to be developed and ready in April 2000
- External events screening tool development in progress with target date April 2000

ASSESSMENT PROCESS

- Provides improved objectivity (subjective judgement is not a central aspect)

- Provides increased predictability through the use of established thresholds for performance and an "Action Matrix" that identifies planned regulatory response

  - Predictability versus rigidity
  - Process for addressing deviations

- Provides opportunity for licensee response/input prior to final NRC determination of issue significance and regulatory response ("due process")

Internal Survey

- Background & Purpose

Purpose: Solicit first-hand insights from pilot plant participants

-End-of-pilot survey sent to regions (11/99)
-Responses from 94 individuals who directly participated in pilot
-*Inside NRC* released information from survey (1/00)

- Results

-Regional administrators (Views: significant improvement, more objective, improved consistency)
Concerns: documentation threshold, inspection of cross-cutting issues, and SDP

-Individual participants (Views: more objective, PIs in appropriate areas, and effective training)
Concerns: timely identification of declining performance, documentation threshold, and SDP

- Actions

-Considered during internal and external lesson learned workshop

-Factored into actions planned for completion prior to and post initial implementation

-Results to be released to internal and external stakeholders

(3)

December 21, 1999

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

SUBJECT: INSPECTION PLAN FOR SEQUOYAH

On December 6, 1999, the NRC staff reviewed the performance of the Sequoyah Nuclear Plant as reflected in the performance indicators and inspection results in order to integrate performance information and to plan for inspection activities at your facility from January 3, 2000, through July 31, 2000. The purpose of this letter is to inform you of our plans for future inspections at your facility.

We have not identified any areas in which you crossed a performance threshold. Therefore we plan to conduct only baseline inspections at your facility over the next seven months. However, the significance determination of the turbine building railroad bay flooding event is still under review and may involve further inspection.

Enclosure 1 details the scheduled inspections that will occur from January 3, 2000, through July 31, 2000. The inspection plan is provided to minimize the resource impact on your staff and to allow for scheduling conflicts and personnel availability to be resolved in advance of inspector arrival onsite. Routine resident inspections are not listed due to their ongoing and continuous nature. The last four months of the inspection plan are tentative and will be revised at the end-of-cycle review meeting.

Enclosure 2 contains a historical listing of plant issues, referred to as the Plant Issues Matrix (PIM), that were identified during the pilot plant inspection program period. The PIM includes items summarized from inspection reports or other docketed correspondence between the NRC and Tennessee Valley Authority.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

If circumstances arise which cause us to change this inspection plan, we will contact you to discuss the change as soon as possible. Please contact Paul Fredrickson at (404) 562-4530 with any questions you may have regarding this letter or the inspection plan.

Sincerely,

(Original signed by Paul E. Fredrickson)

Paul E. Fredrickson, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket Nos. 50-327, 50-328
License Nos. DPR-77, DPR-79

Enclosures:
1. Sequoyah Inspection/Activity Plan
2. Plant Issue Matrix

ccw/encls:
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(4)

December 22, 1999

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Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: INSPECTION PLAN - QUAD CITIES NUCLEAR POWER STATION

Dear Mr. Kingsley:

On December 1, 1999, the NRC staff reviewed the performance of Quad Cities Nuclear Power Station as reflected in the performance indications and inspection results in order to integrate performance information and to plan for inspection activities at your facility from December 1, 1999, to July 31, 2000. The purpose of this letter is to inform you of our plans for future inspections at your facility so that you will have an opportunity to prepare for these inspections and to inform us of any planned inspections which may conflict with your plant activities.

Based on our review of performance at Quad Cities, we identified that the threshold from Green (licensee response band) to White (increased regulatory response band) was crossed for the Heat Removal System Unavailability performance indicator in the third quarter of 1999. We will review this as a supplemental inspection (IP 95001) by the resident inspector staff in accordance with the action matrix of the new assessment process. Another threshold was crossed from Green to White and back to Green in the plant protection area. This was identified by aggressive action on your part to review and report additional performance indicator information for Perimeter Alarm Security Equipment Performance. We have already completed a supplemental inspection to assess this performance indicator color change.

The NRC staff also has had numerous discussions with ComEd personnel about interpretations and information provided for other performance indicators. We understand that your staff has put significant effort into correctly reporting performance indicator data. Based on your efforts, evolving interpretations and definitions for some performance indicators, and the continuing discussions between us, we plan to perform additional baseline inspection using the Performance Indicator Verification procedure during the next 8 months.

Additionally, the staff has identified a potential adverse trend in the cross-cutting area of Problem Identification and Resolution. Our findings in this area include motor operated valves with test results outside of the acceptance criteria, contaminated condensate storage tanks with fewer operable heaters than designed, and repetitive failures of a high pressure coolant injection valve. While the adverse trend has not yet resulted in performance indicators or inspection findings outside of the licensee response band, our inspection plan includes a

Problem Identification and Resolution inspection in June of 2000. This is early in the next assessment period which runs from April 1, 2000, to March 31, 2001.

Enclosure 1 details the scheduled inspections that will occur from December 1, 1999, to July 31, 2000. Also, we will continue to conduct the resident inspector baseline procedures; although the resident inspections are not listed due to their ongoing and continuous nature. The last 4 months of the inspection plan are tentative and may be revised based on the results of our end-of-cycle review meeting. Enclosure 2 is the Plant Issue Matrix that was used as part of the mid-cycle review. Note that Enclosure 2 contains entries for Inspection Report 1999010 which were considered in our review, but were not part of the Pilot Plant Inspection Program.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR).

If circumstances arise which cause us to change this inspection plan, we will contact you to discuss the change as soon as possible. Please contact me at 630/829-9703 with any questions you may have regarding this letter or the inspection plan.

Sincerely,

/s/ M. A. Ring

Mark A. Ring, Chief
Reactor Projects Branch 1
Division of Reactor Projects

Docket Nos. 50-254; 50-265
License Nos. DPR-29; DPR-30

Enclosures: 1. Inspection Plan
2. Plant Issue Matrix

SEE PREVIOUS CONCURRENCES

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DCD - Licensing
J. Dimmette, Jr., Site Vice President
G. Barnes, Quad Cities Station Manager
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GREENS
Chief, NRR/DIPM/IIPB

(5)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

December 27, 1999

J. H. Swailes, Vice President of
Nuclear Energy
Nebraska Public Power District
P.O. Box 98
Brownville, Nebraska 68321

SUBJECT: INSPECTION PLAN - COOPER NUCLEAR STATION

Dear Mr. Swailes:

On December 7, 1999, the NRC staff reviewed the performance of Cooper Nuclear Station as reflected in the performance indicators and inspection results in order to integrate performance information and to plan for inspection activities at your facility from December 1, 1999, through July 31, 2000. The purpose of this letter is to inform you of our plans for future inspections at your facility so that you will have an opportunity to prepare for these inspections and to inform us of any planned inspections which may conflict with your plant activities.

We have not identified any areas in which you crossed a performance threshold. Therefore, we plan to conduct only baseline inspections at your facility over the next 8 months.

This letter advises you of our planned inspection effort resulting from the Cooper midcycle review. Enclosure 1 details the scheduled inspections that will occur from December 1, 1999, through July 31, 2000. The inspection plan is provided to minimize the resource impact on your staff and to allow for scheduling conflicts and personnel availability to be resolved in advance of inspector arrival onsite. Routine resident inspections are not listed due to their ongoing and continuous nature. The last 4 months of the inspection plan are tentative and will be revised at the end-of-cycle review meeting.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR).

If circumstances arise which cause us to change this inspection plan, we will contact you to discuss the change as soon as possible. Please contact me at 817/860-8185 with any questions you may have regarding this letter or the inspection plan.

Sincerely,

/RA/

Charles S. Marschall, Chief
Project Branch C
Division of Reactor Projects

Docket No.: 50-298
License No.: DPR-46

Enclosures:
1. Cooper Nuclear Station Inspection/Activity Plan
2. Plant Issues Matrix

cc w/enclosures:
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Vick L. Cooper, Chief
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Kansas Department of Health
and Environment
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Forbes Field Building 283
Topeka, Kansas 66620

E-Mail all documents to Jim Isom for Pilot Plant Program (JAI)
E-Mail all documents to Sampath Malur for Pilot Plant Program (SKM)

E-Mail notification of issuance of all documents to Nancy Holbrook (NBH).

bcc to DCD (IE01)

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(6)

January 3, 2000

Mr. Michael J. Colomb
Site Executive Officer
New York Power Authority
James A. FitzPatrick Nuclear Power Plant
Post Office Box 41
Lycoming, NY 13093

SUBJECT: MID-CYCLE PERFORMANCE REVIEW AND INSPECTION PLAN - JAMES A. FITZPATRICK NUCLEAR POWER PLANT

Dear Mr. Colomb:

On December 13, 1999, the NRC staff reviewed the plant performance of James A. FitzPatrick Nuclear Power Plant during June 1 - November 30, 1999, as reflected in the performance indicators and inspection results, in order to integrate performance information and to plan for inspection activities at your facility through July 31, 2000. The purpose of this letter is to inform you of our plans for future inspections at your facility so that you will have an opportunity to prepare for these inspections and to inform us of any planned inspections which may conflict with your plant activities.

Our review of performance at the James A. FitzPatrick Nuclear Power Plant noted that all performance indicators (PIs) and inspection areas were green (licensee response band), with the exception of the white (increased regulatory response band) PI for the high pressure coolant injection (HPCI) safety system unavailability performance indicator for the mitigating systems cornerstone. In addition, on December 29, 1999, we issued inspection report 05000333/99009 which contains the staff determination that the HPCI unavailability constituted a significant inspection finding per the NRC's significance determination process (SDP). Because the SDP characterization pertains to the same underlying issue as the performance indicator, the NRC considers this to be a single issue within a cornerstone. Additionally, consistent with pilot plant programmatic inspection requirements, the NRC is planning a supplemental inspection to review your long term corrective actions for this event.

The NRC has also identified a trend in the cross-cutting area of human performance. Although this trend has not resulted in any significant reductions in the margins of safety, we are providing it to enhance your station's performance in this important cross-cutting area. This human performance trend relates primarily to weaknesses in engineering and technical support performance. These weaknesses included testing of the HPCI system that contributed to system unavailability, system walkdowns that missed a number of material condition issues, entry of items into the corrective action system, and delays in ensuring that relevant issues were adequately communicated to operators. This issue does not require additional inspection and we will continue to monitor activities in this area through routine execution of the baseline inspection program.

This letter advises you of our planned inspection effort resulting from the James A. FitzPatrick Nuclear Power Plant mid-cycle performance review. Enclosure 1 lists the scheduled inspections that are planned through July 31, 2000. The inspection plan is provided to minimize the resource impact on your staff, and to allow for scheduling conflicts and personnel availability to be resolved in advance of inspectors arriving onsite. Routine resident inspections are not listed due to their ongoing and continuous nature. The last few months of the inspection plan are tentative and will be revised at the end-of-cycle performance review in April 2000, which we expect to issue to you in May 2000.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR). If circumstances arise which cause us to change this inspection plan, we will contact you to discuss the change as soon as possible. Please contact John Rogge at 610-337-5146 with any questions you may have regarding this letter or the inspection plan.

Sincerely,

Original Signed by:

A. Randolph Blough, Director
Division of Reactor Projects

Docket No. 05000333
License No. DPR-59

Enclosures: 1. James A. FitzPatrick Nuclear Power Plant Inspection/Activity Plan
2. Plant Issue Matrix

cc w/encl:
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R. Hiney, Executive Vice President for Project Operations
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Draft Final Rule

Modification of Event Reporting Requirements

10 CFR 50.72 and 50.73

(7)

Objectives

Section 50.72 provides immediate reporting of significant events where:

- Immediate NRC action may be required to protect the public health and safety or
- The NRC needs timely, accurate information to respond to heightened public concern

Section 50.73:

- Identifies the types of events and problems believed to be significant and useful to the NRC's effort to identify and resolve threats to public safety
- Is designed to provide information needed for engineering studies of anomalies, trend analysis of occurrences, and identification of accident precursors

Current rulemaking:

- Clarifies requirements
- Reduces unneccessary burden, consistent with risk considerations
- Is consistent with NRC program improvements

Principal Changes

Outside the Design Basis of the Plant

System Actuation

Invalid Actuation

Required Initial Reporting Times

Reporting of Historical Problems

Late Surveillance Tests

Outside the Design Basis of the Plant

In the proposed rule, we recommended deleting this criterion

Significant events would be captured by the following criteria:

- Event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to: shut down, maintain safe shutdown conditions, remove residual heat, control radioactive releases, or mitigate accidents.
- Plant in an unanalyzed condition that significantly degraded plant safety
- Principal safety barrier seriously degraded
- Condition or operation prohibited by the plant's technical specifications
- Independent trains or channels inoperable due to a single cause or condition
- A proposed new criterion – component in a degraded or non-conforming condition, such that its ability to perform its specified safety function is significantly degraded and the condition could reasonably be expected to apply to other similar components in the plant

Outside the Design Basis of the Plant (continued)

The draft final rule takes the following approach:

- The requirement to report a condition outside the design basis of the plant is deleted
- The new criterion is modified to address concerns raised in the comments and to more precisely address NRC needs
  - As modified, the criterion requires reporting <u>any event or condition that required corrective action for a single cause or condition in order to ensure the ability of more than one train or channel to perform its specified safety function</u>.
  - Events of this type indicate a condition where the NRC needs to consider taking action to ensure the cause or condition is adequately addressed at the reporting plant and/or other plants as appropriate.

Outside the Design Basis of the Plant (continued)

Additional guidance regarding the new criterion:

- The "reporting clock" starts when it is determined that corrective action is required for a single cause or condition in order to ensure the ability of more than one train or channel to perform its specified safety function.

- A written LER is due within 60 days. No telephone notification is required.

- This criterion involves corrective actions for significant conditions adverse to quality, under Criterion XVI, Appendix B.

- It does not include cases which merely involve checking of multiple trains or channels.

- The combination of removing the design basis criterion and adding this criterion is estimated, on balance, to result in fewer reports.

System Actuation

The proposed rule recommended reporting actuation for a list of systems provided in the rule to provide consistent reporting for actuation of a few standby systems that are risk-significant

Federal Register notice requested public comment, including three alternatives

The draft final rule includes a modified list of systems that provides for:

- Consistent reporting for the named systems, which are risk-significant
- A net reduction in reporting

In the future, as part of the effort to "risk-inform" 10 CFR Part 50, there may be an opportunity to develop plant-specific lists of systems of the most risk-significant systems in accordance with NRC-approved methods and criteria.

Invalid Actuation

In the proposed rule we recommended eliminating telephone notifications for invalid actuations and retaining the requirement for written LERs for these events.

Most commenters opposed any reporting of spurious actuations.

The draft final rule takes the following approach:

- The requirement to provide a telephone notification under §50.72 (i.e., within 8 hours) for an invalid actuation is eliminated.
- The requirement to report these events under §50.73 is retained. However:
  - The licensee has the option of providing a telephone notification.
  - The telephone notification may be made at any time within 60 days.

Required Initial Reporting Times

The draft final rule takes the following approach:

- One-hour reporting is required for:
  - Declaration of an emergency class
  - Deviation from the technical specifications under 10 CFR 50.54(x)

- Four-hour reporting is required for:
  - Unplanned transients (ECCS injection, required shutdown, critical scram)
  - Planned news release or notification to another government agency

- Eight-hour reporting is required for other §50.72 events

- Sixty-day reporting is required for reports submitted under §50.73.

- Three redundant criteria are deleted from §50.72

Reporting of Historical Problems

In the proposed rule we recommended using a three year cutoff for two specific types of events

Public comment recommended:

- Expanding the idea to other types of events
- Reducing the cutoff to two years

The draft final rule:

- Expands the idea to all reports under 50.72 and 50.73
- Uses a cutoff time of three years to better support performance indicators

Late Surveillance Tests

This change will eliminate reporting of late surveillance tests if the equipment, when tested, was still functional

Such events do not involve an impact on the capability to perform a specified safety function

Schedule

02/03/00	Complete briefing of ACRS
02/08/00	Complete briefing of CRGR
03/10/00	Provide final rule and guidelines to Commission
04/07/00	Provide final rule and guidelines to OMB for approval
06/23/00	Publish final rule
09/23/00	Effective date

(8)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
469TH	3	3.1

Title: Proposed Final Amendment to 10 CFR 50.72 and 50.73

Authors: NRC Staff

List of Documents Attached	
E-mail dated January 28, 2000, from Dennis Allison, NRR, to Noel Dudley, ACRS, Subject: Corrected Copy of "Noteworthy Issues"	3

Instructions to Preparer	From Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	N. Dudley

From: Dennis Allison
To: Noel Dudley
Date: Fri, Jan 28, 2000 2:23 PM
Subject: Corrected copy of "Noteworthy Issues"

Noel,

Sorry to trouble you. I discovered some errors in the brief document entitled "Noteworthy Issues" which I sent to you on Monday to support the briefing on 10 CFR 50.72 and 50.73. In a few places the words weren't quite right because they still referred to a different version of the rule. I've corrected these errors in the attached copy.

There are no changes to the longer Federal Register notice that I sent at the same time.

Dennis

CC: Cynthia Carpenter, Melinda Malloy

E-mail Attachment Received January 28, 2000

Noteworthy Issues

Outside the Design Basis of the Plant / Corrective Actions:

In the proposed rule, we recommended deleting the requirement to report when the plant is in a condition outside the design basis of the plant. A condition outside the design basis of the plant would still be reportable if it is significant enough to qualify under other criteria, such as:

- Plant in an unanalyzed condition that significantly degraded plant safety.
- Event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shut down the reactor, remove residual heat, control the release of radioactive material, or mitigate an accident.
- Condition or operation prohibited by the plant's technical specifications.
- Independent trains or channels inoperable due to a single cause or condition.
- Principal safety barrier seriously degraded.
- A proposed new criterion - component in a degraded or nonconforming condition, such that the ability to perform its specified safety function is significantly degraded and the condition could reasonably be expected to apply to other similar components in the plant.

The stated purpose of the proposed new criterion was to ensure continued reporting of design basis or other discrepancies if the capability to perform a specified safety function was significantly degraded and the condition had generic implications. Industry comments objected strongly, indicating that the proposed new criterion would be:

- Unclear and subject to widely varying interpretation.
- Overly burdensome, representing a significant increase in reporting requirements.
- Not in accordance with the stated objectives of the rulemaking.

The draft final rule takes the following approach:

- As was proposed, the requirement to report a condition outside the design basis of the plant is deleted.
- The new criterion has been modified to address the concerns about clarity and unnecessary burden that were raised in the comments. As modified, the criterion requires reporting an event or condition that requires corrective action for a single cause or condition in order to ensure the ability of more than one train or channel to perform its specified safety function.

- Events of this type indicate a condition where the NRC needs to consider taking action to ensure the condition is adequately addressed at the reporting plant and/or other plants as appropriate.

- For events of this type, the "reporting clock" starts when it is determined that corrective action is required for a single cause or condition in order to ensure the ability of more than one train or channel to perform its specified safety function.

- Then a written LER is due within 60 days. No telephone notification is required.

- This reporting criterion involves corrective actions for significant conditions adverse to quality, as required under Criterion XVI, Appendix B.

- However, this reporting criterion does not include cases which merely involve checking of multiple trains or channels.

- The combination of removing the design basis reporting criterion and adding this reporting criterion is estimated, on balance, to result in fewer reports.

System Actuation:

Currently, licensees are required to report actuation of "any ESF, including the RPS." In the proposed rule we recommended reporting actuation for a specific list of systems, to be provided in the rule. The stated purpose was to provide consistent reporting for actuation of a few standby systems that are highly risk-significant and eliminate reporting for events of lesser significance, such as actuation of control room ventilation systems.

Most commenters opposed this approach. They recommended that each plant report actuation for only those systems that have been identified as ESFs in the FSAR. The ACRS recommended that, rather than placing a generic list in the rule, the list of systems be determined for each specific plant, based on risk-significance of systems at that plant.

The draft final rule takes the following approach:

- A list of systems is included in the final rule.

- However, the list is modified as appropriate to address comments regarding specific items on the proposed list.

This provides for:

- Consistent reporting for the named systems, which are highly risk-significant.

- A net reduction in reporting.

In the future, as part of the effort to "risk-inform" 10 CFR Part 50, there may be an opportunity to develop plant-specific lists of systems of the most risk-significant systems in accordance with NRC-approved methods and criteria. At that time it will likely be appropriate to consider limiting the application of this and/or other reporting criteria to those systems.

Invalid Actuation:

In the proposed rule we recommended eliminating telephone notifications for invalid system actuations. This was proposed because spurious actuations, by themselves, are generally not significant events that the NRC needs to review in its search for safety problems. Thus, an immediate notification is not considered necessary.

In the proposed rule we also recommended retaining the requirement for a written LERs for invalid actuations. Information about invalid actuations is needed to support the NRC staff's estimates of equipment reliability.

Most commenters opposed any reporting of spurious ESF actuations. Among other things, they indicated that requiring a written LER is unnecessarily burdensome, considering the use of this particular information.

The draft final rule takes the following approach:

- As was proposed, the requirement to provide a telephone notification under §50.72 for an invalid system actuation is eliminated.

- As was also proposed, the requirement to report these events under §50.73 is retained.

- However, in order to reduce the burden of such reporting:
  - A licensee has the option of providing a telephone notification, which is less burdensome than providing a written LER.
  - The telephone notification may be made at any time within 60 days, because the information is not needed quickly.

Required Initial Reporting Times:

In the proposed rule we recommended that declaration of an emergency class and deviation from the technical specifications under 10 CFR 50.54(x) continue to be reportable within 1 hour. Other events reportable by telephone under §50.72 would be reportable within eight hours. Reports required under §50.73 would be required within 60 days. Most commenters supported this approach, but two States and our Incident Response Organization have expressed concerns about waiting eight hours for reporting of certain events.

The draft final rule takes the following approach:

As was proposed, two types of events continue to be reportable within 1 hour:

- *Declaration of an emergency class.*

- *Deviation from the technical specifications authorized pursuant to §50.54(x).*

Three criteria are deleted from §50.72, but retained in §50.73:

- *A natural phenomenon or other external event that poses an actual threat to plant safety, or significantly hampers site personnel in the performance of duties necessary for safe*

operation. Events of this type are captured by *declaration of an emergency class*, which is reportable within 1 hour.

- *An internal event that poses an actual threat to plant safety, or significantly hampers site personnel in the performance of duties necessary for safe operation, including fires, toxic gas releases, or radioactive releases.* Events of this type are captured by *declaration of an emergency class*, which is reportable within 1 hour.

- *An airborne radioactive release, or liquid effluent release, that exceeds specific limits.* A release that warrants prompt notification is captured by *declaration of an emergency class*, which is reportable within 1 hour. If not reported within 1 hour, an inadvertent release is reportable within 4 hours if *a news release is planned or notification to other government agencies has been or will be made*, as discussed further below.

Four-hour reporting is required for unplanned transients, if not reported in 1 hour. These are events where there may be a need for the NRC to take a reasonably prompt action, such as partially activating its response plan to monitor the course of the event. In summary, they are:

- *Initiation of a shutdown required by the plant's technical specifications.* Previously this was reportable within 1 hour.

- *An event that results or should have resulted in a valid ECCS discharge into the RCS, except when it results from and is part of a pre-planned sequence during testing or operation.* Previously this was reportable within 1 hour, whether or not it was part of a pre-planned sequence during testing or operation.

- *A scram when critical, except when it results from and is part of a pre-planned sequence during testing or operation.* Previously, actuation of any ESF, including the RPS, was reportable within 4 hours, except when it resulted from and was part of a pre-planned sequence during testing or operation.

Four-hour reporting is also required for an *event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials.* These reports are needed promptly because they involve events where there may be a need for the NRC to respond to heightened public concern.

Eight-hour reporting is required for other events reportable under §50.72, if not reported in 1 hour or 4 hours. These are events where there may be a need for the NRC to take an action within about a day, such as initiating a special inspection or investigation. In summary, they are:

- *The plant including its principal safety barriers being in a seriously degraded condition, or the plant being in an unanalyzed condition that significantly degrades plant safety.*

- *A valid actuation of any of the systems named in the rule, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.*

- *An event or condition that at the time of discovery could have prevented fulfillment of the safety function of structures or systems needed to shut down the reactor, remove residual heat, control the release of radioactive material, or mitigate an accident.*

- *Transport of a radioactively contaminated person to an offsite medical facility for treatment.*

- *A major loss of emergency assessment capability, offsite response capability, or offsite communications capability.*

As was proposed, reports required under §50.73 are due within 60 days after discovery of a reportable event or condition. instead of within 30 days as is currently required. This change does not imply that licensees should take longer than they previously did to develop and implement corrective actions. They should continue to do so on a time scale commensurate with the significance of the issue. However, for those cases where it does take longer than 30 days to complete a root cause analysis, there would be fewer LERs that require amendment (by submitting a revised report).

Entry Into Technical Specification 3.0.3 or Its Equivalent:

TS 3.0.3 establishes requirements for actions to be taken when an LCO is not met and: (1) the associated actions are not met, (2) the associated actions direct entry into TS/LCO 3.0.3, or (3) no associated actions are provided. From Mode 1 (Power Operation), TS 3.0.3 typically requires initiation of plant shutdown within 1 hour to place the unit in Mode 2 (Startup) within 7 hours, Mode 3 (Hot Shutdown) within 13 hours, and Mode 4 (Cold Shutdown) within 37 hours, as applicable. The current reporting guidelines in NUREG-1022 indicate that entry into Technical Specification 3.0.3 or its equivalent for any reason is reportable as an "operation or condition prohibited by the plant's technical specifications." Most commenters recommended placing some limitations on the reportablility of these events.

The draft final rule takes the following approach:

- Entry into TS 3.0.3 is not necessarily reportable.

- The event becomes reportable when a required shutdown is initiated.

Licensee Event Reporting System

February 3, 2000

James Davis
Nuclear Energy Institute

NEI

Rulemaking process--

Key to achieving a clear, useable, enforceable rule

- Build on 6 years of effort
- Involve key players--regions, operators, other stakeholders
- Extensive use of workshops
  - ANPR review
  - Table top discussion of specific language
  - Workshop on final language

NEI

Operability Determination

- Is a well defined process
  - Focus on system ability to perform a **safety function**
  - Industry knows how to perform
- Supports Technical Specifications and basis for operational decisions
  - One analysis, highly focused and consistent

NEI

Draft rule, as issued, had a significant problem

- Did not meet NRC's stated objectives
  - Better align with reporting needs--no!
    - Adds design basis at component level
  - Reduce reporting burden--no!
    - Net increase in number of reports
  - Clarify requirements--no!
    - Increased level of confusion--degraded--significant--similar--could reasonably
- Industry could not support

NEI

What is the issue?

- Reporting degraded components
  - 50.73(a)(2)(ii)(C)
  - Not part of the initial rule proposal
  - Added to final rule issued for comment
- Significantly increases burden
  - Must have a backfit analysis
- Requirement is vague and will be a disaster in implementation

NEI

Data Collection?

"A component being in a degraded or non-conforming condition such that the ability of the component to perform its specified safety function is significantly degraded and the condition could reasonably be expected to affect other similar components in the plant"

- ANPR--No longer require reporting…outside design basis.
- FRN--ensure design basis…would continue to be reported

NEI

Examples in 1022 do not make sense

- Licensee would have looked at similar components
- Would be addressed in corrective action program
- System safety function not affected--else would have been reported under other criteria
- Not clear what this is trying to do

NEI

Public Comments

- Degraded Components the key focus of comments to the NRC
  - NEI comments
  - Many facility comments
- Industry appreciates staff effort to resolve issue
- The key issue-- "changed wording" should be tested by operators and regional staff

NEI

We are confused

- The industry needs more information
- Can the staff develop clear examples for NUREG 1022?
- Will the staff hold a public meeting and explain how this will work?
- Can we see the backfit analysis that justifies this new requirement?

NEI

Remove degraded components and---

- The draft rule improves clarity
  - Worked by regional staffs and operators
- Provides a clear focus and nexus to safety
  - One that can be understood
  - Uses consistent operability determination process
- Would eliminate unnecessary reports
- Could be a great end to an 8 year effort

NEI

What next--three options

- Eliminate the requirement
- Separate the degraded component issue from the current rulemaking effort
  - Justify specifically
  - Do the needed backfit analysis
- Stop the process and address event reporting as part of effort to harmonize part 50

NEI

STATUS OF 10 CFR 50.59 GUIDANCE

February 3, 2000

Eileen M. McKenna
Office of Nuclear Reactor Regulation

10

Background

- Final Rule approved June 22, published October 4, 1999 (64 FR 53582)
- Rule revisions become effective 90 days after approval of guidance
- RG is expected to be endorsement of NEI 96-07 (revision)

Current Status

- Draft revisions submitted in 1999 and reviewed by NRC
- Revised NEI 96-07 submitted for NRC endorsement January 18, 2000
- NRC letter with staff comments to be issued early February; meeting with NEI planned for February 9 to discuss open issues
- Commission briefing scheduled for February 29
- Publish draft RG for public comment April 2000

Changes to Rule Requirements

- Organization and format
- Definitions (change, facility, departure from method...)
- Screening capability (using definitions)
- Evaluation criteria ("minimal" increases, design basis limits, departure from methods of evaluation)
- Other Clarifications and Conforming changes

OPEN ISSUES

- Fire Protection plan (and facility) changes
  - GL 86-10 license condition (plan in FSAR, use 50.59)
  - proposal is to use license condition on its own w/o 50.59
  - staff concern is with other process aspects (records, bases)

- Methods
  - clarifications needed on "essentially the same"
  - guidance on plant-specific "approvals"

- Design Basis limits for fission product barriers
  - "subordinate" limits concept not accepted
  - staff concerns with "95/95 DNB" as the fuel DBL

- Screening on design function (examples)

OPEN ISSUES (continued)

- **Numerical values**
  - staff has reached general agreement with the proposal (but some clarifications needed)

- **Relationship to Maintenance Assessments**
  - NEI proposed that "changes associated with maintenance" be covered by maintenance rule (a)(4) assessments, not 50.59
  - details of proposal still under review

(11)

NEI 96-07, Revision 1, Guidelines for 10 CFR 50.59 Evaluations

NEI Presentation to ACRS
February 3, 2000
Russ Bell

Past as Prologue

- NSAC-125 (1989)
- NRC lessons learned reviews
- NEI 96-07 & Industry Initiative
- Draft NUREG-1606 (SECY-97-035)
- Generic Letter 91-18, Revision 1
- Rulemaking ending with SRM/SECY-99-130
- NEI 96-07, Revision 1

Industry Objective

Attain stability and clarity in a key regulatory process that provides licensees with appropriate flexibility to make changes to their facilities

NEI 96-07, Revision 1

- Objectives
  - Clear, comprehensive guidance
  - More consistent, effective implementation
  - Common understanding with NRC via endorsement in a regulatory guide
- Status: Revision entering final stages
  - Industry & NRC comments provided on September 17 draft
  - Final draft sent to NRC January 18

10 CFR 50.59 Process

- Does 10 CFR 50.59 apply to the proposed activity?
- Should the proposed activity be evaluated against the eight criteria of 10 CFR 50.59(c)(2)?
- Does the proposed activity require prior NRC approval?

Screening Process

- Screening is the process for identifying changes that require evaluation under 10 CFR 50.59
- NEI 97-07, R1, provides guidance for more effective screening
- Screening is based definitions of:
  - "Change"
  - "Facility/Procedures as described in the UFSAR"
  - "Tests or experiments not described in the UFSAR"

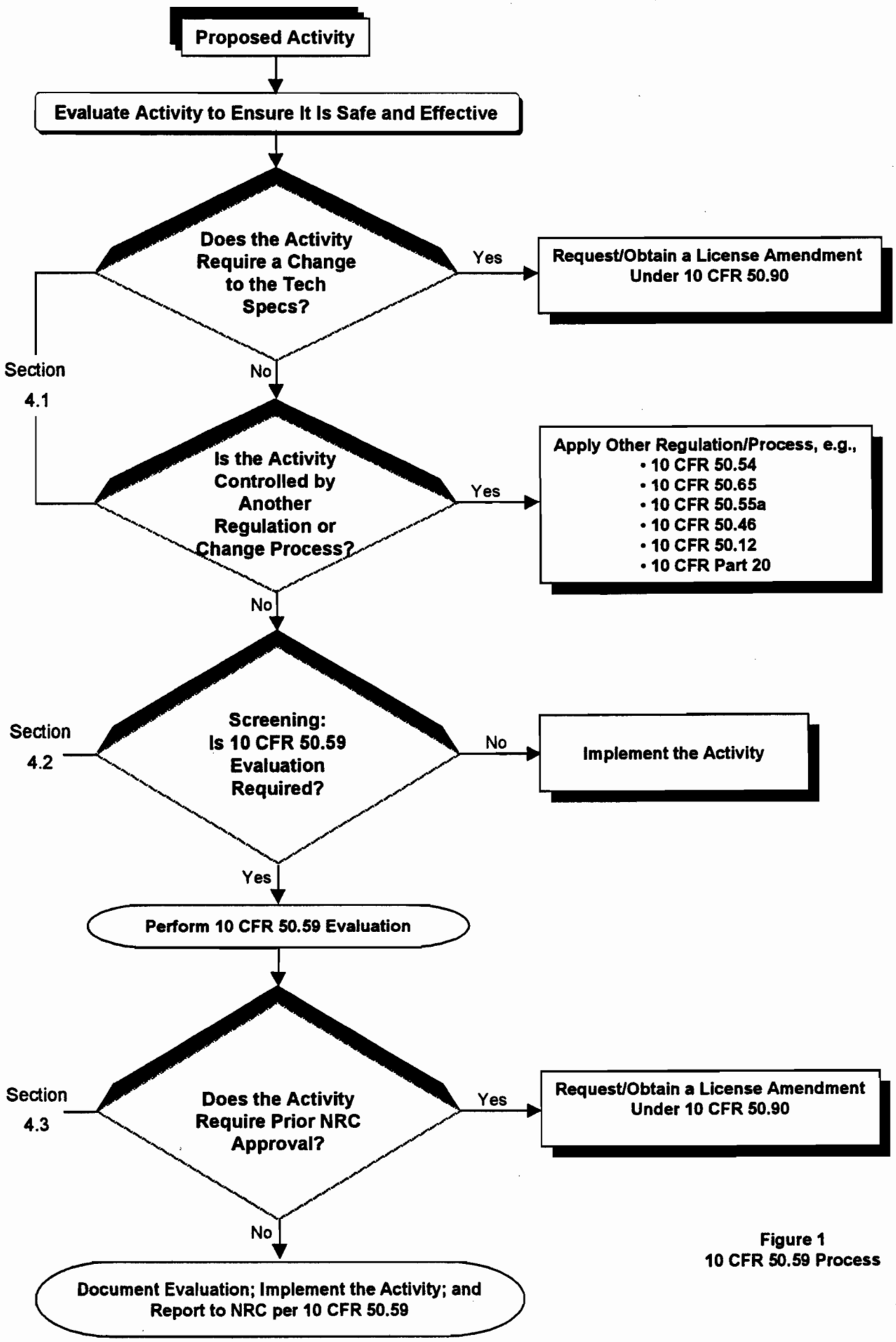

Figure 1
10 CFR 50.59 Process

Screening Questions

- Does the activity affect a UFSAR-described (1) design function, (2) method of performing or controlling the function, or (3) an evaluation that demonstrates intended design functions will be accomplished?
- Is the activity a test or experiment not described in the UFSAR?

Evaluation Process
Is prior NRC approval required?

- Is there more than a minimal increase in the frequency of an accident or likelihood of malfunction [c(2)(i&ii)]?
  - Qualitatively determined
  - Considerations provided in guidance

Is prior NRC approval required? (cont.)

- Is there more than a minimal increase in the consequences of an accident or malfunction [c(2)(iii&iv)]?
  - Quantitatively determined
  - Limits based on GDC 19, Part 100, and SRP
    - 10% of margin to GDC 19 or Part 100 limit
    - Not to exceed applicable SRP guideline
- Is there a possibility of an accident of a different type [c(2)(v)]?

Is prior NRC approval required? (cont.)

- Is there a possibility of malfunction with a different result [c(2)(vi)]?
- Is a design basis limit for a fission product barrier exceed or altered [c(2)(vii)]?
  - Quantitative determination
  - Design basis limits assure confidence in fission product barrier integrity
  - Typical design basis limits identified in NEI 96-07, R1

(c)(2)(vii) - Fission Product Barriers
Example Evaluation

Evaluate acceptance of as-found AFW flow rate, assuming all required functions are met by the reduced rate

- Is a parameter affected that controls the integrity of a fission product barrier?
  - "Yes" (RCS pressure and pressurizer level)
- Are the design basis limits for these parameters exceeded or altered? Compare to:
  - RCS design pressure
  - 100% pressurizer level

Is prior NRC approval required? (cont.)

- Is there a departure from a method of evaluation used in establishing the design bases or in the safety analyses[c(2)(viii)]?
  - If changing an element of a methodology, are results conservative or essentially the same?
  - If changing from one method to another, is the new method approved by the NRC for the intended application?

Summary

- On course toward NRC endorsement of NEI 96-07, Revision 1
- Remaining issues to be addressed this month
- Commission briefing February 29
- NEI workshop set for April 10-11

More consistent, effective and efficient 10 CFR 50.59 implementation ahead

(12)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
469th	4	4

Title

NEI 96-07

Authors

Med El-Zeftawy

List of Documents Attached	
Memorandum dated January 31, 2000, Subject: NEI Document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations"	4

Instructions to Preparer	From Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	Med El-Zeftawy

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

January 31, 2000

MEMORANDUM TO: John D. Sieber
ACRS Member

FROM: Med El-Zeftawy, Sr. Staff Engineer M.Z.
ACRS

SUBJECT: NUCLEAR ENERGY INSTITUTE DOCUMENT 96-07, "GUIDELINES FOR 10 CFR 50.59 SAFETY EVALUATIONS"

The Nuclear Energy Institute (NEI) has modified its document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations" dated December 20, 1999 (Revision 1 C) which I included in the February 3, 2000 ACRS meeting notebook. The attached is NEI 96-07, Revision 1 (Final Draft) , dated January 18, 2000. The final draft reflects consideration of NRC staff comments on the December 20, 1999 version.

The staff is still discussing the final draft with NEI representatives, and the following are the major topics that need coordination on acceptability:

- Methods of evaluation approved for the intended application-- substantial new text and examples were added to describe how licensees would determine if changes to a method are a "departure", including use of a different method, which was approved by the NRC for a particular plant and whether this is acceptable for use by another licensee under 50.59.

- Design basis limits (DBL) for fuel-- latest version proposes "subordinate" characterization for certain parameters that are numerical "limits" in the FSAR related to fission product barrier integrity, but which would not be DBL, e.g., fuel burnup.

- Fire Protection-- discussion indicates that changes to fire protection programs and plans will be according to the license condition standard (not adversely affect the ability to maintain safe shutdown), and that a 50.59 evaluation (against the evaluation criteria) is not required. In parallel, NEI is proposing that draft fire regulatory guide be revised with respect to change process using the license condition directly without a 50.59.

- Temporary Changes (maintenance)-- the text regarding applicability was revised to indicate that "temporary changes", which are performed to support maintenance and will be restored to the original FSAR configuration after the maintenance, are assessed using 50.65 and are not changes per 50.59.

- Minimal increase in frequency of accidents and of likelihood of malfunction (quantitative)-- proposal for a 10E-6 threshold for frequency of a design basis (FSAR

initiator) accident needs to be reviewed. In addition, the likelihood of malfunction and applicability at the component or system level needs further examination.

Also attached , is a list provided by the staff that outlines the modifications made by NEI to the December 20, 1999 version of the NEI 96-07 document.

Attachments: As Stated

cc via e-mail (w/o attachments):

ACRS Members
J. Larkins
H. Larson
R. Savio
S. Duraiswamy

MODIFICATIONS MADE TO 12/20 VERSION OF NEI 96-07

NOTE: This is based upon informal comments from NEI concerning the changes they made. Thus, references within the items to "we" refer to NEI's position. Reference is also made to a January 6 meeting with the NRC staff at which many of these items were discussed. The header in the left margin refers to the general topic area that the changes are applicable to (as for example, maintenance, or screening of changes). Bracketed comments are the staff's.

SCREEN 1. design function (as used as part of definition of "change" in section 3.3) - clarified broad intent during 1/6 meeting; clarifications made to examples 2 & 3 in Sect. 4.2.1.1; [also see item 16, below]. Discussion of design bases definition (section 3.5) has been deleted.

MAINT 2. temp changes to support maintenance - reiterated guidance at 1/6 mtg; some wording changes to guidance in 3.3 and 4.1.2; companion MR guidance submitted on 1/10

SCREEN 3. equivalent replacements - Instead of revising the bullets on equivalence, we've deleted that entire portion of Section 4.1.2 [applicability]. Equivalence is now identified in Section 4.2.1.1 as a basis for screening changes to the facility. Example added to 4.2.1.1.

NUMERIC 4. frequency increases - typos corrected

NUMERIC 5. external event frequencies - 4.3.1&2 guidance changed to focus on frequency of natural phenomena [not just "external events"]

NUMERIC 5.5 [re numerical values]: We believe the existing guidance in 4.3.1 and 4.3.2 is adequate regarding identifying the accidents (plural) and SSCs (plural) that a change may affect. No changes planned. [see also item 17)]

METHODS 6. cross-reference provided from 3.8 to examples in 4.2.1.3.

METHODS 7 Last sentence of Sect 3.4 strengthened [re referencing to section 4.8.3]

METHODS 8. We did not find the words to address the issue of sub-references as a possible source for methods of evaluation subject to c.8.We could address this in a supplemental Q&A, or if you feel strongly that it must be dealt with in the guidance, you could suggest something. Our general agreement with the staff on this subject remains intact based on the Jan 6. discussion. [i.e., Agree that sub-references may contain methods subject to control under 50.59]

METHODS 9. No changes to section 4.2.1.3 on screening of methods to address SERs that lack comprehensive list of limitations/constraints; clarified intent during 1/6 mtg [see also item 19]

METHODS 10. "upgraded" substituted for "updated" in Sect 4.3.8 [re methods approved by NRC]

DBL 11. Sect 4.3.7 (after table) Design basis limits - modified to add fuel burn-up as another example of a parameter that may be considered a dbl for fpb by some licensees

DBL	12. No change concerning 95/95 DNB [as being a DBL for fuel]- clarified at 1/6 meeting.
FIRE	13. Deleted several sentences of FP guidance from 4.1.5 as suggested (deferred to the FP reg guide).
OTHER	14. Added footnote as follows to Sect 4.3.3: "For licensees who adopt the alternative source term, evaluations against this criterion should be in terms of total effective dose equivalent and the limits established by 10 CFR 50.67 (effective January 24, 2000)."
OTHER	15. In 4.3.6, replaced "for reasons" with "in ways" [example regarding FW control]
SCREEN	16. In 4.2.1.1, modified 1st example to eliminate sentence containing "could affect" [diesel relay example]
NUMERIC	17. deleted last 3 sentences of 4.3.2.c.8. Earlier guidance to evaluate changes at the level of analyses existing in the UFSAR is considered sufficient concerning where to apply the factor of 2 [for malfunctions]
SCREEN	18. Added statement to 4.2.2 that tests or experiments on out-of-service SSCs may be screened out provided affected SSCs are appropriately isolated from the facility.
METHODS	19 Added example to 4.2.1.3 (as discussed on 1/6) allowing the change from 3-D to 2-D core modeling, or vice versa, in the case where the topical/SER explicitly allows either.

NUCLEAR ENERGY INSTITUTE

Anthony R. Pietrangelo
DIRECTOR, LICENSING
NUCLEAR GENERATION

January 18, 2000

Mr. David B. Matthews
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

PROJECT NUMBER: 689

Dear Mr. Matthews:

Enclosed for NRC endorsement is NEI 96-07, Revision 1 (Final Draft), *Guidelines for 10 CFR 50.59 Evaluations*. The enclosure reflects consideration of NRC comments on the previous draft discussed in our public meeting on January 6.

We understand that the staff intends to endorse the industry guideline in a draft regulatory guide to be published in the *Federal Register* in February for public comment. Our objective is NRC endorsement of the guidance without exception. Thus we are prepared to discuss with the staff any remaining issues, as needed, to support the current schedule.

If you have any questions concerning final draft NEI 96-07, Revision 1, please contact me at 202-739-8081, or Russ Bell at 202-739-8087.

Sincerely

Anthony R. Pietrangelo

Enclosure

c: Eileen McKenna

1776 I STREET, NW SUITE 400 WASHINGTON, DC 20006-3708 PHONE 202.739.8000 FAX 202.785.4019 www.nei.org

NEI 96-07, Revision 1 [Final Draft]

Nuclear Energy Institute

GUIDELINES FOR 10 CFR 50.59 EVALUATIONS

FINAL DRAFT – January 18, 2000

Nuclear Energy Institute, 1776 I Street N.W., Suite 400, Washington D.C. (202.739.8000)

ACKNOWLEDGMENTS

In 1996, NSAC-125, *Guidelines for 10 CFR 50.59 Safety Evaluations,* was transformed into NEI 96-07 with minor changes to address specific NRC concerns. Much of this longstanding industry guidance continues to underlie the revised guidance presented in this document. We appreciate EPRI allowing NEI to use NSAC-125 in this manner and we recognize the efforts of the individuals that contributed to the development of NSAC-125.

The revised guidance in this document was developed with the invaluable assistance of the 10 CFR 50.59 Task Force and the Regulatory Process Working Group.

Nuclear Energy Institute, 1776 I Street N.W., Suite 400, Washington D.C. (202.739.8000)

FOREWORD

In 1999, the NRC revised its regulation controlling changes, tests and experiments performed by nuclear plant licensees—the first changes to 10 CFR 50.59 in over 30 years. The changes were prompted by the need to resolve differences in interpretation of the rule's requirements by the industry and the NRC that came in clear focus in 1996. These differences existed despite general recognition that licensee implementation of 10 CFR 50.59 has been effective in controlling activities affecting plant design and operation. The rule changes had two principal objectives, both aimed at restoring much-needed regulatory stability to this extensively used regulation:

- Establish clear definitions to promote common understanding of the rule's requirements
- Clarify the criteria for determining when changes, tests and experiments require prior NRC approval

While effective at controlling changes, 10 CFR 50.59 was, at the same time, viewed as overly restrictive of licensee changes and unduly burdensome. License amendment requests were prepared, submitted and reviewed by the NRC for many changes having little or no impact on the plant design or operation. Indeed, some beneficial changes were withdrawn by licensees upon determination that the change would have to go through the burdensome license amendment process. Moreover, substantial resources were expended each year by licensees to process and submit to NRC lengthy evaluations for numerous insignificant changes. The changes approved by the Commission in 1999 made 10 CFR 50.59 more focused and efficient by:

- Providing greater flexibility to licensees, primarily by allowing changes that have minimal safety impact to be made without prior NRC approval
- Clarifying the threshold for "screening out" changes that do not require full evaluation under 10 CFR 50.59, primarily by adoption of key definitions

These changes will conserve both licensee and NRC resources while continuing to ensure that significant changes are thoroughly evaluated and approved by the NRC as appropriate.

This document provides guidance for implementing the revised rule. While it contains new guidance corresponding to new and revised rule criteria, overall, the document reflects a refinement of longstanding industry practice, not a radical new

approach. The basic philosophy behind 10 CFR 50.59 implementation and a substantial amount of guidance reflected in this document can be traced to EPRI/NSAC-125—the original industry guidance document in this area—issued in 1989.

Other past guidance related to 10 CFR 50.59, including NRC generic communications, was also reviewed and reflected in this document as appropriate. The intent is to provide comprehensive guidance that is consistent with the 1999 changes to 10 CFR 50.59.

In parallel with the rulemaking to amend 10 CFR 50.59, the NRC made conforming changes to the analogous provision in 10 CFR Part 72 for control of changes, tests and experiments involving independent fuel storage facilities. The intent of conforming 10 CFR 72.48 to the terms of 10 CFR 50.59 was to provide for consistent implementation of these two analogous regulations. Accordingly, the guidance herein on implementing 10 CFR 50.59 may be applied to support implementation of 10 CFR 72.48.

TABLE OF CONTENTS

APPENDICES

1 INTRODUCTION

1.1 PURPOSE

10 CFR 50.59 establishes the conditions under which licensees may make changes to the facility or procedures and conduct tests or experiments without prior NRC approval. Proposed changes, tests and experiments (hereafter referred to collectively as activities) that satisfy the definitions and one or more of the criteria in the rule must be reviewed and approved by the NRC before implementation. Thus 10 CFR 50.59 provides a threshold for regulatory review—not the final determination of safety—for proposed activities.

The purpose of this document is to provide guidance for developing effective and consistent 10 CFR 50.59 implementation processes.

1.2 RELATIONSHIP OF 10 CFR 50.59 TO OTHER REGULATORY REQUIREMENTS AND CONTROLS

As the process for controlling most activities affecting equipment and procedures at a nuclear power plant, implementation of 10 CFR 50.59 interfaces with many other regulatory requirements and controls. To optimize the use of 10 CFR 50.59, the rule and this guidance should be understood in the context of the proper relationship with these other regulatory processes. These relationships are described below:

1.2.1 Relationship of 10 CFR 50.59 to Other Processes that Control Licensing Basis Activities

10 CFR 50.59 focuses on the effects of proposed activities on the safety analyses that are contained in the updated FSAR (UFSAR) and are a cornerstone of each plant's licensing basis. In addition to 10 CFR 50.59 control of changes affecting the safety analyses, there are several other complementary processes for controlling activities that affect other aspects of the licensing basis:

- Amendments to the Operating License (including the technical specifications) are sought and obtained under 10 CFR 50.90.

- Where changes to the facility or procedures are controlled by more specific regulations (e.g., quality assurance, security and emergency preparedness program changes controlled under 10 CFR 50.54(a),

(p) and (q), respectively; Off-site Dose Calculation Manual changes controlled by technical specifications), 10 CFR 50.59 states that the more specific regulation applies.

- Changes that require an exemption from a regulation are processed in accordance with 10 CFR 50.12.

- Guidance for controlling changes to licensee commitments is provided by NEI 99-04, *Guideline for Managing NRC Commitment Changes*.

- Where a licensee possesses a license condition which specifically permits changes to the NRC-approved fire protection program (i.e., has received the standard fire protection license condition contained in Generic Letter 86-10), subsequent changes to the fire protection program would be controlled under the license condition and not 10 CFR 50.59.

- Maintenance activities, including associated temporary changes, are subject to the technical specifications and are assessed and managed in accordance with the Maintenance Rule, 10 CFR 50.65; screening and evaluation under 10 CFR 50.59 is not required.

Together with 10 CFR 50.59, these processes form a framework of complementary regulatory controls over the licensing basis. To optimize the effectiveness of these controls and minimize duplication and undue burden, it is important to understand the scope of each process within the regulatory framework. This guideline discusses the scope of 10 CFR 50.59 in relation to other processes, including circumstances under which different processes, e.g., 10 CFR 50.59 and 10 CFR 50.90, should be applied to different aspects of an activity.

In addition to controlling changes to the facility and procedures described in the UFSAR under 10 CFR 50.59 as required by the rule, some licensees also control changes to other licensing basis information using the 10 CFR 50.59 process. This may be in accordance with a requirement of the license or commitment to the NRC. An example of documentation that may be outside the UFSAR but that is controlled via 10 CFR 50.59 by many licensees are the Technical Specifications Bases.

1.2.2 Relationship of 10 CFR 50.59 to 10 CFR Part 50, Appendix B

Prior to the operating license, 10 CFR Part 50, Appendix B, assures that the facility design and construction meet applicable requirements, codes and standards in accordance with the safety classification of systems, structures and components (SSCs). Appendix B design control provisions ensure that all changes continue to meet applicable design and quality requirements. The design and licensing bases evolve in accordance with Appendix B requirements up to the time that an operating license is received, and 10 CFR 50.59 is not applicable until after that time. Both Appendix B and 10 CFR 50.59 apply following receipt of an operating license.

Appendix B also addresses corrective action. The application of 10 CFR 50.59 to corrective actions that address degraded and non-conforming conditions is described in Section 4.4.

1.2.3 Relationship of 10 CFR 50.59 to the UFSAR

The 10 CFR 50.59 is the process that identifies when a license amendment is required prior to implementing changes to the facility or procedures described in the UFSAR or tests and experiments not described in the UFSAR. As such, it is important that the FSAR be properly maintained and updated in accordance with 10 CFR 50.71(e). Guidance for updating UFSARs to reflect activities implemented under 10 CFR 50.59 is provided by Regulatory Guide 1.181, which endorses NEI 98-03, Revision 1.

1.2.4 Relationship of 10 CFR 50.59 to 10 CFR 50.2 Design Bases

10 CFR 50.59 controls changes to both 10 CFR 50.2 design bases and supporting design information contained in the UFSAR. In support of 10 CFR 50.59 implementation, Section 4.3.7 of this guideline defines the design basis limits for fission product barriers that are subject to control under 10 CFR 50.59(c)(2)(vii), and Section 4.3.8 provides guidance on the scope of methods of evaluation used in establishing design bases or in the safety analyses that are subject to control under 10 CFR 50.59(c)(2)(viii). Additional guidance for identifying 10 CFR 50.2 design bases is provided in NEI 97-04, Appendix B.

1.3 10 CFR 50.59 PROCESS SUMMARY:

After determining that a proposed activity is safe and effective through appropriate engineering and technical evaluations, the 10 CFR 50.59 process is applied to determine if a license amendment is required prior to implementation. This process involves the following basic steps as depicted in Figure 1:

- **Applicability and Screening:** Determine if a 10 CFR 50.59 evaluation is required.
- **Evaluation:** Apply the eight evaluation criteria of 10 CFR 50.59(c)(2) to determine if a license amendment must be obtained from the NRC.
- **Documentation & reporting:** Document and report to the NRC activities implemented under 10 CFR 50.59.

Later sections of this document discuss key definitions, provide guidance for determining applicability, screening, and performing 10 CFR 50.59 evaluations, and present examples to illustrate the application of the process.

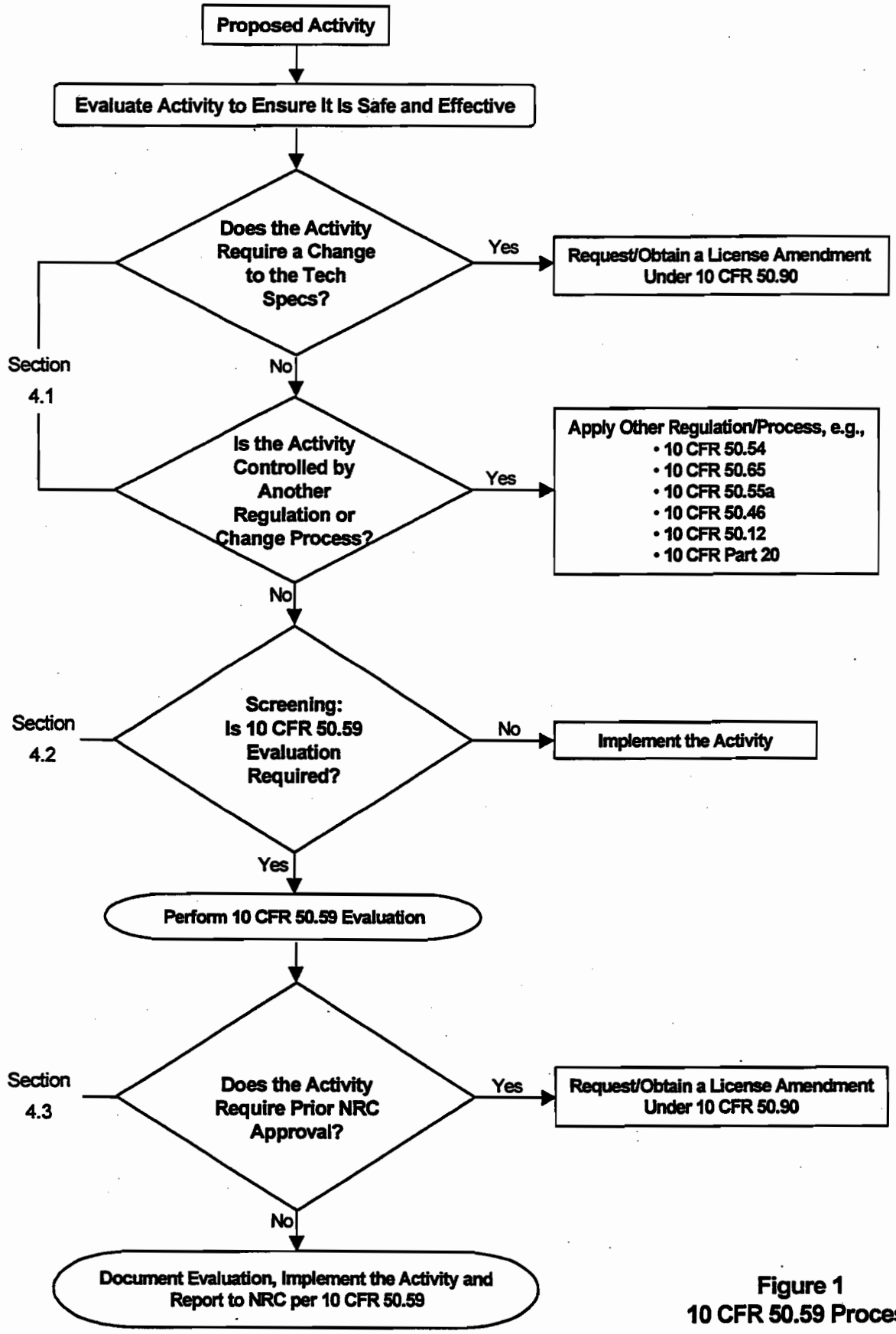

Figure 1
10 CFR 50.59 Process

1.4 APPLICABILITY TO 10 CFR 72.48

Concurrent with the rulemaking to amend 10 CFR 50.59, the NRC made conforming changes to the analogous provisions in 10 CFR 72.48 controlling licensee changes, tests and experiments to independent spent fuel storage installations (ISFSIs). The provisions of 10 CFR 72.48 were also extended to holders of Part 72 Certificates of Compliance. As a result, 10 CFR 72.48 establishes criteria identical to those in 10 CFR 50.59 under which both an ISFSI license holder and a certificate holder may make changes to the facility or cask design, changes to procedures and conduct tests or experiments without prior NRC approval.

The intent of conforming 10 CFR 72.48 to the terms of 10 CFR 50.59 was to provide for consistent implementation of these two analogous regulations. Consistent with this intent, the guidance herein on implementing 10 CFR 50.59 may be applied to support implementation of 10 CFR 72.48.

1.5 CONTENT OF THIS GUIDANCE DOCUMENT

The NRC has established requirements for nuclear plant systems, structures and components to provide reasonable assurance of adequate protection of the public health and safety. Many of these requirements, and descriptions of how they are met, are documented in the updated FSAR (UFSAR). 10 CFR 50.59 allows a licensee to make changes in the facility or procedures as described in the UFSAR, and to conduct tests or experiments not described in the UFSAR, unless the changes require a change in the technical specifications or otherwise require prior NRC approval. In order to perform 10 CFR 50.59 screenings and evaluations, an understanding of the design and licensing basis of the plant and of the specific requirements of the regulations is necessary. Individuals performing 10 CFR 50.59 screenings and evaluations should also understand the rule and concepts discussed in this guidance document.

In Section 2, the relationship between the design criteria established in 10 CFR 50, Appendix A, and 10 CFR 50.59 is discussed as background for applying the rule.

Section 3 presents definitions and discussion of key terms used in 10 CFR 50.59 and this guideline.

Section 4 discusses the application of the definitions and criteria presented in 10 CFR 50.59 to the process of changing the plant or procedures and the

conduct of tests or experiments. This section includes guidance on the applicability requirements for the rule, the screening process for determining when a 10 CFR 50.59 evaluation must be performed, and the eight evaluation criteria for determining if prior NRC approval is required. Examples are provided to reinforce the guidance. Guidance is also provided on dispositioning and documenting 10 CFR 50.59 evaluations and reporting to NRC.

Section 5 provides guidance on documenting 10 CFR 50.59 evaluations and reporting to NRC.

Appendix A provides the text of 10 CFR 50.59 as published in the *Federal Register* on October 4, 1999. Appendix B provides the text of revised 10 CFR 72.48 as well as examples [FUTURE] illustrating the application of this guidance to changes involving independent spent fuel storage installations and spent fuel storage cask designs.

2.0 DEFENSE IN DEPTH DESIGN PHILOSOPHY AND 10 CFR 50.59

One objective of Title 10 of the Code of Federal Regulations is to establish requirements directed toward protecting the health and safety of the public from the uncontrolled release of radioactivity. At the design stage, protection of public health and safety is ensured through the design of the engineered protection of physical barriers to guard against the uncontrolled release of radioactivity. Other sources of radioactivity including radwaste systems are included. The defense-in-depth philosophy includes reliable design provisions to safely terminate accidents and provisions to mitigate the consequences of accidents. The three physical barriers that provide defense-in-depth are:

- Fuel Clad
- Reactor Coolant System Boundary
- Containment Boundary

These barriers perform a health and safety protection function. They are designed to reliably fulfill their operational function by meeting all criteria and standards applicable to mechanical components, pressure components, and civil structures. These barriers are protected extensively by inherent safety features and through the implementation of engineered safety features. The public health and safety protection functions are analytically demonstrated and documented in the UFSAR. Analyses summarized in the UFSAR demonstrate that under the assumed accident conditions, the consequences of accidents challenging the integrity of the barriers will not

exceed limits based on the criteria established in GDC 19 or the guidelines established in 10 CFR 100. Thus, the UFSAR analyses provide the final verification of the nuclear safety design phase by documenting plant performance in terms of public protection from uncontrolled releases of radiation. 10 CFR 50.59 addresses this aspect of design by requiring prior NRC approval of proposed activities which, although safe, require a technical specification change or meet specific threshold criteria for NRC review.

This protection philosophy pervades the UFSAR accident analyses and Title 10 of the CFR. To understand and apply 10 CFR 50.59, it is necessary to understand this perspective of maintaining the integrity of the physical barriers designed to contain radioactivity. This is because:

- UFSAR accidents and malfunctions are analyzed in terms of their effect on the physical barriers. There is a relationship between barrier integrity and dose.

- The principal "consequences" that the physical barriers are designed to preclude is the uncontrolled release of radioactivity. Thus for purposes of 10 CFR 50.59, the term "consequences" means dose.

For many licensees, ANSI standards define categories of accidents or malfunctions. For each category a probability (frequency) and a corresponding acceptable consequence is given in terms of barrier loss and radioactivity release. Consequences resulting from accidents and malfunctions are analyzed and documented in the UFSAR and are evaluated against dose acceptance limits that vary depending on the event frequency.

The design effort and the operational controls necessary to ensure the required performance of the physical barriers during anticipated operational occurrences and postulated accidents are extensive. Because 10 CFR 50.59 provides a mechanism for determining if NRC approval is needed for activities affecting plant design and operation, it is helpful to review briefly the requirements and the objectives imposed by the CFR on plant construction and operation. The review will define more clearly the extent of applicability of 10 CFR 50.59.

Appendix A to 10 CFR Part 50 provides General Design Criteria for most nuclear power plants (for pre-Appendix A plants the criteria are in the UFSAR). Section II of Appendix A includes criteria for protection by multiple fission product barriers. The criteria establish requirements for inherent protection, instrumentation and control, reactor coolant pressure boundary and reactor coolant system design, containment design, control rooms, electric power systems, and related inspection and testing. All of these

requirements concentrate on protecting fission product barriers either through inherent or mitigative means.

Section III of Appendix A establishes extensive requirements on reactor protection and reactivity control systems, the objectives again being the protection of fission product barriers. With similar intent, Sections IV, V and VI provide extensive design, inspection, testing, and operational requirements for the quality of the reactor coolant pressure boundary, fluid systems in general, reactor containment, and fuel and radioactivity control. These requirements ensure inherent and engineered protection of the fission product barriers. Introductory statements of Appendix A address the need for consideration of a single failure criterion and redundancy, diversity and separation of mitigation and protection systems. Section I of Appendix A imposes requirements on the quality of implemented protection and the conditions under which these systems must function without loss of capability to perform their safety functions. These conditions include natural phenomena, fire, operational and accident generated environmental conditions.

The implementation of this design philosophy requires extensive accident analyses to define the correct relationship among nominal operating conditions, limiting conditions for operations and limiting safety systems settings in order to prevent safety limits from being exceeded. The UFSAR presents the set of limiting analyses required by NRC. The limiting analyses are utilized to confirm the systems and equipment design, to identify critical setpoints and operator actions, and to support the establishment of technical specifications. Therefore, the results of the UFSAR accident analyses assume functioning of all the equipment (and under the conditions) specified by NRC regulations or requirements. Changes to plant design and operation and conduct of new tests and experiments have the potential to affect the probability and consequences of accidents, to create new accidents and to impact the integrity of fission product barriers. Therefore, these activities are subject to 10 CFR 50.59.

3.0 DEFINITIONS AND APPLICABILITY OF TERMS

The following definitions and terms are discussed in this section:

3.1 10 CFR 50.59 Evaluation

3.2 Accident Previously Evaluated in the FSAR (as updated)

3.3 Change

3.4 Departure from a Method of Evaluation Described in the FSAR (as updated)

3.5 Design Bases (Design Basis)

3.6 Facility as described in the FSAR (as updated)

3.7 Final Safety Analysis Report (as updated)

3.8 Input Parameters

3.9 Malfunction of an SSC Important to Safety

3.10 Methods of Evaluation

3.11 Procedures as described in the FSAR (as updated)

3.12 Safety Analyses

3.13 Screening

3.14 Tests or experiments not described in the FSAR (as updated)

3.1 10 CFR 50.59 EVALUATION

Definition:

A 10 CFR 50.59 evaluation is the documented evaluation against the eight criteria in 10 CFR 50.59(c)(2) to determine if a proposed change, test or experiment requires prior NRC approval via license amendment under 10 CFR 50.90.

Discussion

It is important to establish common terminology for use relative to the 10 CFR 50.59 process. The definitions of *10 CFR 50.59 Evaluation* and *Screening* are intended to clearly distinguish between the process and documentation of licensee screenings and the further evaluation that may be required of proposed activities against the eight criteria in 10 CFR 50.59(c)(2). While many plant activities are subject to a screening, only changes to the facility or procedures described in the UFSAR, and tests or experiments not described in the UFSAR, require evaluation and reporting to NRC under 10 CFR 50.59. Section 4.3 provides guidance for performing 10 CFR 50.59 evaluations. See also Section 3.13 on the definition of "screening."

The phrase "change made under 10 CFR 50.59" (or equivalent) refers to changes subject to the rule (see Section 4.1) that either screened out of the 10 CFR 50.59 process or did not require prior NRC approval based on the results of a 10 CFR 50.59 evaluation. Similarly, the phrases "10 CFR 50.59 applies [to an activity]" or "[an activity] is subject to 10 CFR 50.59" mean that screening, and if necessary, evaluation is required for the activity. The "10 CFR 50.59 process" includes screening, evaluation, documentation and reporting to NRC of activities subject to the rule.

3.2 ACCIDENT PREVIOUSLY EVALUATED IN THE FSAR (AS UPDATED)

Definition:

Accident previously evaluated in the FSAR (as updated) means a design basis accident or event described in the UFSAR including accidents, such as those typically analyzed in Chapters 6 and 15 of the UFSAR, anticipated operational transients, and events the facility is required to withstand such as floods, fires, earthquakes, other external hazards, anticipated transients without scram (ATWS), and station blackout (SBO).

Discussion:

The term "accidents" refers to the anticipated (or abnormal) operational transients and postulated design basis accidents that are analyzed to demonstrate that the facility can be operated without undue risk to the health and safety of the public. The term "accidents" encompasses other events for which the plant is required to cope and which are described in the UFSAR (e.g., turbine missiles, fire, earthquakes and flooding). Note that, although fire is an event for which a plant is required to cope and is described in the UFSAR (by reference to the Fire Hazards Analysis for some licensees), changes to the fire protection program are governed by licensee requirements other than 10 CFR 50.59, as discussed in Section 4.1.5.

Accidents also include new transients or postulated events added to the licensing basis based on new NRC requirements and reflected in the UFSAR pursuant to 10 CFR 50.71(e), e.g., ATWS and SBO.

3.3 CHANGE

Definition:

Change means a modification or addition to, or removal from, the facility or procedures that affects: (1) a design function, (2) method of performing or

controlling the function, or (3) an evaluation that demonstrates that intended functions will be accomplished.

Discussion:

Additions and removals to the facility or procedures can adversely impact the performance of SSCs and the bases for the acceptability of their design and operation. Thus the definition of change includes modifications of an existing provision (e.g., SSC design requirement, analysis method or parameter), additions or removals (physical removals, abandonment, or non-reliance on a system to meet a requirement) to the facility or procedures.

The definitions of "change...," "facility..." (see Section 3.6), and "procedures..." (see Section 3.11) make clear that 10 CFR 50.59 applies to changes to underlying analytical bases for the facility design and operation as well as for changes to SSCs and procedures. Thus 10 CFR 50.59 should be applied to a change being made to an evaluation for demonstrating adequacy of the facility even if no physical change to the facility is involved. Further discussion of the terms in this definition is provided as follows:

Design function means an SSC function that is credited in safety analyses or that supports or impacts an SSC function credited in safety analyses. This may include (1) functions performed by safety-related SSCs or non-safety-related SSCs, and (2) functions of non-safety-related SSCs that, if not performed, would initiate a plant transient or accident. Design functions include the conditions under which intended functions are required to be performed, such as equipment response times, environmental and process conditions, equipment qualification, and single failure.

Method of performing or controlling a function means how a design function is accomplished as credited in the safety analyses, including specific operator actions, procedural step or sequence, or whether a specific function is to be initiated by manual versus automatic means. For example, substituting a manual actuation for automatic would constitute a change to the method of performing or controlling the function.

Evaluation that demonstrates that intended functions will be accomplished means the method(s) used to perform the evaluation (as discussed in Section 3.10). For example, a thermodynamic calculation that demonstrates the ECCS has sufficient heat removal capacity for responding to a postulated accident.

Temporary Changes

Temporary changes to the facility or procedures, such as jumpering terminals, lifting leads, placing temporary lead shielding on pipes and equipment, and use of temporary blocks, bypasses, scaffolding and supports, are made to facilitate a range of plant activities and are subject to 10 CFR 50.59 as follows:

- 10 CFR 50.59 should be applied to temporary changes proposed as compensatory measures to address degraded or non-conforming conditions as discussed in Section 4.4.

- Other temporary changes to the facility or procedures that are not associated with maintenance are subject to 10 CFR 50.59 in the same manner as permanent changes, to determine if prior NRC approval is required. Screening and, as necessary, evaluation of such temporary changes may be considered as part of the screening/evaluation of the proposed permanent change.

Risk impacts of temporary changes associated with maintenance activities should be assessed and managed in accordance with 10 CFR 50.65(a)(4) and associated guidance, as discussed in Section 4.1.2. Applying 10 CFR 50.59 to such activities is not required provided that temporary changes are removed (i.e., affected SSCs must be restored to their normal, as-designed condition) at the conclusion of the maintenance activity. Temporary changes not associated with maintenance are subject to 10 CFR 50.59 as discussed above.

3.4 DEPARTURE FROM A METHOD OF EVALUATION DESCRIBED IN THE FSAR (AS UPDATED)

Definition:

Departure from a method of evaluation described in the FSAR (as updated) means (i) changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same; or (ii) changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application.

Discussion:

The 10 CFR 50.59 definition of "departure ..." provides licensees with flexibility to make changes in methods of evaluation that are "conservative" or that are not important with respect to demonstrating that SSCs can perform their intended design functions. See also the definition and

discussion of "methods of evaluation" in Section 3.10. Guidance for evaluating changes in methods of evaluation under criterion 10 CFR 50.59(c)(2)(viii) is provided in Section 4.3.8.

Conservative vs. Non-Conservative Evaluation Results

Gaining margin by revising an element of a method of evaluation is considered to be a non-conservative change and thus a departure from a method of evaluation for purposes of 10 CFR 50.59. Such departures require prior NRC approval of the revised method. In other words, analytical results obtained by changing any element of a method are "conservative" relative to the previous results, if they are closer to design bases limits or safety analyses limits (e.g., applicable acceptance guidelines). For example, a change in an element of a method of evaluation that changes the result of a containment peak pressure analysis from 45 psig to 48 psig (with design basis limit of 50 psig) would be considered a conservative change for purposes of 10 CFR 50.59(c)(2)(viii). This is because results closer to limiting values are considered conservative in the sense that the new analysis result provides less margin to applicable limits for making future physical or procedure changes without a license amendment.

If use of a modified method of evaluation resulted in a change in calculated containment peak pressure from 45 psig to 40 psig, this would be non-conservative. This is because the change would result in more margin being available (to the design basis limit of 50 psig) for a licensee to make more significant future changes to the physical plant or procedures.

"Essentially the Same"

Licensees may change one or more elements a method of evaluation such that results move in the non-conservative direction without prior NRC approval, provided the results are "essentially the same" as the previous result. Results are "essentially the same" if they are within the margin of error for the type of analysis being performed. Variation in results due to routine analysis sensitivities or calculational differences (e.g., rounding errors and use of different computational platforms) would typically be within the analysis margin of error and thus considered "essentially the same."

"Approved by the NRC for the Intended Application"

Rather than make a minor change to an existing method of evaluation, a licensee may also adopt completely new methodology without prior NRC approval provided the new method is approved by the NRC for the intended application. A new method is "approved by the NRC for the intended application" if it is approved for the type of analysis being conducted and the

licensee satisfies applicable terms and conditions for its use. Specific guidance for making this determination is provided in Section 4.3.8.2.

3.5 DESIGN BASES (DESIGN BASIS)

Definition:

(10 CFR 50.2) Design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or (2) requirements derived from analysis (based on calculations and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.

Discussion

Guidance and examples for identifying 10 CFR 50.2 design bases are provided in Appendix B of NEI 97-04, *Design Bases Program Guidelines*, Revision 1, [Month] 2000.

3.6 FACILITY AS DESCRIBED IN THE FSAR (AS UPDATED)

Definition:

Facility as described in the final safety analysis report (as updated) means:

- The structures, systems, and components (SSC) that are described in the final safety analysis report (FSAR) (as updated),
- The design and performance requirements for such SSCs described in the FSAR (as updated), and
- The evaluations or methods of evaluation included in the FSAR (as updated) for such SSCs which demonstrate that their intended function(s) will be accomplished.

Discussion:

The scope of information that is the focus of 10 CFR 50.59 is the information presented in the original FSAR to satisfy the requirements of 10 CFR 50.34(b), as updated per the requirements of 10 CFR 50.71(e) and as supplemented pursuant to 10 CFR 54.21(d). The definition of "facility as described in the FSAR (as updated)" follows from the requirement of 10 CFR

50.34(b) that the FSAR (and by extension, the UFSAR) contain "a description and analysis of the SSCs of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished."

3.7 FINAL SAFETY ANALYSIS REPORT (AS UPDATED)

Definition:

Final Safety Analysis Report (as updated) means the Final Safety Analysis Report (or Final Hazards Summary Report) submitted in accordance with 10 CFR 50.34, as amended and supplemented, and as updated per the requirements of 10 CFR 50.71(e) or 10 CFR 50.71(f), as applicable.

Discussion:

The scope of the UFSAR includes its text, tables, diagrams, etc., as well as supplemental information explicitly incorporated by reference. References that are merely listed in the UFSAR and documents that are not explicitly incorporated by reference are not considered part of the UFSAR and therefore are not subject to control under 10 CFR 50.59.

Per 10 CFR 50.59(c)(4), licensees are not required to apply 10 CFR 50.59 to UFSAR information that is subject to other specific change control regulations. For example, licensee Quality Assurance Programs, Emergency Plans and Security Plans are controlled by 10 CFR 50.54(a), (p) and (q), respectively.

Per 10 CFR 50.59(c)(3), the "FSAR (as updated)," for purposes of 10 CFR 50.59, also includes UFSAR update pages approved by the licensee for incorporation in the UFSAR since the last required update was submitted per 10 CFR 50.71(e). The intent of this requirement is to ensure that decisions about proposed activities are made with the most complete and accurate information available. Pending UFSAR revisions may be relevant to a future activity that involves that part of the UFSAR. Therefore, pending UFSAR revisions to reflect completed activities that have received final approval for incorporation in the next required update should be considered as part of the UFSAR for purposes of 10 CFR 50.59 screenings and evaluations, as appropriate. Appropriate configuration management mechanisms should be in place to identify and assess interactions between concurrent changes affecting the same SSCs or the same portion of the UFSAR.

Guidance on the required content of UFSAR updates is provided in Regulatory Guide 1.181 and NEI 98-03, Revision 1, *Guidelines for Updating FSARs*, June 1999.

3.8 INPUT PARAMETERS

Definition:

Input parameters are those values derived directly from the physical characteristics of SSC or processes in the plant, including flow rates, temperatures, pressures, dimensions or measurements (e.g., volume, weight, size, etc), and system response times.

Discussion:

The principal intent of this definition is to distinguish methods of evaluation from evaluation input parameters. Changes to methods of evaluation described in the UFSAR (see Section 3.10) are evaluated under criterion 10 CFR 50.59(c)(2)(viii), whereas changes to input parameters described in the FSAR are considered changes to the facility that would be evaluated under the other seven criteria of 10 CFR 50.59(c)(2), but not criterion (c)(2)(viii).

If a methodology permits the licensee to establish the value of an input parameter on the basis of plant-specific considerations, then that value is an input to the methodology, not part of the methodology. On the other hand, an input parameter is considered to be an element of the methodology if:

- The method of evaluation includes a methodology describing how to select the value of an input parameter to yield adequately conservative results. However, if a licensee opts to use a value more conservative than that required by the selection method, reduction in that conservatism should be evaluated as an input parameter change, not a change in methodology.
- The development or approval of a methodology was predicated on the degree of conservatism in a particular input parameter or set of input parameters. In other words, if certain elements of a methodology or model were accepted on the basis of the conservatism of a selected input value, then that input value is considered an element of the methodology.

Examples illustrating the treatment of input parameters are provided in Section 4.2.1.3.

Section 4.3.8 provides guidance and examples to describe the specific elements of evaluation methodology that would require evaluation under 10 CFR 50.59(c)(2)(viii) and to clearly distinguish these from specific types of input parameters that are controlled by the other seven criteria of 10 CFR 50.59(c)(2).

3.9 MALFUNCTION OF AN SSC IMPORTANT TO SAFETY

Definition:

Malfunction of SSCs important to safety means the failure of SSCs to perform their intended design functions described in the UFSAR (whether or not classified as safety-related in accordance with 10 CFR 50, Appendix B).

Discussion:

Guidance and examples for applying this definition is provided in Section 4.3.

3.10 METHODS OF EVALUATION

Definition:

Methods of evaluation means the calculational framework used for evaluating behavior or response of the facility or an SSC.

Discussion:

Examples of methods of evaluation are presented below. Changes to such methods of evaluation require evaluation under 10 CFR 50.59(c)(2)(viii) only for evaluations used either in UFSAR safety analyses or in establishing the design bases, and only if the methods are described, outlined or summarized in the UFSAR. Methodology changes that are subject to 10 CFR 50.59 include changes to elements of existing methods described in the UFSAR and to changes that involve replacement of existing methods of evaluation with alternative methodologies.

Elements of Methodology	Example
■ Data correlations	■ DNBR correlations
■ Means of data reduction	■ ASME III and Appendix G methods for evaluating reactor vessel embrittlement specimens
■ Physical constants or coefficients	■ Heat transfer coefficients
■ Mathematical models	■ Decay heat models
■ Specific limitations of a computer program	■ No voiding in PWR hot legs for non-LOCA analyses
■ Specified factors to account for uncertainty in measurements or data	■ 120% of 1971 decay heat model
■ Statistical treatment of results	■ Vendor-specific thermal design procedure
■ Dose conversion factors and assumed source term(s)	■ ICRP factors

Methods of evaluation described in the UFSAR subject to criterion 10 CFR 50.59(c)(2)(viii) are:

- Methods of evaluation used in analyses that demonstrate that design basis limits of fission product barriers are met (i.e., for the parameters subject to criterion 10 CFR 50.59(c)(2)(vii))

- Methods of evaluation used in UFSAR safety analyses, including containment, ECCS and accident analyses typically presented in UFSAR Chapters 6 and 15, to demonstrate that consequences of accidents do not exceed 10 CFR 100 or 10 CFR 50, Appendix A, dose limits.

- Methods of evaluation used in supporting UFSAR analyses that demonstrate intended design functions will be accomplished under design basis conditions that the plant is required to withstand, including natural phenomena, environmental conditions, dynamic effects, station blackout, and ATWS.

3.11 PROCEDURES AS DESCRIBED IN THE FSAR (AS UPDATED)

Definition:

Procedures as described in the final safety analysis report (as updated) means those procedures that contain information described in the FSAR (as updated) such as how structures, systems, and components are operated and controlled (including assumed operator actions and response times).

Discussion:

The scope of information that is the focus of 10 CFR 50.59 is the information presented in the original FSAR to satisfy the requirements of 10 CFR 50.34(b), as updated per the requirements of 10 CFR 50.71(e) and as supplemented pursuant to 10 CFR 54.21(d).

For purposes of 10 CFR 50.59, "procedures" are not limited to plant procedures specifically identified in the UFSAR (e.g., operating, chemistry, system, test, surveillance, and emergency procedures). Procedures include UFSAR descriptions of how actions related to system operation are to be performed and controls over the performance of design functions. This includes UFSAR descriptions of operator action sequencing or response times, certain descriptions (text or figure) of SSC operation and operating modes, operational and radiological controls, inspection and testing frequency, and similar information. If changes to these activities or controls are made, such changes are considered changes to procedures described in the UFSAR, and the changes are subject to 10 CFR 50.59.

Even if described in the UFSAR, procedures for performing maintenance, work control, and administrative activities are normally outside the definition of "procedures as described in the UFSAR" because they do not typically contain information on how SSCs are operated or controlled. Section 4.1.4 identifies examples of procedures that are not subject to 10 CFR 50.59.

10 CFR 50.59 screening of procedures is discussed in Section 4.2.1.2.

3.12 SAFETY ANALYSES

Definition:

Safety analyses are analyses performed pursuant to NRC requirement to demonstrate the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in 10 CFR 50.34(a)(1) or 10 CFR 100.11. Safety analyses are required to be presented in the UFSAR per 10 CFR 50.34(b) and 10 CFR 50.71(e) and include, but are not limited to, the accident analyses typically presented in Chapter 15 of the UFSAR.

Discussion:

Safety analyses are those analyses or evaluations that demonstrate that acceptance criteria for the facility's capability to withstand or respond to postulated events are met. Containment, ECCS, and accident analyses typically presented in Chapters 6 and 15 of the UFSAR clearly fall within the meaning of "safety analyses" as defined above. Also within the meaning of this definition are:

- Supporting UFSAR analyses that demonstrate that SSC design functions will be accomplished as credited in the accident analyses
- UFSAR analyses of events that the facility is required to withstand such as turbine missiles, fires, floods, earthquakes, station blackout, and ATWS.

Note that, although fire is an event which a plant is required to withstand and for which it has been analyzed accordingly in the UFSAR (by reference to the Fire Hazards Analysis for some licensees), changes to the fire protection program and associated analyses are governed by licensee requirements other than 10 CFR 50.59, as discussed in Section 4.1.5.

3.13 SCREENING

Definition:

Screening is the process for determining whether a proposed activity requires a 10 CFR 50.59 evaluation to be performed.

Discussion:

Screening is that part of the 10 CFR 50.59 process that determines whether a 10 CFR 50.59 evaluation is required prior to implementing a proposed activity.

The definitions of "change," "facility as described...," "procedures as described...," and "test or experiment not described..." constitute criteria for the 10 CFR 50.59 screening process. Activities that do not meet these criteria are said to "screen out" from further review under 10 CFR 50.59, i.e., may be implemented without a 10 CFR 50.59 evaluation.

Engineering and technical information concerning a proposed activity may be used along with other information as basis for determining if the activity screens out or requires a 10 CFR 50.59 evaluation.

Further discussion and guidance on screening is provided in Section 4.2.

3.14 TESTS OR EXPERIMENTS NOT DESCRIBED IN THE FSAR (AS UPDATED)

Definition:

Tests or experiments not described in the final safety analysis report (as updated) means any activity where any structure, system, or component is utilized or controlled in a manner which is either:

- Outside the reference bounds of the design bases as described in the UFSAR, or
- Inconsistent with the analyses or descriptions in the UFSAR.

Discussion:

10 CFR 50.59 must be applied to tests or experiments not described in the UFSAR. The intent of the definition is to ensure that tests or experiments that put the facility in a situation that has not previously been evaluated (e.g., unanalyzed system alignments) or that could affect the capability of SSCs to perform their intended design functions (e.g., high flow rates, high temperatures) are evaluated before they are conducted to determine if prior NRC approval is required.

Post-modification testing should be evaluated as a test under 10 CFR 50.59 only if an abnormal mode of operation is proposed that is not described in the UFSAR. Post-modification testing may be considered as part of the 10 CFR 50.59 evaluation for the modification itself.

4 IMPLEMENTATION GUIDANCE

Licensees may determine applicability and screen activities to determine if 10 CFR 50.59 evaluations are required as described in Sections 4.1 and 4.2, or equivalent manner.

4.1 APPLICABILITY

As stated in Section (b) of 10 CFR 50.59, the rule applies to each holder of a license authorizing operation of a production or utilization facility, including the holder of a license authorizing operation of a nuclear power reactor that has submitted a certification of permanent cessation of operations required under 10 CFR 50.82(a)(1) or a reactor licensee whose license has been amended to allow possession but not operation of the facility.

4.1.1 Applicability to Licensee Activities

10 CFR 50.59 is applicable to tests or experiments not described in the UFSAR and to changes to the facility or procedures as described in the UFSAR, including changes made in response to new requirements or generic communications, except as noted below:

- Per 10 CFR 50.59(c)(1)(i), proposed activities that require a change to the technical specifications must be made via the license amendment process, 10 CFR 50.90. Aspects of proposed activities that are not directly related to the required technical specification change are subject to 10 CFR 50.59.

- To reduce duplication of effort, 10 CFR 50.59(c)(4) specifically excludes from the scope of 10 CFR 50.59 changes to the facility or procedures that are controlled by other more specific requirements and criteria established by regulation. For example, 10 CFR 50.54 which was promulgated after 10 CFR 50.59, specifies criteria and reporting requirements for changing quality assurance, physical security and emergency plans.

Activities controlled and implemented under other regulations may require related information in the UFSAR to be updated. To the extent the UFSAR changes are directly related to the activity implemented via another regulation, applying 10 CFR 50.59 is not required. UFSAR changes should be identified to NRC as part of the required UFSAR update, per 10 CFR 50.71(e). However, there may be certain activities for which a licensee would need to apply both the requirements of 10 CFR 50.59 and that of another regulation. For example, a modification to a facility involves additional components and substantial piping reconfigurations as well as changes to

protection system setpoints. The protection system setpoints are contained in the facility technical specifications. Thus, a license amendment to revise the technical specifications under 10 CFR 50.90 is required to implement the new system setpoints. 10 CFR 50.59 should be applied to the balance of the modification, including impacts on required operator actions.

4.1.2 Maintenance Activities

Maintenance activities are activities that restore SSCs to their as-designed condition, including activities that implement approved design changes. Maintenance activities include troubleshooting, calibration, refurbishment, post-maintenance testing, identical replacements, housekeeping, and similar activities that do not permanently alter the design or design function of SSCs, and are thus not subject to 10 CFR 50.59.

Licensees should address operability in accordance with the technical specifications and assess/manage the risk impact of maintenance activities per 10 CFR 50.65(a)(4) and NEI 93-01, *Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.*

In addition to assessments required by 10 CFR 50.65(a)(4), 10 CFR 50.59 should also be applied to maintenance activities if the design is not restored to its original condition as a result of the maintenance activity (e.g., if SSCs are removed; if the design, design function or operation is altered; or if a temporary change in support of the maintenance is not removed).

10 CFR 50.59 should be applied to temporary changes proposed as compensatory measures for degraded or non-conforming conditions, as discussed in Section 4.4.

4.1.3 UFSAR Modifications

Per NEI 98-03 (Revision 1, June 1999), as endorsed by Regulatory Guide 1.181 (September 1999), modifications to the UFSAR that are not the result of activities performed under 10 CFR 50.59 are not subject to control under 10 CFR 50.59. Such modifications include reformatting and simplification of UFSAR information and removal of obsolete or redundant information and excessive detail.

Similarly, 10 CFR 50.59 need not be applied to the following types of activities:

- Editorial changes to the UFSAR
- Clarifications to improve reader understanding

- Correction of inconsistencies within the UFSAR (e.g., between sections)
- Minor corrections to drawings, e.g., correcting mislabeled valves
- Similar changes to UFSAR information that do not change the meaning or substance of information presented

4.1.4 Changes to Procedures Governing the Conduct of Operations

Even if described in the UFSAR, changes to managerial and administrative procedures governing the conduct of facility operations are controlled under 10 CFR 50, Appendix B, programs and are not subject to control under 10 CFR 50.59. These include, but are not limited to, procedures in the following areas:

- Operations and maintenance activities such as control of equipment status (tag outs)
- Shift staffing and personnel qualifications
- Changes to position titles when no UFSAR-described organizational responsibilities or relationships are changed
- Control of plant procedures
- Training programs
- On-site/off-site safety review committees
- Plant modification process
- Calculation process

4.1.5 Changes to Approved Fire Protection Programs

Most nuclear power plant licenses contain a section on fire protection. Originally, these fire protection license conditions varied widely in scope and content. These variations created problems for licensees and for NRC inspectors in identifying the operative and enforceable fire protection requirements at each facility.

To resolve these problems, the NRC promulgated guidance in Generic Letter 86-10, "Implementation of Fire Protection Requirements," for licensees to:

- Incorporate the fire protection program and major commitments into the FSAR for the facility, and

- Amend the operating license to substitute a standard fire protection license condition for the previous license condition(s) regarding fire protection.

Under the standard fire protection license condition, licensees may

(1) Make changes to their approved FP programs without prior NRC approval provided that the changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire, and

(2) Alter specific features of the approved program provided such changes do not otherwise involve a change to the license or technical specifications, or require an exemption.

Adoption of the standard fire protection license condition provided a more consistent approach to evaluating changes to the facility, including those associated with the fire protection program. Originally, changes to the FP program under the FP license condition were also subject to 10 CFR 50.59; however, this created confusion as to which regulatory requirement governed FP program changes.

The focus on allowing licensees to make changes that maintain the post-fire safe shutdown capability of a FP program change is analogous to permitting changes with "minimal" effects under 10 CFR 50.59, and is consistent with the 10 CFR 50.59 rulemaking objectives to reduce regulatory burden and more effectively focus licensee and NRC resources on safety significant issues. Fire protection program changes that do not adversely affect post-fire safe shutdown capability do not warrant prior NRC review and approval. Therefore, also applying 10 CFR 50.59 to fire protection program changes is redundant and not necessary because the standard fire protection license condition establishes the appropriate regulatory framework and acceptance criteria for determining when proposed changes require prior NRC approval.

Changes to the fire protection program should be evaluated for impacts on other design functions, and 10 CFR 50.59 should be applied to the non-fire protection related effects of the change, if any.

As with previous fire protection program changes made under the design and configuration control process, licensees are required to maintain, in auditable form, a current record of all such changes, including analysis of the effects of the change on the fire protection program, and shall make those records available to NRC inspectors upon request. All changes to the approved program which result in changes to the UFSAR (including the fire hazards analysis incorporated in the UFSAR) should be reported to the NRC in accordance with 10 CFR 50.71(e).

4.2 SCREENING

Once it has been determined that 10 CFR 50.59 is applicable to a proposed activity, screening is performed to determine if the activity should be evaluated against the evaluation criteria of 10 CFR 50.59(c)(2).

Engineering, design and other technical information concerning the activity and affected SSCs should be used to assess whether the activity is a test or experiment not described in the UFSAR or a modification, addition or removal (i.e., change) that affects:

- A design function of an SSC
- A method of performing or controlling the design function, or
- An evaluation for demonstrating that intended design functions will be accomplished

Sections 4.2.1 and 4.2.2 provide guidance and examples for determining whether an activity is (1) a change to the facility or procedures as described in the UFSAR or (2) a test or experiment not described in the UFSAR. If an activity is determined to be neither, then it screens out and may be implemented without further evaluation. Activities that are screened out from further evaluation under 10 CFR 50.59 should be documented as discussed in Section 4.2.3.

Activities that screen out may nonetheless require UFSAR information to be updated. Licensees should provide updated UFSAR information to the NRC in accordance with 10 CFR50.71(e).

Specific guidance for applying 10 CFR 50.59 to temporary changes proposed as compensatory measures for degraded or non-conforming conditions is provided in Section 4.4.

4.2.1 Is the Activity a Change to the Facility or Procedures as Described in the UFSAR?

Per the definition of "change" discussed in Section 3.3, 10 CFR 50.59 is applicable to additions as well as to changes to and removals from the facility or procedures. Additions should be screened for their effects on the existing facility and procedures as described in the UFSAR and, if required, a 10 CFR 50.59 evaluation should be performed. NEI 98-03 provides guidance for determining whether additions to the facility and procedures should be reflected in the UFSAR per 10 CFR 50.71(e).

The following may be appropriate to consider when determining, based on supporting technical/engineering information, if a proposed activity is a

"change to the facility or procedures as described in the UFSAR" that requires further evaluation under 10 CFR 50.59:

- Does the activity affect an SSC design function credited in the safety analyses or a supporting SSC design function?
- Does the activity affect the reliability of the SSC design function?
- Does the activity reduce existing redundancy, diversity or defense-in-depth?
- Does the activity add or delete an automatic or manual design function of the SSC?
- Does the activity convert a feature that was automatic to manual or vice versa?
- Does the activity introduce an unwanted or previously unreviewed system interaction?
- Does the activity affect the ability or response time to perform required actions, e.g., alter equipment access or add steps necessary for performing tasks?
- Does the activity alter the seismic or environmental qualification of the SSC?
- Does the activity affect other units at a multiple unit site?
- Does the activity use equipment/tools that interface either directly or indirectly with an operable SSC?
- Does the activity introduce intrusive test equipment into the SSC such that an SSC design function is affected?

4.2.1.1 Screening of Changes to the Facility as Described in the UFSAR

Screening to determine that a 10 CFR 50.59 evaluation is required is straightforward when a change affects an SSC design function, method of performing or controlling a design function, or evaluation that demonstrates intended design functions will be accomplished as described in the UFSAR.

However, a facility also contains many SSCs not described in the UFSAR. These can be components, subcomponents of larger components or even entire systems. Changes to SSCs that are not explicitly described in the UFSAR can have the potential to affect SSCs that are described and thus may require a 10 CFR 50.59 evaluation. In such cases, the approach for determining whether a change involves a change to the facility as described in the UFSAR, is to consider the larger, UFSAR-described SSC of which the SSC being modified is a part. If for the larger SSC, the change affects a UFSAR-described design function, method of performing or controlling the design function, or an evaluation demonstrating that intended design functions will be accomplished, then a 10 CFR 50.59 evaluation is required.

Another important consideration is that a change to non safety-related SSCs not described in the UFSAR can indirectly affect the capability of SSCs to perform their UFSAR-described design function(s). For example, increasing the heat load on a non safety-related heat exchanger could compromise the cooling system's ability to cool safety-related equipment.

Seismic qualification, missile protection, flooding protection, fire protection, environmental qualification, high energy line break and masonry block walls are some of the areas where changes to non safety-related SSCs, whether or not described in the UFSAR, can affect the UFSAR-described design function of SSCs through indirect or secondary effects.

Equivalent replacement is a type of change to the facility that does not alter the design functions of SSCs. Licensee equivalence assessments, e.g., consideration of performance/operating characteristics and other factors, may thus form the basis for screening determinations that no 10 CFR 50.59 evaluation is required.

The following examples illustrate the 10 CFR 50.59 screening process as applied to proposed facility changes:

- A licensee proposes to replace a relay in the overspeed trip circuit of an emergency diesel generator with a non-equivalent relay. The relay is not described in the UFSAR, but the design functions of the overspeed trip circuit and the emergency diesel generator are. Based on engineering/technical information supporting the change, the licensee determines if replacing the relay would affect the design function of either the overspeed trip circuit or EDG. If the licensee concludes that the change would not affect the UFSAR-described design function of the circuit or EDG, then this determination would form the basis for screening out the change, and no 10 CFR 50.59 evaluation would be required.

- A licensee proposes a non-equivalent change to the operator on one of the safety injection accumulator isolation valves. The UFSAR describes that these isolation valves are open with their circuit breakers open during normal operation. These are motor operated, safety related valves required for pressure boundary integrity and to remain open so that flow to the RCS will occur during a LOCA as pressure drops below ~600 psi. They are remotely operated so that they can be closed during a normal shutdown and not inject when not required. This change would screen out because the change affects only the valve operator—not the UFSAR-described design function (pressure boundary integrity) that supports safety injection performance credited in the safety analyses.

- A licensee proposes to replace a globe valve with a ball valve in a vent/drain application to reduce the propensity of this valve to leak. The UFSAR-described design function of this valve is to maintain the integrity of the system boundary when closed. The vent/drain function of the valve does not relate to design functions credited in the safety analyses, and the licensee has determined that a ball valve is adequate to support the vent/drain function and is superior to the globe valve in terms of its isolation function. Thus the proposed change affects the design of the existing vent/drain valve—not the design function that supports system performance credited in the safety analyses—and evaluation/reporting under 10 CFR 50.59 is not required. The screening determination should be documented, and the UFSAR should be updated per 10 CFR 50.71(e) to reflect the change.

- The bolts for retaining a rupture disk are being replaced with bolts of a different material and fewer threads, but equivalent load capacity and strength, such that the rupture disk will still relieve at the same pressure as before the change. Because the replacement bolts are equivalent in function to the original bolts and the rupture disk continues to meet the same functional requirements, this activity may be screened out as an equivalent change.

4.2.1.2 Screening of Changes to Procedures as Described in the UFSAR

Changes to procedures are "screened in" (i.e., require a 10 CFR 50.59 evaluation) if the change affects how SSC design functions are performed or controlled, as described in the UFSAR (including assumed operator actions and response times). Changes to a procedure that does not affect how SSC design functions described in the UFSAR are performed or controlled would screen out. The following examples illustrate the 10 CFR 50.59 screening process as applied to proposed procedure changes:

- Emergency Operating Procedures include operator actions and response times associated with response to design basis events, which are described in the UFSAR, but also address operator actions for severe accident scenarios that are outside the design basis and not described in the UFSAR. A change would screen out at this step if the change was to those procedures or parts of procedures dealing with operator actions during severe accidents.

- If the UFSAR description of the reactor startup procedure contains eight fundamental sequences, the licensee's decision to eliminate one of the sequences would screen in. On the other hand, if the licensee consolidated the eight fundamental sequences and did not affect the method of controlling or performing reactor startup, the change would screen out.

- The UFSAR states that a particular flow path is isolated by a locked closed valve when not in use. A procedure change to remove the lock from this valve such that it becomes a normally closed valve would screen in as a change to procedures described in the UFSAR. In this case, the design function is to remain closed and the method of performing the design function has changed from locked closed to administratively closed. Thus this change would screen in and require a 10 CFR 50.59 evaluation to be performed.

- Operations proposes to revise its procedures to change from 8-hour shifts to 12-hour shifts. This change results in mid-shift rounds being conducted every 6 hours as opposed to every 4 hours. The UFSAR describes high energy line breaks including mitigation criteria. Operator action to detect and terminate the line break is described in the UFSAR which specifically states that 4 hours is assumed for the pipe break to go undetected before it would be identified during operator mid-shift rounds. The change from 4 to 6 hour rounds is a change to a procedure as described in the UFSAR because it affects the timing of operator actions credited in the safety analyses for limiting the effects of high energy line breaks. Therefore, this change screens in, and a 10 CFR 50.59 evaluation is required.

- The UFSAR states that station batteries are tested in accordance with IEEE 450-1995, describes the testing frequency, and lists the title and designation of the plant surveillance procedure. Battery test method and frequency is thus a procedure described in the UFSAR related to the design function of station batteries to supply power to SSCs upon loss of AC power. Revisions to the battery test procedure could affect the reliability of station batteries to perform their design function. Changes that deviate from the existing test frequency or IEEE 450-

1995 methods screen in and would require a 10 CFR 50.59 evaluation. Listing of the procedure title and designation does not mean that all revisions to the procedure are "changes to procedures described in the UFSAR."

- The UFSAR states that the Shift Supervisor will authorize all radioactive liquid releases. Assigning this function to another individual would not require a 10 CFR 50.59 evaluation because the change does not involve performance or control of design functions credited in the safety analyses. The licensee would be required to reflect the change in the next required update of the UFSAR, per 10 CFR 50.71(e).

4.2.1.3 Screening Changes to UFSAR Methods of Evaluation

As discussed in Section 3.6, methods of evaluation included in the UFSAR to demonstrate that intended SSC design functions will be accomplished are considered part of the "facility as described in the UFSAR." Thus use of new or revised methods of evaluation (as defined in Section 3.10) is considered to be a change that is controlled by 10 CFR 50.59 and needs to be considered as part of this screening step. Changing elements of a method of evaluation included in the UFSAR, or use of an alternative method, must be evaluated under 10 CFR 50.59(c)(2)(viii) to determine if prior NRC approval is required (see Section 4.3.8). Changes to methods of evaluation (only) do not require evaluation against the first seven criteria.

Changes to methods of evaluation not included in the UFSAR or to methodologies included in the UFSAR that are not used in the safety analyses or to establish design bases would screen out at this step.

Methods of evaluation that may be identified in references listed at the end of UFSAR sections or chapters are not subject to control under10 CFR 50.59 unless the UFSAR states they were used for specific analyses within the scope of 10 CFR 50.59(c)(2)(viii).

Changes to methods of evaluation included in the UFSAR do not require evaluation under 10 CFR 50.59 if the changes are within the constraints and limitations associated with use of the method, e.g., identified in a topical report and/or SER.

The following examples illustrate the screening of changes to methods of evaluation:

- The UFSAR identifies the name of the computer code used for performing containment performance analyses, with no further

discussion of the methods employed within the code for performing those analyses. Changes to the computer code may be screened out provided that the changes are within the constraints and limitations identified in the associated topical report and SER. A change that goes beyond restrictions on the use of the method should be evaluated under 10 CFR 50.59(c)(2)(viii) to determine if prior NRC approval is required.

- The UFSAR describes the methods used for atmospheric heat transfer and containment pressure response calculations contained within the CONTEMPT computer code. The code is also used for developing long term temperature profiles (post-recirculation phase of LOCA) for environmental qualification through modeling of the residual heat removal system. Neither this application of the code nor the analysis method is discussed in the UFSAR. A revision to CONTEMPT to incorporate more dynamic modeling of the residual heat removal system transfer of heat to the ultimate heat sink would screen out because this application of the code is not described in the UFSAR as being used in the safety analyses or to establish design bases. Any changes to CONTEMPT that affect the atmospheric heat transfer or containment pressure predictions would not screen out (because the UFSAR describes this application in the safety analyses), and would require a 10 CFR 50.59 evaluation.

- The steamline break mass and energy release calculations were originally performed at a power level of 105% of the nominal power (plus uncertainties) in order to allow margin for a future power uprate. The utility later decided that it would not pursue the power uprate and wished to use the margin to address other equipment qualification issues. The steamline break mass and energy release calculations were re-analyzed, using the same methodology, at 100% power (plus uncertainties). This change would screen out as a methodology change because the proposed activity involved a change to an input parameter (% power) and not a methodology change. This change should be screened per Section 4.2.1.1 to determine if it constitutes a change to the facility as described in the UFSAR that requires evaluation under 10 CFR 50.59(c)(2)(i-vii).

- The LOCA mass and energy release calculations were originally performed at a power level of 105% of the nominal power, plus uncertainties. Some of the assumptions in the analysis were identified as non-conservative, but the NRC concluded in the associated SER that the overall analysis was conservative because of the use of the higher initial power. The utility later decided that it would not pursue the power up-rate and wished to use the margin to address other equipment qualification issues. The LOCA break mass and

energy release calculations were re-analyzed, using the same methodology, at 100% power (plus uncertainties). This change would not screen out because the proposed activity involved a change to an input parameter that was integral to the NRC approval of the methodology.

- Due to fuel management changes, core physics parameters change for a particular reload cycle. The topical report and associated SER that describe how the core physics parameters are to be calculated explicitly allow use of either 2-D or 3-D modeling for the analysis. A change to add or remove discretionary conservatism via use of 3-D methods instead of 2-D methods or vice-versa would screen out because the change is within the terms and conditions of the SER.

4.2.2 Is the Activity a Test or Experiment Not Described in the UFSAR?

As discussed in Section 3.14, tests or experiments not described in the UFSAR are activities where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or inconsistent with analyses or description in the UFSAR.

Tests and experiments that are described in the UFSAR may be screened out at this step. Tests and experiments that are not described in the UFSAR may be screened out provided the test or experiment is bounded by tests and experiments that are described. Similarly, tests and experiments not described in the UFSAR may be screened out provided that affected SSCs will be appropriately isolated from the facility.

Examples of tests that would "screen in" at this step (assuming they were not described in the UFSAR) would be:

- For BWRs, hydrogen injection into the reactor coolant system to minimize stress corrosion cracking.
- For BWRs, zinc injection into the reactor coolant system to reduce activation.
- For PWRs, ECCS flow tests that affect the ability to remove decay heat.
- Operation with fuel demonstration assemblies.

Examples of tests that would "screen out" would be:

- Steam generator moisture carryover tests (provided such testing is described in the UFSAR)
- Balance-of-plant heat balance test
- Information gathering that is non-intrusive to the operation or function of the associated SSC

4.2.3 Screening Documentation

10 CFR 50.59 recordkeeping requirements apply to 10 CFR 50.59 evaluations performed for activities that screened in, not to screening records for activities that screened out. However, documentation should be maintained in accordance with plant procedures of screenings that conclude a proposed activity screened out (i.e., that a 10 CFR 50.59 evaluation was not required). The basis for the conclusion should be documented to a degree commensurate with the safety significance of the change. Typically, the screening documentation is retained as part of the change package. This documentation does not constitute the record of changes required by 10 CFR 50.59, and thus is not subject to 10 CFR 50.59 documentation and reporting requirements. Screening records need not be retained for activities for which a 10 CFR 50.59 evaluation was performed or for activities that were never implemented.

4.3 EVALUATION PROCESS

Once it has been determined that a given activity requires a 10 CFR 50.59 evaluation, the written evaluation must address the applicable criteria of 10 CFR 50.59(c)(2). These eight criteria are used to evaluate the effects of proposed activities on accidents and malfunctions previously evaluated in the UFSAR and their potential to cause accidents or malfunctions whose effects are not bounded by previous analyses.

Criteria (c)(2)(i—vii) are applicable to activities other than changes in methods of evaluation. Criterion (c)(2)(viii) is applicable to changes in methods of evaluation. If any of these criteria are met, the licensee must apply for and obtain a license amendment per 10 CFR 50.90 prior to implementing the activity. The evaluation against each criterion should be appropriately documented as discussed in Section 4.5. Subsections 4.3.1 through 4.3.8 provide guidance and examples for evaluating proposed activities against the eight criteria.

Each element of a proposed activity must undergo a 10 CFR 50.59 evaluation, except in instances where linking elements of an activity is

appropriate, in which case the linked elements can be evaluated together. A test for linking elements of proposed changes is interdependence.

It is appropriate for discrete elements to be evaluated together if (1) they are interdependent as in the case where a modification to a system or component necessitates additional changes to other systems or procedures; or (2) they are performed collectively to address a design or operational issue. For example, a pump upgrade modification may also necessitate a change to a support system, such as cooling water.

If concurrent changes are being made which are not linked, each must be evaluated separately and independently of each other.

The effects of a proposed activity being evaluated under 10 CFR 50.59 should be assessed against each of the evaluation criteria separately. For example, an increase in frequency/likelihood of occurrence cannot be compensated for by additional mitigation of consequences.

Specific guidance for applying 10 CFR 50.59 to temporary changes proposed as compensatory measures for degraded or non-conforming conditions is provided in Section 4.4.

4.3.1 Does the Activity Result in More than a Minimal Increase in the Frequency of Occurrence of an Accident?

In answering this question, the first step is to identify the accidents that have been evaluated in the UFSAR that are affected by the proposed activity. Then a determination should be made as to whether the frequency of these accidents occurring would be more than minimally increased.

Accidents and transients have been divided into categories based upon a qualitative assessment of frequency. For example, ANSI standards define the following categories for plant conditions for most PWRs as follows:

- Normal Operations - Expected frequently or regularly in the course of power operation, refueling, maintenance or maneuvering.
- Incidents of Moderate Frequency - Any one incident expected per plant during a calendar year.
- Infrequent Incidents - Any one incident expected per plant during plant lifetime.
- Limiting Faults - Not expected to occur but could release significant amounts of radioactive material thus requiring protection by

design.

ANSI standards for BWRs have slightly different but equivalent definitions.

During initial plant licensing, accidents were assessed in relative frequencies, as described above. Minimal increases in frequency resulting from subsequent licensee activities do not significantly change the licensing basis of the facility and do not impact the conclusions reached about acceptability of the facility design.

Since accident and transient frequencies were considered in a broad sense as described above, a change from one frequency category to a more frequent category is clearly an example of a change that results in more than a minimal increase in the frequency of occurrence of an accident. Changes within a category could also result in more than a minimal increase in the frequency of occurrence of an accident. Normally, the determination of a frequency increase is based upon a qualitative assessment using engineering evaluations consistent with the UFSAR analysis assumptions. However, a plant-specific accident frequency calculation or PRA may be used to evaluate a proposed activity in a quantitative sense. It should be emphasized that PRAs are just one of the tools for evaluating the effect of proposed activities, and their use is not required to perform 10 CFR 50.59 evaluations. In general, frequencies of accidents considered to be credible are nominally greater than 1E-7 per year of reactor operation (e.g., tornado-generated missiles, aircraft hazards, etc.). In the event that the change in frequency of an accident is calculated, the result is considered to be not more than a minimal increase in the frequency of occurrence provided (1) the increase in the pre-change accident or transient frequency is less than 10 percent,[1] or (2) the resultant frequency of occurrence remains below 1E-6 or applicable regulatory threshold.

Reasonable engineering practices, engineering judgment, and PRA techniques, as appropriate, should be used in determining whether the frequency of occurrence of an accident would more than minimally increase as a result of implementing a proposed activity. A large body of knowledge has been developed in the area of accident frequency and risk significant sequences through plant-specific and generic studies. This knowledge, where applicable, should be used in determining what constitutes more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR. The effect of a proposed activity on the frequency of an accident must be discernable and attributable to the proposed activity in order to exceed the more than minimal increase standard. A proposed

1 The proposed 10 percent increase threshold is consistent with the NRC report, "Options for Incorporating Risk Insights into 10 CFR 50.59 Process," December 17, 1998, Section 6.4.1.

activity is considered to have a negligible effect on the frequency of occurrence of an accident when the change in frequency is so small or the uncertainties in determining whether a change in frequency has occurred are such that it cannot be reasonably concluded that the frequency has actually changed (i.e., there is no clear trend towards increasing the frequency). A proposed activity that has a negligible effect satisfies the minimal increase standard.

The following considerations may be useful in making this determination:

a) Will the proposed activity meet the design, material, and construction standards applicable to the SSC being modified? If the answer is "yes", this aspect of the proposed activity is judged not to be more than a minimal increase in the frequency of occurrence of an accident. If the answer is "no," then either a justification for saying there is not more than a minimal increase in the frequency of an accident occurring should be provided, or it should be concluded that the frequency of an accident occurring would more than minimally increase.

b) Will the proposed activity affect overall system performance in a manner that could more than minimally increase the frequency of occurrence of an accident? Typical considerations include:

   (1) Will the proposed activity use instrumentation with accuracies or response characteristics that are different than existing instrumentation such that an accident is more likely to occur?

   (2) Will the proposed activity cause systems to be operated outside of their current design or testing limits (e.g., imposing additional loads on electrical systems, operating a piping system at higher than normal pressure, operating a motor outside of its rated voltage and amperage, etc.)?

   (3) Will the proposed activity cause system vibration or water hammer, fatigue, corrosion, thermal cycling or degradation of the environment of equipment important to safety that would exceed the design limits?

   (4) Will the proposed activity cause a change to any system interface in a way that would increase the frequency of an accident?

If the proposed activity affects the overall system performance in a manner that could cause an accident previously evaluated to shift to a higher frequency category, or result in a calculated frequency increase to be 10% or

greater (unless the resultant frequency of occurrence remains below 1E-6 or applicable regulatory threshold), then the proposed activity would more than minimally increase the frequency of occurrence of an accident previously evaluated in the UFSAR.

Although this criterion allows minimal increases, licensees must still meet applicable regulatory requirements and other acceptance criteria to which they are committed (such as contained in Regulatory Guides and nationally recognized industry consensus standards, e.g., the ASME B&PV Code and IEEE standards). Further, departures from the design, fabrication, construction, testing, and performance standards as outlined in the General Design Criteria (Appendix A to Part 50) are not compatible with a "no more than minimal increase" standard.

Because frequencies of occurrence of natural phenomena were established as part of initial licensing and are not expected to change, changes in design requirements for earthquakes, tornadoes and other natural phenomena should be treated as potentially affecting the likelihood of a malfunction rather than the frequency of occurrence of an accident.

4.3.2 Does the Activity Result in More than a Minimal Increase in the Likelihood of Occurrence of a Malfunction of an SSC Important to Safety?

The term "malfunction of an SSC important to safety" refers to the failure of structures, systems and components (SSCs) to perform their intended design functions—including both non-safety-related and safety-related SSCs. The cause and mode of a malfunction should be considered in determining whether there is a change in the likelihood of a malfunction. The effect or result of a malfunction should be considered in determining whether a malfunction with a different result is involved per Section 4.3.6.

In determining whether there is more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC to perform its design function as described in the UFSAR, the first step is to determine what SSCs are affected by the proposed activity. Next, the effects of the proposed activity on the affected SSCs should be determined. This evaluation should include both direct and indirect effects.

Direct effects are those where the proposed activity affects the SSCs (e.g., a motor change on a pump). Indirect effects are those where the proposed activity affects one SSC and this SSC affects the capability of another SSC to perform its UFSAR described design function. Indirect effects also include the effects of proposed activities on the design functions of SSCs credited in

the safety analyses. The safety analysis assumes certain design functions of SSCs in demonstrating the adequacy of design. Thus, certain design functions, while not specifically identified in the safety analysis, are credited in an indirect sense.

After determining the affect of the proposed activity on the important to safety SSCs, a determination is made of whether the likelihood of a malfunction of the important to safety SSCs has increased more than minimally. Qualitative engineering judgment and/or an industry precedent is typically used to determine if there is more than a minimal increase in the likelihood of occurrence of a malfunction. An appropriate calculation can be used to demonstrate the change in likelihood in a quantitative sense, if available and practical. The effect of a proposed activity on the likelihood of malfunction must be discernable and attributable to the proposed activity in order to exceed the more than minimal increase standard. A proposed activity is considered to have a negligible effect on the likelihood of a malfunction when a change in likelihood is so small or the uncertainties in determining whether a change in likelihood has occurred are such that it cannot be reasonably concluded that the likelihood has actually changed (i.e., there is no clear trend towards increasing the likelihood). A proposed activity that has a negligible effect satisfies the minimal increase standard.

Evaluations of a proposed activity for its effect on likelihood of a malfunction would be performed at level of detail that is described in the UFSAR. The determination of whether the likelihood of malfunction is more than minimally increased is made at a level consistent with existing UFSAR-described failure modes and effects analyses. While the evaluation should take into account the level that was previously evaluated in terms of malfunctions and resulting event initiators or mitigation impacts, it also needs to consider the nature of the proposed activity. Thus, for instance, if failures were previously postulated on a train level because the trains were independent, a proposed activity that introduces a cross-tie or credible common mode failure (e.g., as a result of an analog to digital upgrade) should be evaluated further to see whether the likelihood of malfunction has been increased.

The following considerations, as applicable, may be useful in determining if an activity involves more than a minimal increase in likelihood of malfunction:

a. Will the proposed activity meet the design requirements for material and construction practices considering:

1. Does the proposed activity satisfy applicable design bases (e.g., seismic or wind loadings, etc.)?

2. Does the change cause applicable design stresses to exceed their code allowables or other applicable stress or deformation limit (if any), recognizing that, to ensure pump functionality, vendor-specified stress limits for a pump casing may be well below the ASME Code allowable.

3. Are the seismic specifications met (such as use of proper supports, proper lugging at terminals, and isolation of lifted leads)?

4. Are separation criteria met (such as minimum distance between circuits in separate divisions, channels in the same division, and jumpers run in conduit)?

5. Are the environmental qualification criteria met (such as use of materials qualified for the environment, e.g., radiation, chemical, thermal, etc., in which they will be used)?

b. Will the proposed activity adversely affect the safety analyses by:

1. Degrading the performance of a safety system assumed to function in the safety analyses below the level of performance assumed in the safety analysis?

2. Increasing challenges to safety systems assumed to function in the safety analyses.

c. Will the proposed activity degrade SSC reliability below the assumed level of performance by:

1. Imposing additional loads not analyzed in the design requirements?

2. Deleting or modifying system/equipment protection features?

3. Downgrading the support system performance necessary for reliable operation of the important to safety equipment?

4. Reducing system/equipment redundancy, diversity or independence?

5. Increasing the frequency of operation of important to safety SSCs?

6. Imposing increased or more severe testing requirements on important to safety SSCs?

7. Adding more components that are subject to failure?

8. Increasing the likelihood of occurrence of malfunction by more than a factor of two[2]?

   Note: Item 8 is for use when the change in likelihood of a malfunction is calculated in support of a 10 CFR 50.59 evaluation.

Changes in design requirements for earthquakes, tornadoes, and other natural phenomena should be treated as potentially affecting the likelihood of malfunction.

Although this criterion allows minimal increases, licensees must still meet applicable regulatory requirements and other acceptance criteria to which they are committed (such as contained in Regulatory Guides and nationally recognized industry consensus standards, e.g., the ASME B&PV Code and IEEE standards). Further, departures from the design, fabrication, construction, testing, and performance standards as outlined in the General Design Criteria (Appendix A to Part 50) are not compatible with a "no more than minimal increase" standard.

Below are examples where there is less than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety:

1. The change involves installing additional equipment or devices (e.g., cabling, manual valves, protective features) provided all applicable design, functional and quality requirements (including applicable codes, standards, etc.) continue to be met. For example, adding protective devices to breakers or installing an additional drain line (with appropriate isolation capability) would not cause more than a minimal increase the likelihood of malfunction.

2. The change involves substitution of one type of component for another of similar function, provided all applicable design, functional and quality requirements (including applicable codes, standards, etc.) continue to be met and any new failure modes are bounded by the existing analysis.

2 The proposed factor of two threshold is consistent with the NRC report, "Options for Incorporating Risk Insights into 10 CFR 50.59 Process," December 17, 1998, Section 6.4.1.

3. The change involves a new or modified operator action that supports a design function credited in safety analyses, including manual action that substitutes for automatic action, provided:

   - The action (including required completion time) is reflected in plant procedures and operator training programs
   - The licensee has demonstrated that the action can be completed in the time required considering the aggregate affects, such as workload or environmental conditions, expected to exist when the action is required
   - The evaluation of the change considers the ability to recover from credible errors in performance of manual actions and the expected time required to make such a recovery
   - The evaluation considers the effect of the change on plant systems

4.3.3 Does the Activity Result in More than a Minimal Increase in the Consequences of an Accident?

The UFSAR, based on logic similar to ANSI standards, provides an acceptance criterion and frequency relationship for "conditions for design". When determining which activities represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that the objective of the regulation is the protection of public health and safety. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators. Changes in barrier performance or other outcomes of the proposed activity that do not result in increased radiological dose to the public or to control room operators are addressed under Section 4.3.7, Integrity of Fission Product Barriers, or the other criteria of 10 CFR 50.59(c)(2).

NRC regulates compliance with the provisions of 10 CFR 50 and 10 CFR 100 to assure adequate protection of the public health and safety. Activities affecting onsite dose consequences that may require prior NRC approval are those that impede required actions inside or outside the control room to mitigate the consequences of reactor accidents.

The consequences covered include dose resulting from any accident evaluated in the UFSAR. The accidents include those typically covered in UFSAR Chapters 6 and 15 and other events with which the plant is designed to cope and are described in the UFSAR (e.g., turbine missiles and flooding). The consequences referred to in 10 CFR 50.59 do not apply to occupational exposures resulting from routine operations, maintenance, testing, etc.

Occupational doses are controlled and maintained As Low As Reasonably Achievable (ALARA) through formal licensee programs.

10 CFR Part 20 establishes requirements for protection against radiation during normal operations, including dose criteria relative to radioactive waste handling and effluents. 10 CFR 50.59 accident dose consequence criteria and evaluation guidance are not applicable to proposed activities governed by 10 CFR Part 20 requirements.

The dose consequences referred to in 10 CFR 50.59 are those calculated by licensees—not the results of independent, confirmatory dose analyses by the NRC that may be documented in Safety Evaluation Reports.

The evaluation should determine the dose that would likely result from accidents associated with the proposed activity. If a proposed activity would result in more than a minimal increase in dose from the existing calculated dose for any accident, then the activity would require prior NRC approval. Where a change in consequences is so small or the uncertainties in determining whether a change in consequences has occurred are such that it cannot be reasonably concluded that the consequences have actually changed (i.e., there is no clear trend towards increasing the consequences), the change need not be considered an increase in consequences.

General Design Criterion 19 of Appendix A to 10 CFR 50 requires radiation protection to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, for the duration of the accident. 10 CFR 100 establishes requirements for exclusion area and low population zones around the reactor so that an individual located at any point on its boundary immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose of 300 rem to the thyroid for iodine exposure. In the Standard Review Plan (SRP), NUREG-0800, the NRC established acceptance guidelines for certain events that are considered of greater likelihood than the limiting accidents. For example, for a steam generator tube rupture, the SRP acceptance guideline is that the dose be less than or equal to a small fraction (i.e., 10 percent) of the 10 CFR 100 thyroid dose value, or 30 rem.

Therefore, for a given accident, calculated or bounding dose values for that accident would be identified in the UFSAR. These dose values should be within the GDC 19 or 10 CFR 100 limits, as applicable, as modified by SRP guidelines (e.g., small fraction of 10 CFR 100), as applicable. An increase in consequences from a proposed activity is defined to be no more than minimal if the increase (1) is less than or equal to 10 percent of the difference between the current calculated dose value and the regulatory guideline value (10 CFR

100 or GDC 19, as applicable), and (2) the increased dose does not exceed the current SRP guideline value for the particular design basis event. The current calculated dose values are those documented in the most up-to-date analyses of record. This approach establishes the current SRP guideline values as a basis for minimal increases for all facilities, not just those that were specifically licensed against those guidelines[3].

For some licensees the current calculated dose consequences may already be in excess of the SRP guidelines for some events. In such cases minimal is defined as less than or equal to 0.1 rem.

In determining if there is more than a minimal increase in consequences, the first step is to determine which accidents evaluated in the UFSAR may have their radiological consequences affected as a direct result of the proposed activity. Examples of questions that assist in this determination are:

(1) Will the proposed activity change, prevent or degrade the effectiveness of actions described or assumed in an accident discussed in the UFSAR?

(2) Will the proposed activity alter assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR?

(3) Will the proposed activity play a direct role in mitigating the radiological consequences of an accident described in the UFSAR?

The next step is to determine if the proposed activity does, in fact, increase the radiological consequences of any of the accidents evaluated in the UFSAR. If it is determined that the proposed activity does have an effect on the radiological consequences of any accident analysis described in the UFSAR, then either:

(1) Demonstrate and document that the radiological consequences of the accident described in the UFSAR are bounding for the proposed activity (e.g., by showing that the results of the UFSAR analysis bound those that would be associated with the proposed activity), or

(2) Revise and document the analysis taking into account the proposed activity and determine if more than a minimal increase has occurred as described above.

[3] For licensees who adopt the alternative source term, evaluations against this criterion should be in terms of total effective dose equivalent and the limits established by 10 CFR 50.67 (effective January 24, 2000).

The following examples illustrate the implementation of this criterion. In each example it is assumed that the calculated consequences do not include a change in the methodology for calculating the consequences. Changes in methodology would need to be separately considered under 10 CFR 50.59(c)(2)(viii) as discussed in Section 4.3.8.

Example 1

The calculated fuel handling accident (FHA) dose is 50 rem to the thyroid at the exclusion area boundary. As a result of a proposed change, the calculated FHA dose would increase to 70 rem. Ten percent of the difference between the calculated value and the regulatory limit is 25 rem [10% of (300 rem- 50 rem)]. The SRP acceptance guideline is 75 rem. Since the calculated increase is less than 25 rem and the total is less than the SRP guideline, the licensee may make the change without prior NRC review.

Example 2

The calculated dose consequence for a steam generator tube rupture accident is 25 rem thyroid at the exclusion area boundary. As a result of a proposed change, the calculated dose consequence would increase to 29 rem thyroid. The change can be made without prior NRC approval because the new calculated dose does not exceed the established SRP guideline of 30 rem thyroid nor does the incremental change in consequences (4 rem) exceed 10 percent of the difference between the previous calculated value and the regulatory limit of 300 rem thyroid. Ten percent of the difference between the regulatory limit (300 rem) and the calculated value (25 rem) is 27.5 rem (10% of 275). Since 4 rem is less than 27.5, this change is a minimal increase permissible under 10 CFR 50.59.

Example 3

The calculated dose consequence of a fuel handling accident is 25 rem to the thyroid at the exclusion area boundary. Because of a proposed change, the calculated dose consequence would increase to 65 rem. The SRP guideline for this accident is 75 rem and is still met. The incremental increase in dose consequence (40 rem), however, exceeds 10 percent of the difference to the regulatory limit or 27.5 rem [10% of (300 rem - 25 rem)]. Therefore, the change results in more than a minimal increase in consequences and thus requires prior NRC approval.

Example 4

The calculated dose to the control room operators following a loss of coolant accident is 4 rem whole body. A change is proposed to the control room ventilation system such that the calculated dose would increase to 4.5 rem.

The regulations dictate that the control room doses are to be controlled to less than 5 rem by General Design Criterion 19. Although the new calculated dose is less than the regulatory limits, the incremental increase in dose (0.5 rem) exceeds the value of 10 percent of the difference between the previously calculated value and the regulatory value or 0.1 rem [10% of (5 rem - 4 rem)]. This change would require prior NRC review as a more than minimal change in consequences.

Example 5

The existing safety analysis for a fuel handling accident predicts an offsite dose to the thyroid of 77 rem. The SRP guideline for this event is 75 rem. A proposed change would result in an increase in the calculated dose from 77 to 77.1 rem. In this case, the proposed change would be a minimal increase in consequences because the new calculated value, even though greater than the SRP value, is within the guideline limit of 0.1 rem.

4.3.4 Does the Activity Result in More than a Minimal Increase in the Consequences of a Malfunction?

In determining if there is more than a minimal increase in consequences, the first step is to determine which malfunctions evaluated in the UFSAR have their radiological consequences affected as a result of the proposed activity. The next step is to determine if the proposed activity does, in fact, increase the radiological consequences and, if so, are they more than minimally increased. The guidance for determining whether a proposed activity results in more than a minimal increase in the consequences of a malfunction is the same as that for accidents. Refer to Section 4.3.3.

4.3.5 Does the Activity Create a Possibility for an Accident of a Different Type?

The set of accidents that a facility must postulate for purposes of UFSAR safety analyses, including LOCA, other pipe ruptures, rod ejection, etc., are often referred to as "design basis accidents." The terms accidents and transients are often used in regulatory documents (e.g., in Chapter 15 of the Standard Review Plan), where transients are viewed as the more likely, low consequence events and accidents as less likely but more serious. In the context of probabilistic risk assessment, transients are typically viewed as initiating events, and accidents as the sequences that result from various combinations of plant and safety system response. This criterion deals with creating the possibility for accidents of similar frequency and significance to those already included in the licensing basis for the facility. Thus, accidents

that would require multiple independent failures or other circumstances in order to "be created" would not meet this criterion.

Certain accidents are not discussed in the UFSAR because their effects are bounded by other related events that are analyzed. For example, a postulated pipe break in a small line may not be specifically evaluated in the UFSAR because it has been determined to be less limiting than a pipe break in a larger line in the same area. Therefore, if a proposed design change would introduce a small high energy line break into this area, postulated breaks in the smaller line need not be considered an accident of a different type.

The possible accidents of a different type are limited to those that are as likely to happen as those previously evaluated in the UFSAR. The accident must be credible in the sense of having been created within the range of assumptions previously considered in the licensing basis (e.g., random single failure, loss of offsite power, etc.). A new initiator of an accident previously evaluated in the UFSAR is not a different type of accident. Such a change or activity, however, which increases the frequency of an accident previously thought to be incredible to the point where it becomes as likely as the accidents in the UFSAR, could create the possibility of an accident of a different type. For example, there are a number of scenarios, such as multiple steam generator tube ruptures, that have been analyzed extensively. However, these scenarios are of such low probability that they may not have been considered to be part of the design basis. However, if a change or activity is proposed such that a scenario such as a multiple steam generator tube rupture becomes credible, the change or activity could create the possibility of an accident of a different type. In some instances these example accidents could already be discussed in the UFSAR.

In evaluating whether the proposed change or activity creates the possibility of an accident of a different type, the first step is to determine the types of accidents that have been evaluated in the UFSAR. The types of credible accidents that the proposed activity could create that are not bounded by UFSAR-evaluated accidents are accidents of a different type.

4.3.6 Does the Activity Create a Possibility for a Malfunction of an SSC Important to Safety with a Different Result?

Malfunctions of SSCs are generally postulated as potential single failures to evaluate plant performance with the focus being on the result of the malfunction rather than the cause or type of malfunction. A malfunction that involves an initiator or failure whose effects are not bounded by those explicitly described in the UFSAR is a malfunction with a different result. A

new failure mechanism is not a malfunction with a different result if the result or effect is the same as, or is bounded by, that previously evaluated in the UFSAR. The following examples illustrate this point:

- If a pump is replaced with a new design, there may be a new failure mechanism introduced that would cause a failure of the pump to run. But if this effect (failure of the pump to run) was previously evaluated and bounded, then a malfunction with a different result has not been created.

- If a feedwater control system is being upgraded from an analog to a digital system, new components may be added which could fail in ways other than the components in the original design. Provided the end result of the component or subsystem failure is the same as, or is bounded by, the results of malfunctions currently described in the UFSAR (i.e., failure to maximum demand, failure to minimum demand, failure as-is, etc.), then this upgrade would not create a "malfunction with a different result."

Certain malfunctions are not explicitly described in the UFSAR because their effects are bounded by other malfunctions that are described. For example, failure of a lube oil pump to supply oil to a component may not be explicitly described because a failure of the supplied component to operate was described.

The possible malfunctions with a different result are limited to those that are as likely to happen as those described in the UFSAR. For example, a seismic induced failure of a component that has been designed to the appropriate seismic criteria will not cause a malfunction with a different result. However, a proposed change or activity that increases the likelihood of a malfunction previously thought to be incredible to the point where it becomes as likely as the malfunctions assumed in the UFSAR, could create a possible malfunction with a different result.

In evaluating a proposed activity against this criterion, the types and results of failure modes of SSCs that have previously been evaluated in the UFSAR and that are affected by the proposed activity should be identified. This evaluation should be performed consistent with any failure modes and effects analysis (FMEA) described in the UFSAR, recognizing that certain proposed activities may require a new FMEA to be performed. Attention must be given to whether the malfunction was evaluated in the accident analyses at the component level or the overall system level. While the evaluation should take into account the level that was previously evaluated in terms of malfunctions and resulting event initiators or mitigation impacts, it also needs to consider the nature of the proposed activity. Thus, for instance, if

failures were previously postulated on a train level because the trains were independent, a proposed activity that introduces a cross-tie or credible common mode failure (e.g., as a result of an analog to digital upgrade) should be evaluated further to see whether new outcomes have been introduced.

Once the malfunctions previously evaluated in the UFSAR and the results of these malfunctions have been determined, then the types and results of failure modes that the proposed activity could create are identified. Comparing the two lists can provide the answer to the criterion question. An example that might create a malfunction with a different result could be the addition of a normally open vent line in the discharge of an emergency core cooling system pump. The different result of a malfunction could be potential voiding in the system causing it not to operate properly.

4.3.7 Does the Activity Result in A Design Basis Limit for a Fission Product Barrier Being Exceeded or Altered?

10 CFR 50.59 evaluation under criterion (c)(2)(vii) focuses on the fission product barriers—fuel cladding, reactor coolant system boundary, and containment—and on the critical design information that supports their continued integrity. Guidance for applying this criterion is structured around a two-step approach:

- Identification of affected design basis limits for a fission product barrier
- Determination of when those limits are exceeded or altered.

<u>Identification of affected design basis limits for a fission product barrier</u>

The first step is to identify the fission product barrier design basis limits, if any, that are affected by a proposed activity. Design basis limits for a fission product barrier are the controlling numerical values established during the licensing review as presented in the UFSAR for any parameter(s) used to determine the integrity of the fission product barrier. These limits have three key attributes:

- **The parameter is fundamental to the barrier's integrity.** Design basis limits for fission product barriers establish the reference bounds for design of the barriers, as defined in 10 CFR 50.2. They are the limiting values for parameters that directly determine the performance of a fission product barrier. That is, design bases limits are fundamental to barrier integrity and may be thought of as the point at which confidence in the

barrier begins to decrease.

For purposes of this evaluation, design bases parameters should be distinguished from other parameters that—while they may affect fission product barrier performance—are of secondary importance. For example, a change to fuel burn-up limits would be evaluated for its effect on clad strain to determine if it caused the limiting value for fuel internal gas pressure to be exceeded. Thus fuel internal gas pressure is a fundamental design bases limit for fuel cladding integrity, and fuel burn-up is a secondary/subordinate parameter/limit. Similarly, linear heat rate and RCS usage factor limits affect the fuel cladding and RCS boundary but are subordinate, respectively, to the design bases limits for fuel temperature and RCS stresses.

- **The limit is expressed numerically.** Design basis limits are numerical values used in the overall design process, not descriptions of functional requirements. Design basis limits are typically the numerical event acceptance criteria utilized in the accident analysis methodology. The facility's design and operation associated with these parameters as described in the UFSAR will be at or below (more conservative than) the design basis limit.

- **The limit is identified in the UFSAR.** As required by 10 CFR 50.34(b), design basis limits were presented in the original FSAR and continue to reside in the UFSAR. They may be located in a vendor topical report that is incorporated by reference in the UFSAR.

Consistent with the discussion of 10 CFR 50.59 applicability in Section 4.1, any design basis limit for a fission product barrier that is controlled by another, more specific regulation or Technical Specification would not require evaluation under Criterion (c)(2)vii. The effect of the proposed activity on those parameters would be evaluated in accordance with the more specific regulation. Effects (either direct or indirect—see discussion below) on design basis parameters covered by another regulation or Technical Specification need not be considered as part of evaluations under this criterion.

Examples of typical fission product barrier design basis limits are identified in the following table:

Barrier	Design Bases Parameter	Typical Design Basis Limit
Fuel Cladding	DNBR/MCPR	95/95 DNB
	Fuel temperature	Centerline fuel melting temperature
	Fuel enthalpy	Cal/gm associated with dispersion
	Clad strain	Internal pressure associated with clad lift-off
	Clad temperature *	2200 degrees F
	Clad Oxidation *	17% local and 1 % overall
RCS Boundary	Pressure	Designated limit in safety analysis for specific accident
	Stresses *	ASME code compliance for normal, upset, faulted, etc., as appropriate for accident
	Heat-up/Cool-down*	Applicable ASME Code stress limits
Containment	Pressure	Containment design pressure

* These parameters are commonly controlled by 10 CFR 50.55a, 10 CFR 50.46 and/or a specific Technical Specification and therefore would not be subject to evaluation under this criterion.

The list above may vary slightly for a given facility and/or fuel vendor and may include other parameters for specific accidents. For example,

- PWR licensees may utilize 100% pressurizer level as a limiting parameter to ensure RCS integrity for some accident sequences.

- Licensees may consider fuel burn-up limit as a design bases limit for the fuel cladding rather than more specific (fundamental) fuel integrity parameters, such as rod internal pressure.

If a given facility has these or other parameters incorporated into the UFSAR as a design basis limit for a fission product barrier, then changes affecting it should be evaluated under this criterion.

Two of the ways that a licensee can evaluate proposed activities against this criterion are as follows. The licensee may identify all design bases parameters for fission product barriers and include them explicitly in the procedure for performing 10 CFR 50.59 evaluations. Alternatively, the effects of a proposed activity could be evaluated first to determine if the change affects design bases parameters for fission product barriers. The results of these two approaches are equivalent provided the guidance for "exceeded or altered" described below is followed. In all cases, the direct and indirect effects of proposed activities must be included in the evaluation.

Exceeded or altered

A specific proposed activity requires a license amendment if the design basis limit for a fission product barrier is "exceeded or altered." The term "exceeded" means that as a result of the proposed activity, the facility's predicted response would be less conservative than the numerical design basis limit identified above. The term "altered" means the design basis limit itself is changed.

The effect of the proposed activity includes both direct and indirect effects. Extending the maximum fuel burn-up limits until the fuel rod internal gas pressure exceeds the design basis limit is a direct effect that would require a license amendment. Indirect effects provide for another parameter or effect to cascade from the proposed activity to the design basis limit. For example, reducing the design flow of auxiliary feedwater pumps following a loss of main feedwater could reduce the heat transferred from the RCS to the steam generators. That effect could increase the RCS temperature, which would raise RCS pressure and pressurizer level. The 10 CFR 50.59(c)(2)(vii) evaluation of this change would focus on whether the design basis limit associated with RCS pressure for that accident sequence would be exceeded.

Altering a design basis limit for a fission product barrier is not a routine activity, but it can occur. An example of this would be changing the DNBR value such that it no longer corresponds to a 95/95 DNB, perhaps as a result of a new fuel design being implemented with the existing correlation. (A new correlation or a new value for 95/95 DNB with the same fuel type would be evaluated under criterion (c)(2)(viii) of the rule.) Another example is redesigning portions of the RCS boundary to no longer comply with the code of construction. These are infrequent activities affecting key elements of the defense-in-depth philosophy. As such, no distinction has been made between a conservative and non-conservative change in the limit.

Evaluations performed under this criterion may incorporate a number of refinements to simplify the review. For example, if an engineering evaluation demonstrates that no parameters are affected that have design

basis limits for fission product barriers associated with them, no 10 CFR 50.59(c)(2)(vii) evaluation is required. Similarly, most parameters that require evaluation under this criterion have calculations or analyses supporting the facility's design. If an engineering evaluation demonstrates that the analysis presented in the UFSAR remains bounding, then no 10 CFR 50.59(c)(2)(vii) evaluation is required. When using these techniques, both indirect and direct effects must be considered to ensure that important interactions are not overlooked.

Examples illustrating the two-step approach for evaluations under this criterion are provided below:

Example 1

It is proposed to delay the automatic start of the stand-by condensate booster pump to eliminate spurious automatic starts. The proposed change is of sufficient magnitude such that it "screens in" as affecting a UFSAR-described design function.

<u>Identification of design basis limits</u>

The direct effects of a reduction in condensate flow would be reviewed to identify potentially affected design basis parameters. In addition, the indirect effect on feedwater flow and feedwater pump NPSH of a possible transient reduction in condensate flow/pressure would be considered. Likewise, consideration of indirect effects would be extended to the reactor or steam generator (BWR or PWR, as applicable). The review concludes that no design basis limits are either directly or indirectly affected.

The change in the probability of a reactor trip as a result of normal condensate system malfunctions would be evaluated under other 10 CFR 50.59 criteria.

<u>Exceeded or altered</u>

Since no design basis limits were identified, this element of the evaluation is not applicable.

Example 2

The heat transfer capability of an RHR heat exchanger tube bundle has degraded, and it is proposed to accept the condition "as-is."

Identification of design basis limits

The effects of the reduced heat transfer capability would be reviewed. The direct effect would include the increased temperature of the suppression pool or containment sump [BWR or PWR, as applicable]. The indirect effects would include increasing the peak containment post-accident pressure and increased enthalpy of ECCS flow. The increased ECCS enthalpy would also affect peak clad temperature (PCT). Thus, the proposed activity affects two design basis limits: containment pressure and PCT. In this example, the design basis limits would most likely serve as the acceptance criteria for the two parameters in the LOCA analysis described in the UFSAR. (Most licensees use containment design pressure and 2200 degrees F for those values.)

Exceeded or altered

Any increase in peak containment post-accident pressure would be compared to the design basis limit, in this case, containment design pressure. If the revised peak post-accident containment pressure exceeded the design basis limit, then a license amendment would be required.

On the other hand, PCT is governed by a more specific regulation, 10 CFR 50.46. Therefore, the evaluation under this criterion would not address the impact on this parameter. Rather, any changes or corrections to an acceptable evaluation model or application of such a model that affects the PCT calculation would be evaluated per the requirements of 10 CFR 50.46(3)(ii).

In this example, the design basis limits for containment pressure or PCT are not being "exceeded or altered." Therefore, this element of the review is not applicable.

Example 3

Recently identified corrosion inside the primary containment has prompted a re-evaluation of the existing containment design pressure of 55 psig. This re-evaluation has concluded that a design pressure of 48 psig is the maximum supportable. As the final resolution to the degraded containment condition, the licensee proposes to reduce the containment design pressure as reflected in UFSAR safety analyses from 55 to 48 psig.

Identification of Design Basis Limit

The affected parameter is post accident peak containment pressure. This parameter directly affects the containment barrier. Its design basis limit from the UFSAR is the existing containment design pressure of 55 psig.

Exceeded or altered

The design basis limit itself has been "altered" and thus a license amendment is required. The issue of conservative vs. non-conservative is not germane to requiring a submittal. That is, prior NRC approval is required regardless of direction because this is a fundamental change in the facility's design.

4.3.8 Does the Activity Result in a Departure from a Method of Evaluation Described in the UFSAR Used in Establishing the Design Bases or in the Safety Analyses?

The UFSAR contains design and licensing basis information for a nuclear power facility, including description on how regulatory requirements for design are met and how the facility responds to various design basis accidents and events. Analytical methods are a fundamental part of demonstrating how the design meets regulatory requirements and why the facility's response to accidents and events is acceptable. As such, in cases where the analytical methodology was considered to be an important part of the conclusion that the facility met the required design bases, these analytical methods were described in the UFSAR and received varying levels of NRC review and approval during licensing.

Because 10 CFR 50.59 provides a process for determining if prior NRC approval is required before making changes to the facility as described in the UFSAR, changes to the methodologies described in the UFSAR also fall under the provisions of the 10 CFR 50.59 process, specifically criterion (c)(2)(viii). In general, licensees can make changes to elements of a methodology without first obtaining a license amendment if the results are essentially the same as, or more conservative than, previous results. Similarly, licensees can also use different methods without first obtaining a license amendment if those methods have been approved by the NRC for the intended application.

If the proposed activity does not involve a change to a method of evaluation, then the 10 CFR 50.59 evaluation should reflect that this criterion is not applicable. If the activity involves only a change to a method of evaluation,

then the 10 CFR 50.59 evaluation should reflect that criteria 10 CFR 50.59(c)(2)(i—vii) are not applicable.

The first step in applying this criterion is to identify the methods of evaluation that are affected by the change. This is accomplished during application of the screening criteria in Section 4.2.1.3.

Next, the licensee must determine whether the change constitutes a departure from a method of evaluation that would require prior NRC approval. As discussed further below, for purposes of evaluations under this criterion, the following changes are considered a departure from a method of evaluation described in the UFSAR:

- Changes to any element of analysis methodology that yield results that are non-conservative or not essentially the same as the results from the analyses of record.

- Use of new or different methods of evaluation that are not approved by NRC for the intended application.

By way of contrast, the following changes are <u>not</u> considered departures from a method of evaluation described in the UFSAR:

- Departures from methods of evaluation that are not described, outlined or summarized in the UFSAR (such changes may have been screened out as discussed in Section 4.2.1.3);

- Use of an upgraded or new NRC-approved methodology (e.g., computer code) to reduce uncertainty, provide more precise results, or other reason, provided such use is (a) based on sound engineering practice, (b) appropriate for the intended application, and (c) within the limitations of the applicable SER. The basis for this determination should be documented in the licensee evaluation.

- Use of a methodology revision that is documented as providing results that are consistent with or more conservative than either the previous revision of the same methodology or with another methodology previously accepted by NRC through issuance of an SER.

Subsection 4.3.8.1 provides guidance for making changes to one or more elements of an existing method of evaluation used to establish the design bases or in the safety analyses. Subsection 4.3.8.2 provides guidance for adopting an entirely new method of evaluation to replace an existing one.

Examples illustrating the implementation of this criterion are provided in Section 4.3.8.3.

4.3.8.1 Guidance for Changing One or More Elements of a Method of Evaluation

The definition of "departure ..." provides licensees with the flexibility to make changes under 10 CFR 50.59 to methods of evaluation whose results are "conservative" or that are not important with respect to the demonstrations of performance that the analyses provide. Changes to elements of analysis methods that yield conservative results, or results that are essentially the same would not be departures from approved methods.

Conservative vs. Non-Conservative Results

Gaining margin by changing one or more elements of a method of evaluation is considered to be a non-conservative change and thus a departure from a method of evaluation for purposes of 10 CFR 50.59. Such departures require prior NRC approval of the revised method. Analytical results obtained by changing any element of a method are "conservative" relative to the previous results, if they are closer to design bases limits or safety analyses limits (e.g., applicable acceptance guidelines). For example, a change from 45 psig to 48 psig in the result of a containment peak pressure analysis (with design basis limit of 50 psig) using a revised method of evaluation would be considered a conservative change when applying this criterion. In other words, the revised method is more conservative if it predicts more severe conditions given the same set of inputs. This is because results closer to limiting values are considered conservative in the sense that the new analysis result provides less margin to applicable limits for making potential physical or procedure changes without a license amendment.

In contrast, if the use of a modified method of evaluation resulted in a change in calculated containment peak pressure from 45 psig to 40 psig, this would be a non-conservative change. That is because the change would result in more margin being available (to the design basis limit of 50 psig) for the licensee to make more significant changes to the physical facility or procedures.

"Essentially the Same"

Licensees may change one or more elements of a method of evaluation such that results move in the non-conservative direction without prior NRC approval, provided the revised result is "essentially the same" as the previous result. Results are "essentially the same" if they are within the margin of

error for the type of analysis being performed. Variation in results due to routine analysis sensitivities or calculational differences (e.g., rounding errors and use of different computational platforms) would typically be within the analysis margin of error and thus considered "essentially the same." For example, when a method is applied using a different computational platform (mainframe vs workstation), results of cases run on the two platforms differed by less than 1%, which is the margin of error for this type of calculation. Thus the results are essentially the same, and do not constitute a departure from a method that requires prior NRC approval.

The determination of whether a new analysis result would be considered "essentially the same" as the previous result can be made through benchmarking the revised method to the existing one, or may be apparent from the nature of the differences between the methods. When benchmarking a revised method to determine how it compares to the previous one, the analyses that are done must be for the same set of plant conditions to ensure that the results are comparable. Relative to the original method, the revised method may result in differences in the details, or intermediate results, of an analysis; however, the end results of the existing and revised analyses must be essentially the same.

4.3.8.2 Guidance for Changing from One Method of Evaluation to Another

The definition of "departure ..." provides licensees with the flexibility to make changes under 10 CFR 50.59 from one method of evaluation to another provided that the new method is approved by the NRC for the intended application. A new method is approved by the NRC for intended application if it is approved for the type of analysis being conducted, and applicable terms, conditions and limitations for its use are satisfied.

NRC approval has typically followed one of two paths. Most reactor or fuel vendors and several utilities have prepared and obtained NRC approval of topical reports that describe methodologies for the performance of a given type or class of analysis. Through a Safety Evaluation Report, the NRC approved the use of the methodologies for a given class of power plants. In some cases, the NRC has accorded "generic" approval of analysis methodologies. Terms, conditions and limitations relating to the application of the methodologies are usually documented in the topical reports, the SER, and correspondence between the NRC and the methodology owner that is referenced in the SER or associated transmittal letter.

The second path is the approval of a specific analysis rather than a more generic methodology. The NRC's approval has tended to be limited to a given

plant design and a given application. Again, terms, conditions and limitations relating to the application of the methodologies are usually documented in the original license amendment request, the SER, and any correspondence between the NRC and the analysis owner that is referenced in the SER or associated transmittal letter.

It is incumbent upon the user of a new methodology—even one generically approved by the NRC—to ensure that all conditions and limitations under which the method received NRC approval are identified. The applicable terms and conditions for the use of a methodology are not limited to a specific analysis; the qualification of the organization applying the methodology is also a consideration. Through Generic Letter 83-11, Supplement 1, the NRC has established a method by which utilities can demonstrate they are generally qualified to perform safety analyses. Utilities thus qualified can apply methods that have been reviewed and approved by the NRC, or that have been otherwise accepted as part of another plant's licensing basis, without requiring prior NRC approval. Licensees that have not satisfied the guidelines of Generic Letter 83-11, Supplement 1, may, of course, continue to seek plant-specific approval to use new methods of evaluation.

When considering the application of a methodology, it is necessary to adopt the methodology *en toto* and apply it consistent with applicable terms, conditions and limitations. Mixing attributes of new and existing methodologies is considered a revision to a methodology and must be evaluated as such per the guidance in Section 4.3.8.1.

Considerations for Determining if New Methods are Technically Appropriate for the Intended Application

The following questions highlight important considerations for determining that a particular application of a different method is technically appropriate for the intended application, within the bounds of what has been found acceptable by NRC, and does not require prior NRC approval.

- Is the application of the methodology consistent with the facility's licensing basis (e.g., NUREG-0800 or other plant-specific commitments)? Will the methodology supersede a methodology addressed by other regulations such as 10 CFR 50.46, 10 CFR 50.55a or the plant Technical Specifications (Core Operating Limits Report or Pressure/Temperature Limits Report)? Is the methodology consistent with relevant industry standards?

  If application of the new methodology requires exemptions from regulations or plant-specific commitments, exceptions to relevant industry

standards and guidelines, or is otherwise inconsistent with a facility's licensing basis, then prior NRC approval may be required. The applicable change process must be followed to make the plant's licensing basis consistent with the requirements of the new methodology.

- If a computer code is involved, has the code been installed in accordance with applicable software Quality Assurance requirements? Has the plant-specific model been adequately qualified through benchmark comparisons against test data, plant data, or approved engineering analyses? Is the application consistent with the capabilities and limitations of the computer code? Has industry experience with the computer code been appropriately considered?

  The computer code installation and plant-specific model qualification is not directly transferable from one organization to another. The installation and qualification should be in accordance with the licensee's Quality Assurance program.

- Is the plant configuration the same as described in the methodology? If the plant configuration is similar, but not the same, the following types of considerations should be addressed to assess the applicability of the methodology:

  - How could those differences affect the methodology?
  - Are additional sensitivity studies required?
  - Should additional single failure scenarios be considered?
  - Are analyses of limiting scenarios, effects of equipment failures, etc., applicable for the specific plant design?
  - Can analyses be made while maintaining compliance with both the intent and literal definition of the methodology?

Differences in the plant configurations and licensing bases could invalidate the application of a particular methodology. For example, the licensing basis of older vintage plants may not include an analysis of the feedwater line break event that is required in later vintage plants. Some plants may be required to postulate a loss of offsite power or a maximum break size for certain events; other may have obtained exemptions to these requirements from the NRC. The existence of these differences does not preclude application of a new methodology to a facility; it only requires the analyst to thoroughly understand and document the effects of these differences on the application of the methodology to ensure compliance with the terms, conditions, and limitations of the NRC approval.

- Is the facility for which the methodology has been approved designed and operated in the same manner as the facility to which the methodology is to be applied? If the facilities are not designed and operated in the same manner, the following types of considerations should be addressed to assess the applicability of the methodology:

  - Is the equipment the same? Does the equipment have the same pedigree (e.g., Class 1E, Seismic Category I, etc.)? If similar, but not the same, what additional allowances must be made? Are the relevant failure modes and effects analyses the same? If slight modifications to the methodology are required, are these within the terms, conditions, and limitations on which NRC approval of the methodology was based?

  - Even if the basic facility configuration is nearly the same between two units, differences in plant specific components may make the application of a methodology to another plant inappropriate. For example, some plants may have pressurizer power-operated relief valves that are qualified for water relief; other plants do not. In addition, plant specific failure modes and effects analyses may reveal new potential single failure scenarios that were not considered in the original methodology. The existence of these differences does not preclude application of a new methodology to a facility; it only requires the analyst to thoroughly understand the effects of these differences on the application of the methodology to ensure compliance with the terms, conditions, and limitations of the NRC approval.

4.3.8.3 EXAMPLES

The following examples illustrate the implementation of this criterion:

Example 1 - The UFSAR states that a damping value of 0.5 percent is used in the seismic analysis of safety-related piping. The licensee wishes to change this value to 2 percent to reanalyze the seismic loads for the piping. Using a higher damping value to represent the response of the piping to the acceleration from the postulated earthquake in the analysis would result in lower calculated stresses because the increased damping reduces the loads. Since this analysis was used in establishing the seismic design bases for the piping, and since this is a change to an element of the method that is not conservative and is not essentially the same, this change would require prior NRC approval under this criterion.

On the other hand, had NRC approved an alternate method of seismic analysis that allowed 2 percent damping provided certain other assumptions were made, and the licensee used the complete set of assumptions to perform

its analysis, then the 2 percent damping under these circumstances would not be a departure because this method of evaluation is considered "approved by the NRC for the intended application."

Example 2 - A facility has a design basis containment pressure limit of 50 psig. The current worst-case design basis accident calculation results in a peak pressure of 45 psig. The licensee revises the method of evaluation, and the recalculated result is 40 psig. This change would require prior NRC approval because the result of the recalculation is not conservative. If the licensee used a different method that was approved by the NRC and met all the terms and conditions of the method, a recalculated result of 40 psig would not require prior NRC approval.

Example 3 - A licensee revises the seismic analysis described in the UFSAR to include an inelastic analysis procedure. This revised method is used to demonstrate that cable trays have greater capacity than previously calculated. This change would require prior NRC approval as it would not produce results that are essentially the same.

Example 4 - Licensee X has received NRC approval for the use of a method of evaluation at Facility A for performing steamline break mass and energy release calculations for environmental qualification evaluations. The terms and conditions for the use of the method are detailed in the NRC SER. The SER also describes limitations associated with the method. Licensee Y wants to apply the method at its Facility B. Licensee Y has satisfied the guidelines of GL 83-11, Supplement 1. After reviewing the method, approved application, SER and related documentation, to verify that applicable terms, conditions and limitations are met and to ensure the method is applicable to their type of plant, Licensee Y conducts a 10 CFR 50.59 evaluation. Licensee Y concludes that the change is not a departure from a method of evaluation because it has determined the method is appropriate for the intended application, the terms and conditions for its use as specified in the SER have been satisfied, and the method has been approved by the NRC.

Example 5 - The NRC has approved the use of computer code and the associated analysis of a steamline break for use in the evaluation of component stresses. A licensee uses the same computer code and analysis methodology to replace their evaluation of the containment temperature response. This change would require prior NRC approval unless the methodology had been previously approved for evaluating containment temperature response.

4.4 APPLYING 10 CFR 50.59 TO COMPENSATORY ACTIONS TO ADDRESS NONCONFORMING OR DEGRADED CONDITIONS

Three general courses of action are available to licensees to address non-conforming and degraded conditions. Whether or not 10 CFR 50.59 must be applied, and the focus of a 10 CFR 50.59 evaluation if one is required, depends on the corrective action chosen by the licensee, as discussed below:

- If the licensee intends to restore the SSC back to its previous condition (as described in the UFSAR), then this corrective action should be performed in accordance with 10 CFR 50, Appendix B (i.e., in a timely manner commensurate with safety). This activity is not subject to 10 CFR 50.59.

- If an interim compensatory action is taken to address the condition and involves a temporary procedure or facility change, 10 CFR 50.59 should be applied to the temporary change. The intent is to determine whether the temporary change/compensatory action itself (not the degraded condition) impacts other aspects of the facility or procedures described in the UFSAR. In considering whether a temporary change impacts other aspects of the facility, a licensee should pay particular attention to ancillary aspects of the temporary change that result from actions taken to directly compensate for the degraded condition.

- If the licensee corrective action is either to accept the condition "as-is" resulting in something different than described in the UFSAR, or to change the facility or procedures to something different than described in the UFSAR, 10 CFR 50.59 should be applied to the corrective action, unless another regulation applies, e.g., 10 CFR 50.55a. In these cases, the final resolution becomes the proposed change that would be subject to 10 CFR 50.59.

The following example illustrates the process for implementing a temporary change as a compensatory measure to address a degraded/non-conforming condition:

A level transmitter for one Reactor Coolant Pump (RCP) lower oil reservoir failed while at power. The transmitter provides an alarm function, but not an automatic protective action function. The transmitter and associated alarm are described in the UFSAR, as protective features for the RCPs, but no technical specification applies. Loss of the transmitter does not result in the loss of operability for any technical specification equipment. The transmitter fails in a direction resulting in a continuous alarm in the control room. The alarm circuitry provides a common alarm for both the upper and lower oil reservoir circuits, so transmitter failure causes a hanging alarm and a

masking of proper operation of the remaining functional transmitter. Precautionary measures are taken to monitor lower reservoir oil level as outlined in the alarm manual using available alternate means. An interim compensatory action is proposed to lift the leads (temporary change) from the failed transmitter to restore the alarm function for the remaining functioning transmitter.

Lifting the leads is a compensatory action (temporary change) which is subject to 10 CFR 50.59. The 10 CFR 50.59 screening would be applied to the temporary change itself (lifted leads) not the degraded condition (failed transmitter), to determine its impact on other aspects of the facility described in the UFSAR. If screening determines that no other UFSAR-described SSCs would be affected by this compensatory action, the temporary change would screen out, i.e., not require a 10 CFR 50.59 evaluation.

4.5 DISPOSITION OF 10 CFR 50.59 EVALUATIONS

There are two possible conclusions to a 10 CFR 50.59 evaluation:

(1) The proposed activity may be implemented without prior NRC approval.

(2) The proposed activity requires prior NRC approval.

Where an activity requires prior NRC approval, the activity must be approved by the NRC via license amendment in accordance with 10 CFR 50.90 prior to implementation. An activity is considered "implemented" when it provides its intended function, that is, when it is placed in service and declared operable. Thus, a licensee may design, plan, install, and test a modification prior to receiving the license amendment to the extent that these preliminary activities do not themselves require prior NRC approval under 10 CFR 50.59.

For example, a modification to a facility involved the replacement of a train of a safety system with one including diverse primary components (diesel-driven pump vice a motor-driven pump). The installation of the replacement train was largely in a new, separate structure. Ultimately the modification would require NRC approval because of impacts on the facility technical specifications as well as due to differences in reliability of the replacement pump in some situations. There was insufficient time to seek and gain NRC approval prior to construction. The facility prepared a 10 CFR 50.59 screening to support construction of the stand-alone facility through preliminary testing. The limited interfaces with the existing facility were assessed and determined to not affect the facility as described in the UFSAR. Upon receipt of the license amendment the final tie-in, testing and operation were fully authorized. 10 CFR 50.59 should be applied to any aspects of the

activity not adequately addressed in the license amendment request and/or associated Safety Evaluation Report.

For proposed activities that are determined to require prior NRC approval, there are three possible options:

(1) Cancel the planned change.

(2) Redesign the proposed activity so that the it may proceed without prior NRC approval.

(3) Apply for and obtain a license amendment under 10 CFR 50.90 prior to implementing the activity. Technical and licensing evaluations performed for such activities may be used as part of the basis for license amendment requests.

In resolving degraded or nonconforming conditions, the need to obtain NRC approval for a change does not affect the licensee's authority to operate the plant. The licensee may make mode changes, restart from outages, etc., provided that necessary SSCs are operable and the degraded condition is not in conflict with the technical specifications or the license.

It is important to remember that determining that a proposed activity requires prior NRC approval does not determine whether it is safe. In fact, a proposed activity that requires prior NRC approval may significantly enhance overall plant safety at the expense of a small adverse impact in a specific area. It is the responsibility of the utility to assure that proposed activities are safe, and it is the role of the NRC to confirm the safety of those activities that are determined to require prior NRC review.

5.0 DOCUMENTATION AND REPORTING

10 CFR 50.59(d) requires the following documentation and recordkeeping:

(1) The licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test or experiment does not require a license amendment pursuant to paragraph (c)(2) of this section.

(2) The licensee shall submit, as specified in § 50.4, a report containing a brief description of any changes, tests, and experiments, including a

summary of the evaluation of each. A report must be submitted at intervals not to exceed 24 months.

(3) The records of changes in the facility must be maintained until the termination of a license issued pursuant to this part or the termination of a license issued pursuant to 10 CFR Part 54, whichever is later. Records of changes in procedures and records of tests and experiments must be maintained for a period of 5 years.

The documentation and reporting requirements of 10 CFR 50.59(d) apply to activities that require evaluation against the eight criteria of 10 CFR 50.59(c)(2) and are determined not to require prior NRC approval. That is, the phrase in 10 CFR 50.59(d)(1), "made pursuant to paragraph (c)," refers to those activities that were evaluated against the eight evaluation criteria (because, for example, they affect the facility as described in the UFSAR), but not to those activities or changes that were screened out. Similarly, documentation and reporting under 10 CFR 50.59 is not required for activities that are canceled or that that are determined to require prior NRC approval and are implemented via the license amendment request process.

Documenting 10 CFR 50.59 Evaluations

In performing a 10 CFR 50.59 evaluation of a proposed activity, the evaluator must address the eight criteria in 10 CFR 50.59(c)(2) to determine if prior NRC approval is required. Although the conclusion in each criterion may be simply "yes," "no," or "not applicable," there must be an accompanying explanation providing adequate basis for the conclusion. Consistent with the intent of 10 CFR 50.59, these explanations should be complete in the sense that another knowledgeable reviewer could draw the same conclusion. Restatement of the criteria in a negative sense or making simple statements of conclusion is not sufficient and should be avoided. It is recognized, however, that for certain very simple activities, a statement of the conclusion with identification of references consulted to support the conclusion would be adequate and the 10 CFR 50.59 evaluation could be very brief.

The importance of the documentation is emphasized by the fact that experience and engineering knowledge (other than models and experimental data) are often relied upon in determining whether evaluation criteria are met. Thus the basis for the engineering judgment and the logic used in the determination should be documented to the extent practicable and to a degree commensurate with the safety significance and complexity of the activity. This type of documentation is of particular importance in areas where no established consensus methods are available, such as for software

reliability, or the use of commercial-grade hardware and software where full documentation of the design process is not available.

Since an important goal of the 10 CFR 50.59 evaluation is completeness, the items considered by the evaluator must be clearly stated.

Each 10 CFR 50.59 evaluation is unique. Although each applicable criteria must be addressed, the questions and considerations listed throughout this guidance document to assist evaluating the criteria are not requirements for all evaluations. Some evaluations may require that none of these questions be addressed while others will require additional considerations beyond those addressed in this guidance.

When preparing 10 CFR 50.59 evaluations, licensees may combine responses to individual criteria or reference other portions of the evaluation.

As discussed in Section 4.2.3, licensees may elect to use screening criteria to limit the number of activities for which written 10 CFR 50.59 evaluations are performed. A documentation basis should be maintained for determinations that the changes meet the screening criteria, i.e., screen out. This documentation does not constitute the record of changes required by 10 CFR 50.59, and thus is not subject to the recordkeeping requirements of the rule.

Reporting to NRC

A summary of 10 CFR 50.59 evaluations for activities implemented under 10 CFR 50.59 must be provided to NRC. Activities that were screened out, canceled or implemented via license amendment need not be included in this report. The 10 CFR 50.59 reporting requirement (every 24 months) is identical to that for UFSAR updates such that licensees may provide these reports to NRC on the same schedule.

Appendix A

10 CFR 50.59 Changes, tests, and experiments.

(a) Definitions for the purposes of this section:

(1) *Change* means a modification or addition to, or removal from, the facility or procedures that affects a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished.

(2) *Departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses* means (i) changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same; or (ii) changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application.

(3) *Facility as described in the final safety analysis report (as updated means*:

(i)The structures, systems, and components (SSC) that are described in the final safety analysis report (FSAR) (as updated),

(ii) The design and performance requirements for such SSCs described in the FSAR (as updated), and

(iii)The evaluations or methods of evaluation included in the FSAR (as updated) for such SSCs which demonstrate that their intended function(s) will be accomplished.

(4) *Final Safety Analysis Report (as updated)* means the Final Safety Analysis Report (or Final Hazards Summary Report) submitted in accordance with § 50.34, as amended and supplemented, and as updated per the requirements of § 50.71(e) or § 50.71(f), as applicable.

(5) *Procedures as described in the final safety analysis report (as updated)* means those procedures that contain information described in the FSAR (as updated) such as how structures, systems, and components are operated and controlled (including assumed operator actions and response times).

(6) *Tests or experiments not described in the final safety analysis report (as updated)* means any activity where any structure, system, or component is utilized or controlled in a manner which is either:

(i) Outside the reference bounds of the design bases as described in the final safety analysis report (as updated) or

(ii) Inconsistent with the analyses or descriptions in the final safety analysis report (as updated).

(b) Applicability. This section applies to each holder of a license authorizing operation of a production or utilization facility, including the holder of a license authorizing operation of a nuclear power reactor that has submitted the certification of permanent cessation of operations required under § 50.82(a)(1) or a reactor licensee whose license has been amended to allow possession but not operation of the facility.

(c) (1) A licensee may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described in the final safety analysis report (as updated), and conduct tests or experiments not described in the final safety analysis report (as updated) without obtaining a license amendment pursuant to § 50.90 only if:

(i) A change to the technical specifications incorporated in the license is not required, and

(ii) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section.

(2) A licensee shall obtain a license amendment pursuant to § 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:

(i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated);

(ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated);

(iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated);

(iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated);

(v) Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated);

(vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated);

(vii) Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered; or

(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses

(3) In implementing this paragraph, the FSAR (as updated) is considered to include FSAR changes resulting from evaluations performed pursuant to this section and analyses performed pursuant to § 50.90 since submittal of the last update of the final safety analysis report pursuant to § 50.71 of this part.

(4) The provisions in this section do not apply to changes to the facility or procedures when the applicable regulations establish more specific criteria for accomplishing such changes.

(d) (1) The licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test or experiment does not require a license amendment pursuant to paragraph (c)(2) of this section.

(2) The licensee shall submit, as specified in § 50.4, a report containing a brief description of any changes, tests, and experiments, including a summary of the evaluation of each. A report must be submitted at intervals not to exceed 24 months.

(3) The records of changes in the facility must be maintained until the termination of a license issued pursuant to this part or the termination of a license issued pursuant to 10 CFR Part 54, whichever is later. Records of changes in procedures and records of tests and experiments must be maintained for a period of 5 years.

Appendix B

10 CFR 72.48 Changes, tests, and experiments.

(a) Definitions for the purposes of this section:

(1) *Change* means a modification or addition to, or removal from, the facility or spent fuel storage cask design or procedures that affects a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished.

(2) *Departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses* means (i) changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same; or (ii) changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application.

(3) *Facility* means either an independent spent fuel storage installation (ISFSI) or a Monitored Retrievable Storage facility(MRS).

(4) The *facility or spent fuel storage cask design as described in the Final Safety Analysis Report (FSAR) (as updated)* means:

(i) The structures, systems, and components (SSC) that are described in the FSAR (as updated),

(ii) The design and performance requirements for such SSCs described in the FSAR (as updated), and

(iii) The evaluations or methods of evaluation included in the FSAR (as updated) for such SSCs which demonstrate that their intended function(s) will be accomplished.

(5) *Final Safety Analysis Report (as updated)* means:

(i) For specific licensees, the Safety Analysis Report for a facility submitted and updated in accordance with § 72.70;

(ii) For general licensees, the Safety Analysis Report for a spent fuel storage cask design, as amended and supplemented; and

(iii) For certificate holders, the Safety Analysis Report for a spent fuel storage cask design submitted and updated in accordance with § 72.248.

(6) *Procedures as described in the Final Safety Analysis Report (as updated)* means those procedures that contain information described in the FSAR (as updated) such as how SSCs are operated and controlled (including assumed operator actions and response times).

(7) *Tests or experiments not described in the Final Safety Analysis Report (as updated)* means any activity where any SSC is utilized or controlled in a manner which is either:

(i) Outside the reference bounds of the design bases as described in the FSAR (as updated) or

(ii) Inconsistent with the analyses or descriptions in the FSAR (as updated).

(b) This section applies to:

(1) Each holder of a general or specific license issued under this part, and

(2) Each holder of a Certificate of Compliance (CoC) issued under this part.

(c) (1) A licensee or certificate holder may make changes in the facility or spent fuel storage cask design as described in the FSAR (as updated), make changes in the procedures as described in the FSAR (as updated), and conduct tests or experiments not described in the FSAR (as updated), without obtaining either (i) A license amendment pursuant to § 72.56 (for specific licensees) or (ii) A CoC amendment submitted by the certificate holder pursuant to § 72.244 (for general licensees and certificate holders) if:

(A) A change to the technical specifications incorporated in the specific license is not required; or

(B) A change in the terms, conditions, or specifications incorporated in the CoC is not required; and

(C) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section.

(2) A specific licensee shall obtain a license amendment pursuant to § 72.56, a certificate holder shall obtain a CoC amendment pursuant to § 72.244, and a general licensee shall request that the certificate holder obtain a CoC amendment pursuant to § 72.244, prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:

(i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR (as updated);

(ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a system, structure, or component (SSC) important to safety previously evaluated in the FSAR (as updated);

(iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR;

(iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR (as updated);

(v) Create a possibility for an accident of a different type than any previously evaluated in the FSAR (as updated);

(vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR (as updated);

(vii) Result in a design basis limit for a fission product barrier being exceeded or altered as described in the FSAR (as updated); or

(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.

(3) In implementing this paragraph, the FSAR (as updated) is considered to include FSAR changes resulting from evaluations performed pursuant to this section and analyses performed pursuant to §§ 72.56 or 72.244 since the last update of the FSAR pursuant to §§ 72.70, or 72.248 of this part.

(4) The provisions in this section do not apply to changes to procedures when the applicable regulations establish more specific criteria for accomplishing such changes.

(d) (1) The licensee and certificate holder shall maintain records of changes in the facility or spent fuel storage cask design, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license or CoC amendment pursuant to paragraph (c)(2) of this section.

(2) The licensee and certificate holder shall submit, as specified in § 72.4, a report containing a brief description of any changes, tests, and experiments, including a summary of the evaluation of each. A report shall be submitted at intervals not to exceed 24 months.

(3) The records of changes in the facility or spent fuel storage cask design shall be maintained until:

(i)Spent fuel is no longer stored in the facility or the spent fuel storage cask design is no longer being used, or

(ii)The Commission terminates the license or CoC issued pursuant to this part.

(4) The records of changes in procedures and of tests and experiments shall be maintained for a period of 5 years.

(5) The holder of a spent fuel storage cask design CoC, who permanently ceases operation, shall provide the records of changes to the new certificate holder or to the Commission, as appropriate, in accordance with § 72.234(d)(3).

(6) (i) A general licensee shall provide a copy of the record for any changes to a spent fuel storage cask design to the applicable certificate holder within 60 days of implementing the change.

(ii) A specific licensee using a spent fuel storage cask design, approved pursuant to subpart L of this part, shall provide a copy of the record for any changes to a spent fuel storage cask design to the applicable certificate holder within 60 days of implementing the change.

(iii) A certificate holder shall provide a copy of the record for any changes to a spent fuel storage cask design to any general or specific licensee using the cask design within 60 days of implementing the change.

13

Modifications to the Reactor Safety Goal Policy Statement

Presentation to ACRS

Joseph A. Murphy
Office of Nuclear Regulatory Research
February 3, 2000

Reactor Safety Goal Policy Statement

Background

- SECY -99-191 informed the Commission of progress in developing recommendations regarding modifications to Safety Goal Policy Statement and proposed feasibility study of overarching safety principles.
- Related SRM stated staff should provide a recommendation regarding the policy statement, but disapproved study of the feasibility of developing overarching principles.

Relationship Between Safety Goals and Regulations

- Regulations establish requirements which enable the agency to conclude there is no undue risk to the public health and safety.
- Policy statements provide a high level expression of the safety philosophy and expectations of the agency.
- Safety Goal Policy Statement, coupled with the PRA Policy Statement, provides a foundation for risk-informed regulation of reactors.
- Safety Goal Policy Statement provides the overall targets for safety when considering modification to existing regulations or the addition of new regulations.

Changes to Reflect Current Policy

Proposed Recommendations

- Incorporate five principles from R.G. 1.174 (generalized to reflect broader usage) into Regulatory Implementation portion of the policy statement.
- Incorporate positions taken in 6/15/90 SRM that safety Goals establish a level of safety considered safe enough and that they represent a risk level to strive for, utilizing the provisions contained in the Backfit Rule.
- Provides foundation for risk-informed regulation.

Five Generalized Principles

- Plants are expected to meet current regulations and any applicable exemptions.
- The defense-in-depth philosophy should be maintained.
- Sufficient safety margins should be maintained.
- Where changes in risk might occur, increases in risk or core damage frequency should be small.
- Plant performance should be monitored.

Treatment of Core Damage Frequency as a Fundamental Goal

Proposed Recommendations

- Elevate qualitative statement in policy statement presented below to status of a qualitative goal (with editing).
  - "...the Commission intends to continue to pursue a regulatory program that has as its objective providing reasonable assurance, while giving appropriate consideration to the uncertainties involved, that a severe core damage accident will not occur at a U.S. nuclear power plant."
- Retain CDF of 10^{-4}/RY as a subsidiary objective and include it in the policy statement.
  - Coupled with LERF, provides practical implementation guidance for QHOs.

Treatment of Uncertainty

Staff Recommendation

- Uncertainty is discussed at some length in policy statement.
- Update discussion of uncertainty in policy statement to reflect the guidance provided in R.G. 1.174.

Defense in Depth

- Current policy statement discusses importance of prevention and mitigation.
- Defense in depth included in five principles from R.G. 1.174
- Note ACRS/ACNW ongoing efforts.
- Incorporate statement on use of Defense in depth from Commission's White Paper

Defense in depth

White Paper

- *"Risk insights can make the elements of defense-in-depth more clear by quantifying them to the extent practicable. Although the uncertainties associated with the importance of some elements of defense may be substantial, the fact that these elements and uncertainties have been quantified can aid in determining how much defense makes regulatory sense. Decisions on the adequacy of or the necessity for elements of defense should reflect risk insights gained through identification of the individual performance of each defense system in relation to overall performance."*

Safety Goal Structure

- ACRS in 5/11/98 letter stated:
  - "The current Policy Statement specifies only a single goal for each objective...An upper limit and a goal define three regions. For risk levels above the upper limit, immediate action should be taken. For risk levels between the upper limit and the goal, the possibility of reducing the estimated metric should be investigated, taking into account costs and benefits. For risk levels below the goal, no action would be required"

Safety Goal Structure and Adequate protection

Backfit Rule

- Structure proposed by ACRS is similar to framework in Backfit Rule (50.109)
  - Backfits required if necessary to ensure adequate protection (a)(5)
  - Backfits allowed if substantial increase in overall protection and costs are justified by increased protection (a)(3)
  - Backfits not allowed because cannot pass tests above.
- Safety Goals help define lower limit since they were used in deriving the Regulatory Analysis Guidelines.

Safety Goal Structure

Adequate Protection

- SECY-99-246 noted "risk estimates serve as an important measure of plant safety, but do not embody the full range of considerations that enter into the judgment regarding adequate protection. The judgment regarding adequate protection derives from a more diverse set of considerations, such as acceptable design, construction, operation, maintenance, modification, and quality assurance measures, together with compliance with NRC requirements including, license conditions, orders, and regulations"

Land Contamination and Overall Societal Impact

- Significant portion of population dose comes from ground shine and ingestion. Strongly affected by protective measures.
- Calculations in NUREG-1150 based on EPA Protective Action Guides. Relocation if projected 1st year dose exceeds 2 rem or any succeeding year exceeds 0.5 rem.
- Current Level 3 PRA tools have significant weaknesses that limit utility of predictions at significant distances from the plant.

Land Contamination and Overall Societal Impact

Recommendations

- Add qualitative goal on protecting the environment. Note CDF and LERF provide a level of protection.
- Land contamination is considered in Regulatory Analysis Guidelines, based on NUREG-1150 results and in EISs.
- Development of improved tools will be considered in the planning and budgeting process.

Temporary Changes in Risk

- The existing safety goal states
  - *The Commission's first qualitative safety goal is that the risk from nuclear power plant operation should not he a significant contributor to a person's risk of accidental death or injury.*
- Temporary changes are covered in principle.
- Clarify applies to temporary changes as well as annual average risk.

(14)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
469th	5.0	5-1

Title: ACRS Proposed Review of the Commission's Safety Goal Policy Statement for Reactors

Author: P. BOEHNERT

List of Documents Attached

- ACRS Review of Proposed Revisions to Commission Safety Goal Policy Statement - Specific Recommendations

5

Instruction to Preparer	From Staff Person
2. Paginate Attachments 3. Punch holes 4. Place Copy in file box	P. BOEHNERT

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

February 2, 2000

MEMORANDUM TO: ACRS Members

FROM: P. Boehnert, Senior Staff Engineer

SUBJECT: ACRS REVIEW OF PROPOSED REVISIONS TO COMMISSION SAFETY GOAL POLICY STATEMENT - SPECIFIC RECOMMENDATIONS

The ACRS will review the proposed reversions to the Commission Safety Goal Policy Statement (SGPS) during its February 2000 meeting. Below, I have provided some detail on the specific NRC staff recommendations concerning revision of the SGPS. As noted below, formal comment summarizing the status of the ACRS review of this matter appears appropriate.

ACRS Staff Engineer M. Markley provided you a copy of the SGPS Paper, via Express Mail, upon its receipt in this Office last week. Mike also prepared the handout in the meeting Notebooks including a copy of the Paper, given my attendance at the Retreat in Florida. Please note that this Paper is considered **DRAFT PREDECISIONAL**.

BACKGROUND

Over the past few years the staff and the ACRS have identified several issues that point to a need to revise the Commission's Safety Goal Policy Statement (SGPS). In response, the Commission directed, via a October 16, 1997 SRM, that the staff provide a recommendation on the need for revision of the Policy Statement.

In April 1999, the Committee provided comment on the status of the staff's efforts regarding revision of the SGPS (copy of ACRS report attached). The Committee supported expeditious revision of the Policy Statement. It also expressed support for the staff's plan to develop an "overarching" Policy Statement for all NRC regulated activities, but the Committee was not of one mind with regard to the objectives, scope, utility, feasibility, or schedule for development of this Policy Statement. The staff subsequently issued SECY-99-191 (copy attached) to provide the Commission a status report on development of recommendations regarding possible modifications of the SGPS and to outline a feasibility study for development of an overarching Policy Statement. The staff committed to providing its recommendations to the Commission by March 30, 2000.

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its objective that a severe core-damage accident will not occur at a U.S. nuclear power plant to the status of a qualitative goal, and. (2) retain a quantitative value of 10-4 per reactor-year as a useful subsidiary performance objective.

- **Treatment of Uncertainty**

  Reference in the Policy Statement (Section IV) the appropriate section of Regulatory Guide 1.174 (Section 2.2.5), with incorporation of the more general portions as appropriate.

- **Defense-in-Depth**

  Incorporate the relative portion of the Commission's White Paper dealing with the role of defense-in-depth in a risk-informed regulatory framework (page 7 of Paper). The staff also takes note that **"the ACRS and ACNW are developing additional recommendations to the Commission in the area of defense-in-depth"** (emphasis added).

- **Safety Goal Structure**

  The staff quotes from the ACRS's May 11, 1998 letter (discussed in the Committee's April 19, 1999 letter attached to the handout in the Committee Notebook), regarding the Committee's recommendations relative to revision of the SGPS. The staff recommends that no change be made to the SGPS relative to the Committee's recommendations in this regard. The argument made is that a structure similar to that recommended by the Committee already exists in the regulations and other implementing documents. Consistent with Commission guidance in the SECY-99-191 SRM, the staff will consider revising the "reasonable assurance of adequate protection" guidance as well as regulatory and backfit analyses after experience is gained with use of risk information in regulatory practices.

- **General Performance Guideline for Frequency of a Large Release of Radioactive Material**

  Delete reference to the "General Performance Guideline" (a housekeeping chore as this effort was terminated via SECY-93-138), and incorporate a LERF subsidiary goal of 10-5 per reactor year (consistent with ACRS recommendation in its May 11, 1998 report).

- **Societal Risk**

  Two issues were considered: (1) should the Policy Statement or the Regulatory Analysis Guidelines be modified?, and (2) should these two documents be made consistent

relative to treatment of societal dose? The staff argued that both documents serve different purposes and the use of a 10-mile zone in the SGPS and a 50-mile zone in the Guidelines should not be changed.

- **Land Contamination and Overall Societal Impact**

  The staff recommends that no additional safety goal be developed for this area. The arguments for not doing so (page 12) seem to rest on the lack of adequate tools (PRA's and IPEs) for determination of the extent of land contamination and the resulting societal impact. It is stated that development of the needed tools will be considered pursuant to the Agency's normal Planning, Budgeting, and Performance Management process.

- **Temporary Changes in Risk**

  The staff noted in SECY-99-191 that consideration should be given to the impact of temporary changes in plant risk arising from equipment failures, maintenance activities, and human actions. The staff position is that such temporary changes are already covered in principle. To make it clear, however, the staff suggests the SGPS be clarified regarding its applicability to changes in equipment operability and configuration.

COMMITTEE DISCUSSION

Mr. J. Murphy, RES, is scheduled to make a presentation to the Committee on this matter. In addition, Mr. B. Bradley, NEI, has requested time to make a brief (10 minute) set of comments to the Committee on this matter.

COMMITTEE ACTION

As noted above, the staff has a deadline of March 30, 2000 to provide its recommendation to the Commission on whether to modify the SGPS. Given this, the Committee could postpone providing formal comment on this matter until the March meeting, should the press of business during this meeting so dictate.

cc: R. Savio

cc w/o attach (via E-mail):
J. Larkins
H. Larson
S. Duraiswamy
ACRS Technical Staff & Fellows

(15)

Review of Power Uprate Applications and Potential Synergistic Safety Issues

August W. Cronenberg
ACRS Fellow

Outline

- Power Uprate Overview
  Applications, NRC Review Procedures, Altered Plant Conditions
- Events Noted for Uprated Plants
- Potential Synergistic Safety Issues
- Observations & Recommendations

Summary of Power Uprate Applications

Plant	Reactor Type	Year Startup	Original Power(MWt)	Year Power Uprate	Uprated Power(MWt)	% Power Increase
Oyster Creek	BWR	1969	1690	1971	1930	14.20
Calvert Cliffs-1	PWR	1977	2560	1977	2700	5.46
Main Yankee	PWR	1972	2440	1978	2630	7.79
Millstone-2	PWR	1975	2560	1979	2700	5.47
Fort Calhoun	PWR	1973	1420	1980	1500	5.63
St. Lucie-1	PWR	1976	2560	1981	2700	5.46
Cook-2	PWR	1978	3391	1983	3411	0.56
Duane Arnold	BWR	1975	1593	1985	1658	4.08
St. Lucie-2	PWR	1983	2560	1985	2700	5.47
Salem-1	PWR	1977	3338	1986	3411	2.19
North Anna	PWR	1978	2775	1986	2893	4.25
Callaway	PWR	1985	3411	1988	3565	4.51
Main Yankee*	PWR	1972	2440	1989	2700	10.65
Indian Point-2	PWR	1974	2758	1990	3071	11.35
Fermi-2	BWR	1987	3293	1992	3458	5.01
Wolf Creek	PWR	1985	3411	1993	3565	4.51
Vogel 1&2	PWR	1987	3411	1993	3565	4.51
Peach Bottom-2	BWR	1974	3293	1994	3458	5.01
Susquehanna 1&2	BWR	1983	3293	1994	3441	4.49
Surry 1&2	PWR	1972	2441	1995	2546	4.30
Nine Mile-2	BWR	1988	3323	1995	3467	4.33
Hatch 1&2	BWR	1975	2436	1995	2558	5.00
Limerick-2	BWR	1988	3293	1995	3458	5.01
Limerick-1	BWR	1985	3293	1996	3458	5.01

* Denotes second power uprate, percent power uprate based on original power level.

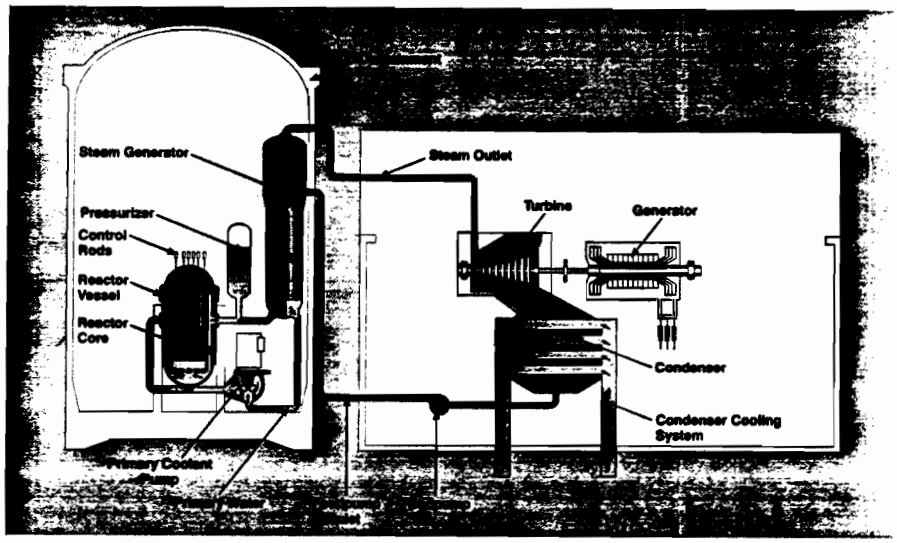
Slide 3

Licensee Uprate Responsibilities

- Uprate Application takes form of License Amendment Request
- Application Must Include Power Uprate Safety Analysis Report (SAR)
- SAR Generally Follows GE-BWR Generic Guidance (Series of Licensing Topical Reports through-1996) or W-PWR Generic Review Plan (WCAP-10263, 1983)
- SAR Centers on Re-evaluation of Design Basis Accidents, No Significant Hazards Assessment, Changes to Plant Conditions and Technical Specifications

Slide 4

Plant Characteristics Impacted by Power Uprates



Primary Coolant System (PCS)
Core Power
Core Inlet/Outlet Enthalpy
Vessel Outlet/Inlet Coolant Temperatures
Fuel Temperature
Primary Coolant Flow Rate

Secondary Coolant System (SCS)
Steam Generator Steam Flow Rate
Feedwater Flow Rate
Feedwater Temperature
Feedwater Pumping Requirements

NRC Uprate Responsibilies

- NRC staff reviews licensee's uprate SAR
- NRC reports findings in Safety Evaluation Report (SER)
- NRC review centers on an evaluation of the impact of power increase on plant operations and safety
- No agency uprate Standard Review Plan or standardized acceptance criteria
- No agency requirements for independent code analysis of plant uprate conditions

Power Uprate Events

- Maine Yankee: Deliberate/Faulty LOCA Analysis by Licensee
- Wolf Creek/North Anna: Control Rod Insertion Problems
- Callaway/Surry: Guillotine Breaks in Feedwater Line
- Brunswick: Faulty use of DBA Criteria
- Limerick: Predicted $\Delta K/K$ less than measured

Maine Yankee

- Allegations of deliberate/ faulty LOCA submittal, DBA clad limit of 2200°F exceeded for uprate conditions.
- LOCA analysis performed with altered decay heat & critical flow models.
- NRC-SER did not question licensee analysis. No independent analysis performed by NRC staff.
- Subsequent investigations concurred with allegations.

LESSON: Need for independent NRC staff analysis (code calculations) for uprate reviews.

Wolf Creek / North Anna

Wolf Creek: 4.5-% Uprate 1996
Control rod insertion problems for 5 high burnup rods (47 GWD).

North Anna: 4.3-% Uprate 1986
2 control rod assemblies fail to fully insert in high burnup (47-49 GWD) assemblies in spent fuel pool.

NRC-SER: Neither Uprate SER addressed potential changes in fuel or control rod performance for high-burnup/high-power conditions.

LESSON: Need for NRC staff review of potential synergistic high-burnup/high-power effects.

Pipe Ruptures

- Surry, Callaway, Tsuruga, etc
  (53 rupture events pipe Diam. > 2")
- Erosion (flow)/ Corrosion (age) mechanism a major cause of ruptures
- Empirical evidence for synergistic flow/aging effects

LESSON: Need for NRC staff review of potential synergistic flow/aging effects

A Publication of The McGraw-Hill Companies

Incorporating NUCLEONICS

NUCLEONICS WEEK

Vol. 40 No. 51 December 23, 1999

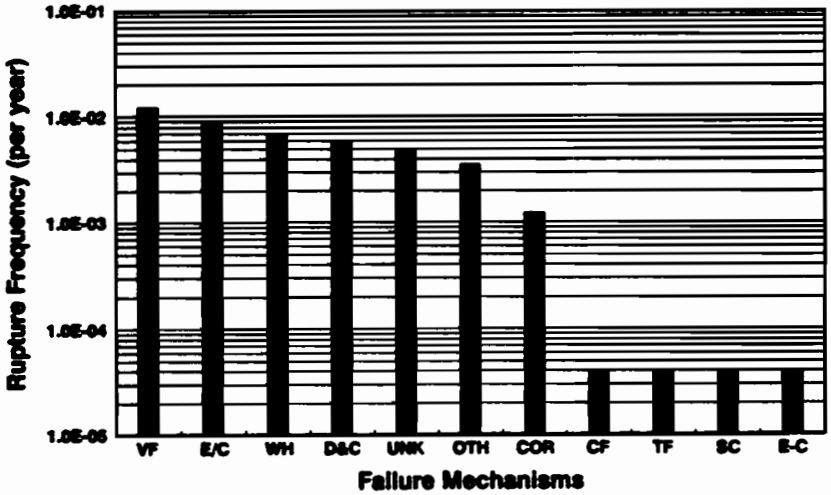
SUSQUEHANNA SHUTDOWN MAY SIGNAL NEW GENERIC PROBLEM FOR BWRs

" A recent Susquehanna-2 forced outage could be the result of weld fatigue from increased vibrations from a power uprate in 1995, and NRC is looking at potential generic implications for other uprated BWRs..."

" **BWR uprates have increased the speed** of recirculation pumps and caused increased vibrations in the recirculation systems, said Sam Hansell, NRC resident inspector at Susquehanna."

Slide 11

Frequencies of Pipe Ruptures Mechanisms Through 1995



VF	Vibrational Fatigue	COR	Corrosion Attack
E/C	Erosion/Corrosion	CF	Corrosion Fatigue
WH	Water Hammer	TF	Thermal Fatigue
D&C	Design & Construction	SC	Stress Corrosion Cracking
UNR	Unreported Cause	E-C	Erosion -Cavitation
OTH	Others (human error, frozen pipes, etc.)		

Slide 12

Other Events

Brunswick 1 & 2 (BWR)

- Licensee DBA Analysis based on wet-well design limit of 220°F.
- NRC did not challenge 220°F analysis.
- Correct value of 200°F (Licensee).

Limerick-1 (BWR)

- Code predicted $\Delta K/K$ for restart less than measured for high-power/high-burnup core.
- Code inadequacies for high-power/high-burnups.

LESSON: Need for agency uprate SRP and assessment of adequacy of core physics codes for high-power/high-burnup/high enrichments.

Potential Synergistic Effects

High-power/High-Burnup Synergisms

Rod Fretting: Flow induced rod vibration leading to contact wear with adjacent structures. Increased core flow at uprated power and Zry-irradiation growth for high burnup may lead to fretting.

Axial Power Offset: Boron added to compensate for excess reactivity of high-enrichment/high-burnup rods. Crud buildup for long fuel duty times. Boron gettered by crud, leads to axial power offset. Effect is compounded at high-power core locations.

Potential Synergistic Effects

High-Power/Ageing Synergisms

Electrical Components: Insulation breakdown due to irradiation effects, exacerbated by elevated temperatures.

Fluid -Mechanical Components (pumps, piping, valves): Subject to flow assisted corrosion (FAC). Elastomer seal degradation at high temperatures.

Instrumentation & Control Systems: Ageing of I&C exacerbated at elevated temperatures and fluid velocities (vibration).

Observations & Recommendations

- Current uprate application/ review process largely centers on re-evaluation of DBA conditions at uprated power. Nil consideration of potential synergistic effects.
- Events show indirect evidence for synergistic power-uprate/ageing and power-uprate/burnup effects.
- NRC Standard Review Plan (SRP) for power uprates should be issued (in progress). SRP should include acceptance criteria for synergistic effects.
- NRC uprate reviews should include staff T-H and core physics code calculations to verify licensee analysis.
- NRC uprate reviews should include comparison of safety measures (CDF, QHO, LERF) for prior & uprate power.

Risk-Informed Regulation - Challenges and Importance Measures

Presented to:
Advisory Committee on Reactor Safeguards

Presented by:
Tom King, RES
Gary Holahan, NRR
Marty Virgilio, NMSS

February 4, 2000

(6)

Risk-Informed Regulation: Key Elements

- Policy
- Strategy (road map)
- Informed, knowledgeable staff
- Decision making
- Tools
- Communication

Implementation

Element	Activities	Challenges
• Policy	– PRA Policy Statement – Reactor Safety Goal Policy – SECY-99-100 (NMSS)	– development of safety goals for non-reactor activities – voluntary vs mandatory (two different regulatory approaches)
• Strategy	– PRA Implementation Plan – Risk-informed Regulation Implementation Plan	– what should be RI? – how much of the industry will utilize risk-informed approach?
• Informed, Knowledgeable Staff	– training – SRA Program	– staffing (level and capabilities) – how much NRC prior review and approval is necessary?

Implementation (Cont'd)

Element	Activities	Challenges
• Decision-Making	– guidance documents (e.g. RG 1.174) – plant oversight + enforcement – risk-informing Part 50 – risk-informing non-reactor activities	– selective implementation – guidance for non-reactor activities
• Tools	– analytical tools – risk assessment methods – PRA standard – data bases	– lack of completeness of risk assessments – lack of standard – adequacy of tools and methods
• Communication	– pilot programs – stakeholder meetings – communications plan	– perception that risk-informed equals burden reduction

Role of Importance Measures in Decision-making

- Primary role of Importance Measures is to perform an initial identification of risk significant SSCs or groups of SSCs
- Risk significance is confirmed by sensitivity analyses and qualitative assessment
- Safety principle that only "small changes" in risk are allowed can be met by
  - calculating ΔCDF and ΔLERF, or
  - by showing that only SSCs of low significance are being addressed and the impact of the change on them is relatively small. This approach can allow a qualitative judgement that the change in risk is small.

Importance Measures

- The staff guidance on using importance measures is presented in Appendix A of Reg Guide 1.174, and Appendix C of the Standard Review Plan, Chapter 19.
  - input to the categorization of SSCs with respect to safety significance
- Issues related to the use of Importance Measures include:
  - Technical issues related to completeness of PRAs, truncation, treatment of implicitly modeled SSCs, common cause failures, etc.
  - Importance measures are typically evaluated for individual SSCs, whereas changes are typically at a programmatic level (groups of SSCs)
  - Importance measures do not translate simply to changes in risk metrics (CDF and LERF)

Importance Measures (Cont'd)

- The Staff's conclusions with respect to these issues can be summarized as follows:
  - "Most of the issues can be resolved by the use of sensitivity analyses or by appropriate quantification techniques"
  - "When performed and interpreted correctly, component-level importance measures can provide valuable input..."
  - "The criteria for categorization into low and high safety significance should be related to the acceptance criteria for changes in CDF and LERF"
  - "If component-level criteria are used they should be established taking into account that the allowable risk associated with the change should be based on simultaneous changes to all members in the category."

Risk-Informing 10CFR50

TopEvent Prevention (TEP)
A Deterministic Application of PSA

Advisory Committee on Reactor Safeguards
Bethesda, Md.
February 4, 2000

CMS ENERGY

Outline

- What is TEP and how does it work?
- Results of actual applications
- Implementation at CMS Energy
- Consistency with current Industry and NRC guidance

What Is Top Event Prevention Analysis (TEP)?

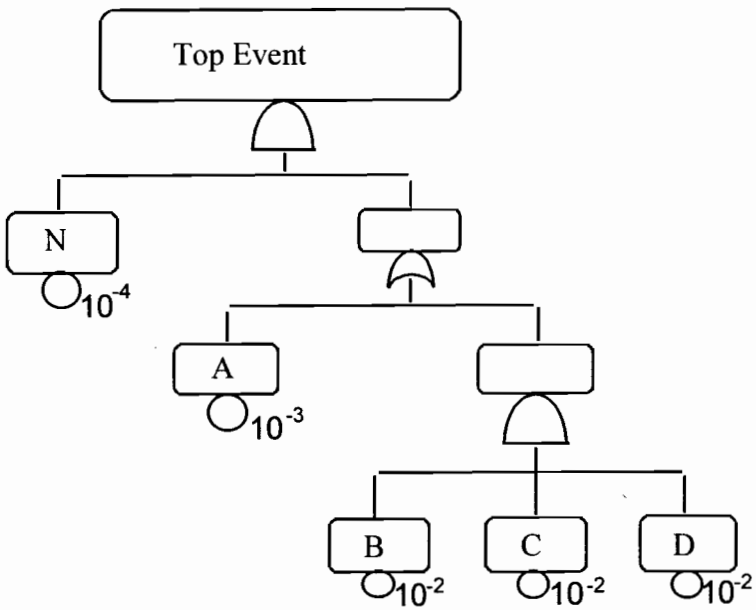
TEP is a deterministic application of your PRA

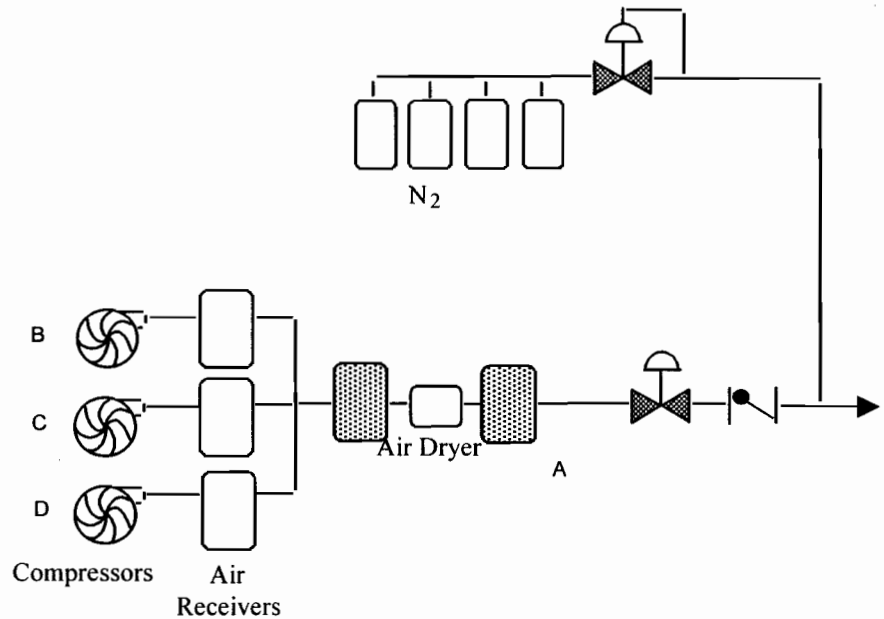
- Performs a detailed defense-in-depth review of your PRA
  - Cut set by cut set
- Identifies complete sets of success paths necessary to preclude the occurrence of the top event (i.e., CDF, LERF,etc.)
  - Evaluates the whole plant (all functions, all sequences, all components)
  - Assures consistency between risk-informed programs
- Addresses a number of limitations normally attributed to PSA methods
  - Combinatorial problems
  - Truncation

What is Top Event Prevention Analysis (TEP)?

- TEP identifies the ***minimum*** combinations of events important to the PRA results
  - It is this minimum set of components on which plant resources should be focused from a safety perspective (maintenance, testing, QA).
- TEP generates multiple ***options*** from which to choose
  - If the different options (or combinations of events) can be ranked against one another (such as by cost of maintenance, testing or QA), then the least cost optimal solution can be chosen to focus plant resources.

Top Event Prevention Analysis
Simple Example





10^{-7} N * A
10^{-10} N * B * C * D

	N	A	B	C	D
Fussell-Vesely	1.0	~1.0	10^{-3}	10^{-3}	10^{-3}
Risk Achievement Worth	10^{4}	10^{3}	1.1	1.1	1.1

Top Event

N * A +
N * B * C * D.

Prevention Sets Level 2

(N*A) *
(N*B + N*C + N*D +
B*C + B*D + C*D).

Minimal Prevention Sets

N * A * B +
N * A * C +
N * A * D.

Top Event Prevention Analysis
Practical Example (Check Valve Testing)

Steps in the Process

- Build and solve model for core damage (> 80,000 cutsets)
  - 170 Check Valves modeled

- Prevention Level (2)
  - Credited Events (~2,000)
  - Excluded Events (~ 200)

- For each cutset, generate expression that represents prevention by 2 credited events

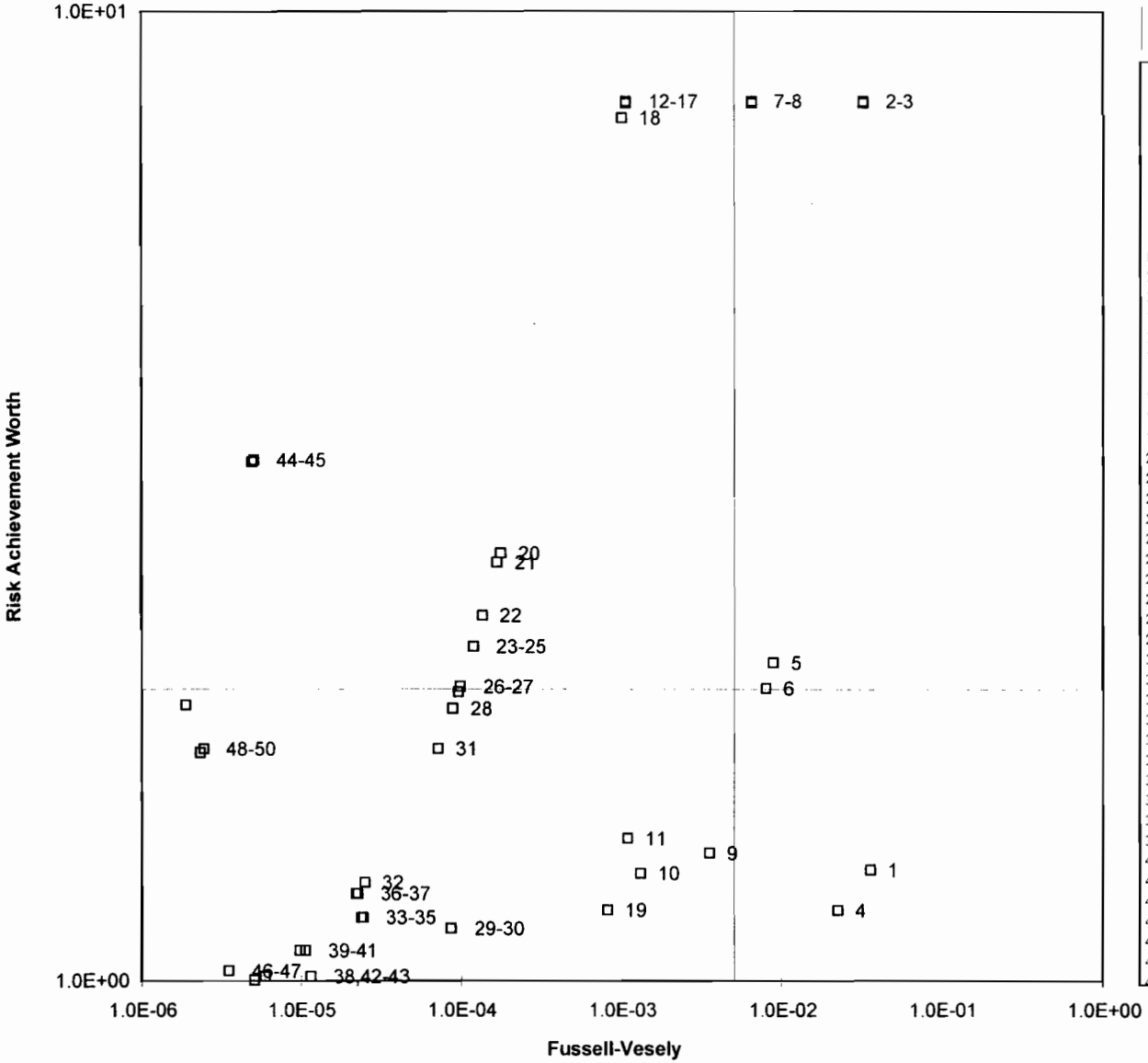
- Form Boolean product
  - Expand
  - Simplify (55,000 Prevention Sets)

Palisades PSA Check Valve Sensitivity Study

	Accident Class	Base Case	Prevention Set (1)	Importance Measures(2)
IA	Transient with failure of OTC injection	1.85E-05	1.94E-05	4.31E-04
IB	Transient with failure of OTC recirc	1.08E-05	1.98E-05	3.01E-03
II	Containment heat removal failure	1.31E-07	2.98E-07	6.55E-03
IIIA	LOCA at hi RX pressure with failure of injection	7.75E-06	9.38E-06	2.09E-04
IIIB	LOCA at hi rx pressure with failure of injection	6.88E-06	9.07E-06	1.53E-02
IIIC	LOCA at low rx press with failure of recirc	3.53E-07	5.13E-07	1.62E-04
IIID	LOCA at low rx pressure with failure of recirc	3.53E-07	5.13E-07	1.62E-04
IV	ATWS	4.35E-06	6.46E-06	1.46E-04
VB	Steam generator tube rupture	1.39E-06	1.87E-06	2.53E-04
		5.03E-05	6.69E-05	

Note (1) Select one prevention set and fail all check valves <u>not</u> in that prevention set

Check Valve Importance Measures



Check Valves

1 CK263 Inst Air to ESF Recirc
2 CK3166 Sump to ESS Suction
3 CK3181 Sump to ESS Suction
4 CK427 Inst Air to ESF Recirc
5 CK410 Hi Pres Air to ESF Recirc
6 CK428 Hi Press Air to ESF Recirc
7 CK3332 HPSI Recirc to SIRWT
8 CK3331 HPSI Recirc to SIRWT
9 CK2161
10 CK414
11 CK402
12 CK3183 HPSI Suction
13 CK3186 HPSI Discharge
14 CK3340 HPSI Recirc to SIRWT
15 CK3339 HPSI Recirc to SIRWT
16 CK3168 HPSI Suction
17 CK3177 HPSI discharge
18 CK3411 HPSI Injection to Vessel
19 CK426
20 CK743 AFW Pump Discharge
21 CK400 Makeup to CST
22 CK402 Steam to AFW Turbine
23 CK2456
24 CK2457
25 CK2459
26 CK726 AFW pump discharge
27 CK413
28 C425
29 CK401
30 CK450
31 Ck741
32 CK725
33 CK3131
34 CK3146
35 CK3116
36 CK728
37 CK729
38 CK2171
39 CK3132
40 CK3147
41 CK3117
42 CK402
43 CK2105
44 CK403 Air Compressor Discharge
45 CK405 Air Compressor Discharge
46 CK3216

Top Event Prevention Analysis Summary of Other Applications

Analysis	# of Prevtn Sets	# of comp / plant	# of comp modelled in the PRA	# of comp in minimum prevention set	Change in CDF (Note 1)
BWR4 MOVs	21	~100	80	14	1.8
PWR MOVs	12	~200	80	26	1.9
MCCB (fail to trip)	7000	100-200	30(2)	2	1.1
PWR check valves	55,000	Hundreds	170	58	1.3
BWR4 AOVs	Millions	~1000	70	16	1.1
PWR AOVs	842	~1000	50	27	2.7
PWR Pumps	Millions	Dozens	36	16	1.7

Notes
(1) Change in CDF after all components not in the selected prevention are set to 1.0 (Failed)
(2) Super components added to model after cut sets were generated

Top Event Prevention Advantages

Addresses Combinatorial Problems

The combinations of events important to the PRA results are identified, not simply one event at a time.

Component importance is not masked by contributions from other sequences - e.g., conservative results from external events.

Component importance is not insensitive to multiple layers of redundancy.

Variables that are important in combination are identified explicitly and can be reviewed for common factors.

Addresses Truncation Issues

The cut sets that have been truncated from the original results can be tested.

Allows Presentation to Plant Staff/Expert Panel in Terms of the Plant Design

Explict identification of accident sequences for which a component is important.

Where a component is not considered important, what is being credited instead can be explicitly identified

Top Event Prevention Limitations

Limited Use of Probabilities

All credited components are treated equally, components within a prevention set are not ranked in any way

Completeness/Accuracy

Prevention sets are dependent on the completeness of the models and their accuracy with respect to the manner in which the plant will respond to accident and transient conditions.

Top Event Prevention Analysis
Consistency with Regulatory Guidance

- Elements of Regulatory Guide 1.174
  - ➢ Defense in depth

    TEP performs a defense in depth analysis for Core damage/Large early release
  - ➢ ΔCDF/ΔLERF low

    Demonstrate based on artificially high failure rates for low risk significant components (not a part of the prevention set)
  - ➢ Implement monitoring program

    Assign very relaxed performance criteria for low risk significant components (not a part of the prevention set)

Top Event Prevention Analysis

Would the ACRS and NRC staff consider a risk-informed submittal based on TEP as meeting the intent of

- R.G. 1.174
- R.G. 1.175 (IST)
- R.G. 1.176 (QA)
- R.G. 1.178 (ISI)
- The ANPR?

Top Event Prevention Analysis References

The theory behind the TEP methodology is presented in the paper

"Top Event Prevention in Complex Systems",
R. W. Youngblood and R.B. Worrell,
PVP-Vol. 296/SERA-Vol. 3, 1995.

Application of the TEP methodology is presented in a number of papers

"Using Top Event Prevention Analysis to Select a Safety-Significant Subset of Check Valves for Testing"
R.A. White(Palisades), R.B. Worrell and D.P. Blanchard
TopSafe '98, 1998

"Use of Top Event Prevention Analysis to Select a Safety-Significant Subset of Air-Operated Valves for Testing"
C.F. Nierode(Monticello), R.B. Worrell and D.P. Blanchard
PSAM 4, 1998

"Identification of Risk-Significant Circuit Breakers Using Top Event Prevention Analysis"
B.A. Brogan(Big Rock Point), R.B. Worrell and D.P. Blanchard
PSA '96, 1996

Importance Measures (Issues/Alternatives)

- Importance measures can identify what is important, but do not necessarily identify what is not important.
- Importance measures are among a number of acceptable tools to get to an answer, but are not the answer themselves.
- Regardless of the methods used, sensitivity studies should be performed following classification of components into risk-significance categories to confirm the classification.
- Given these sensitivity studies, it does not matter how the classification was performed or whether importance measures were used. It only matters that the collection of components selected for special treatment are effective in managing risk.

CMS ENERGY

Importance Measures (Alternatives/Sensitivity Studies)

Probabilistic

Given realistic estimates of changes in reliability for affected components, is the change in risk acceptable?

or

Given bounding assumptions for the reliability of affected components, is the risk acceptable?

Deterministic

Are there accident sequences dependent on low risk significant components for defense-in-depth?

CMS ENERGY

Impedements to the Use of Risk-Informed Regulation

There continues to be significant uncertainty regarding what it costs and how long it takes to get approval for a Risk-Informed submittal

- Even post-pilot plant submittals have taken significant resources and have had to deal with generic issues

The PSA Quality issue requires significant effort as compared to technical details associated with a Risk-Informed submittal

There is significant effort needed to meet the detailed requirements for a Risk-Informed submittal from a quantitative standpoint

- The focus of current guidance seems to be on quantitative results as opposed to making decisions based on the insights generated from the quantitative results

CMS ENERGY

(19)

IMPEDIMENTS TO RISK-INFORMED REGULATION AND RISK IMPORTANCE MEASURES

Presented to
Advisory Committee on Reactor Safeguards

Presented by
Thomas G. Hook
Manager, Nuclear Safety Oversight
San Onofre Nuclear Generating Station

February 4, 2000

1

IMPEDIMENTS TO RISK-INFORMED REGULATION

- Difficulty in quantifying costs and benefits
- Variations in PRA quality and scope
- Regulatory review process (e.g., RAIs and duration)
- Lack of PRA standards to establish quality
- PRA staffing inadequacies
- Current PRA focus on MRule and SDP
- Previous pilots marginally successful
- Insufficient credit for certification/Owners Group peer reviews

2

IMPORTANCE MEASURES

- RAW & Fussell-Vesely/RRW are acceptable for screening
- Importance measures that evaluate extrema (0,1) are acceptable only when augmented by sensitivity analyses
- Uncertainty analysis is underutilized by most licensees
- Sensitivity analyses should include model requantification with all impacted parameters (e.g., SSC reliability /availability, HRA, init event freq.) adjusted to bounding values
- Should consider SSC safety functions for all scopes, plant operating modes, and initiating events before reducing SSC quality requirements
- Generally concur with draft ANPR (Appendix T)

3

(20)

IMPEDIMENTS TO RISK INFORMED REGULATION

STP Nuclear Operating Company

Comments to ACRS

February 4, 2000

REGULATORY IMPEDIMENTS

- No regulatory quantitative limits for establishing the importance/non-importance of components
- No differentiation between design basis events and events that are likely to occur during plant life (i.e., design basis events vs. operational basis events)
- Due to some of the cultural impediments (noted below), resolution of some RAI inquiries may be difficult. Opportunities for gaining experience and lessons learned may be lost.
- No mechanism or path to change safety-related component classifications based on risk information
- Lack of clarity or criteria in the degree to which risk informed applications could be approved using only qualitative approaches versus those that use quantitative methods or both.
- No tangible and recognizable incentives for developing, approving, and implementing risk informed applications
- Lack of defined roles and responsibilities for organizations tasked with developing risk informed processes, up to and including approval, in accordance with Option 3 of the regulations identified for risk informing 10CFR50

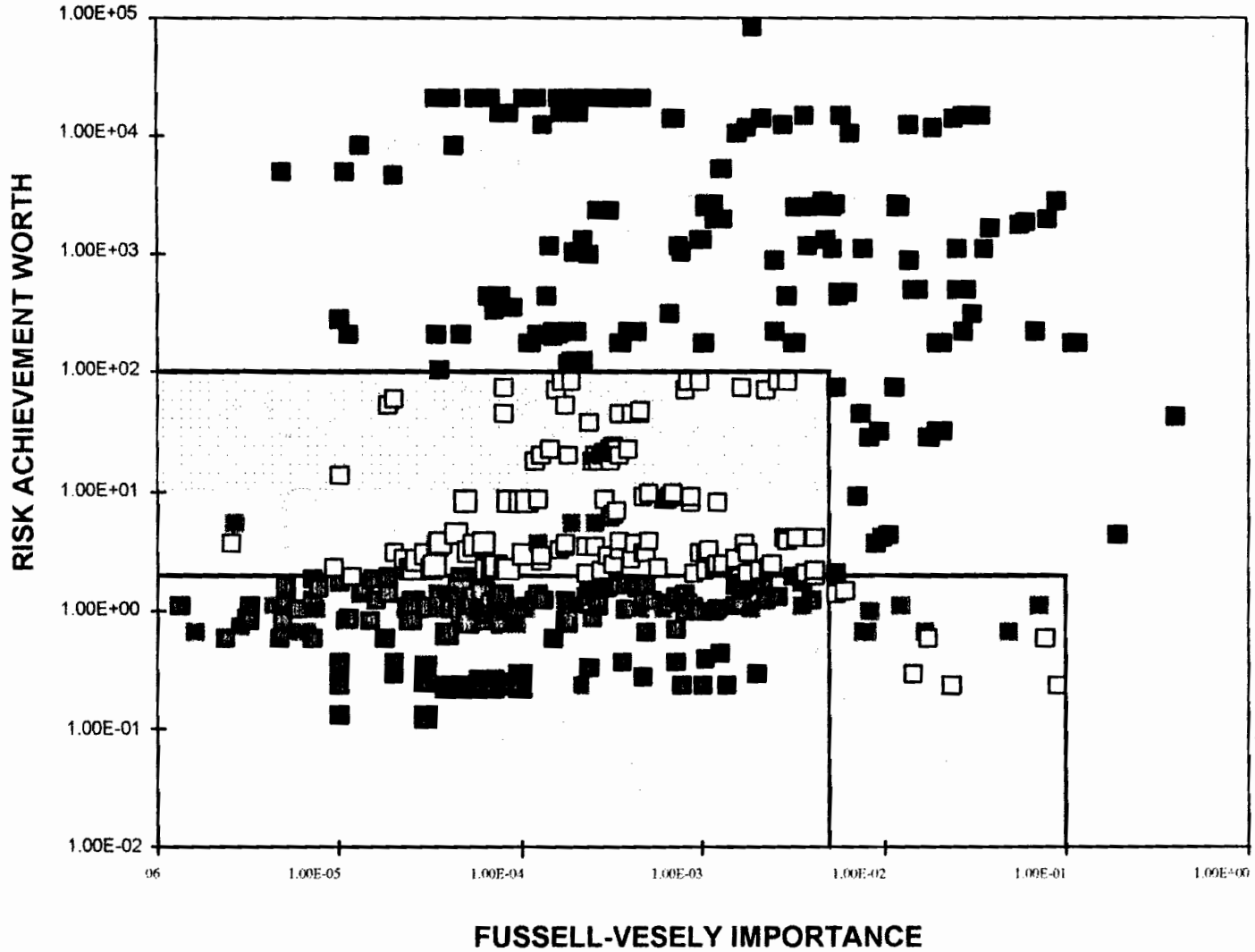
CULTURAL IMPEDIMENTS

- Lack of training that demonstrates the complementary effect of blending deterministic and probabilistic approaches in decision-making
- Reluctance to "let go" of selected activities on safety-related low or non-risk significant components even on a pilot basis
- Lack of data or other pertinent studies on the reliability/availability of safety-related components versus non safety-related components
- Resistance to change (turf protection) for both regulator and utilities
- Quickness to declare victory (i.e., implementation of a risk informed application) with a limited risk informed application (e.g., testing frequency changes without scope changes)
- Attempts to develop risk informed applications that are solely qualitative
- Misconception that PRA analyses are too expensive relative to the benefits
- Misconception that PRA analyses are unproven technology
- Weaknesses in the understanding of PRA at management levels in some utility organizations
- Need for improvements to and formalization of the training, organization, and oversight of plant expert panels

PRA INSTITUTIONAL IMPEDIMENTS

- Limited availability of PRA practitioners for both regulator and utilities
- Delays in legitimizing the PRA discipline (ASME/ANS standards)
- Resolving probabilistic approaches against institutional requirements (e.g., ASME Code, IEEE, NFPA, 10CFR50 Special Treatment requirements, etc.)
- Risk Ranking methods need further development
- PRA Importance Measures need further development
- PRA Uncertainty Analyses need further development
- Human and organizational analyses need further development

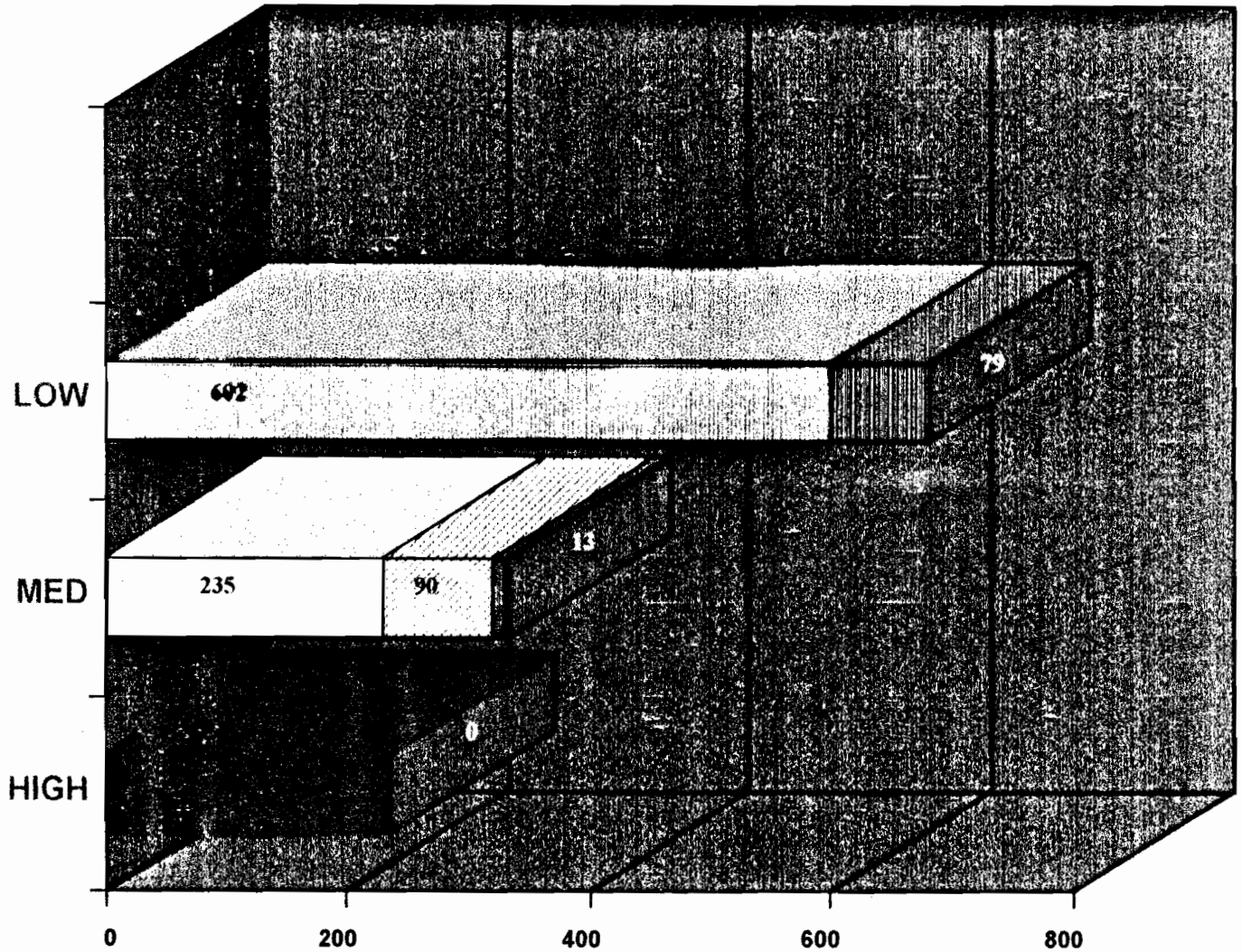
RISK RANKING OF ALL PSA BASIC EVENTS



HIGH Full QA	■	RAW>2.0 and FV>0.005 **OR** RAW>100 **OR** FV>0.1
MED Basic QA	□	RAW>2.0 and <100 FV<0.005 **OR** RAW<2.0 and FV>0.005
LOW Basic QA	▒	RAW>2 and FV<0.005
LOW MED HIGH	■	Non-Safety Related
Full + Basic (Critical Attributes Only)	▨	RAW>10 and <100

2

TOTAL BASIC EVENTS RISK RANKING



HIGH Full QA	■	RAW>2.0 and FV>0.005 **OR** RAW>100 **OR** FV>0.1
MED Basic QA	□	RAW>2.0 and <100 FV<0.005 **OR** RAW<2.0 and FV>0.005
LOW Basic QA	□	RAW>2 and FV<0.005
LOW MED HIGH	□	Non-Safety Related
Full + Basic (Critical Attributes Only)	□	RAW>10 and <100

Risk Ranking Sensitivity Studies

Average

Description: Quantify the at-power nominal average operation PRA separately for Core Damage Frequency and Large Early Release Frequency.

Scheduled Maintenance

Description: Quantify the at-power nominal average operation PSA separately for each one of the scheduled-maintenance state types modeled under top event GENST in event tree PMET.

Purpose: Provides risk ranking with respect to equipment unavailability due to preventive maintenance activities.

Recovery

Description: Quantify available and appropriate risk models based on the removal of all operator recovery actions.

Purpose: Provides risk ranking with primary emphasis on equipment availability, reliability, and removes credit for human intervention.

Common Cause Failures

Description: Quantify available and appropriate risk models based on the removal of all common cause failure contributions.

Purpose: Provides focus of risk ranking based equipment combinations outside the scope of common cause failures.

Multi-System Effects of a Component Type

Description: For selected component types common to more than one system that are low risk and candidates for changes in QA requirements, vary the failure rates and requantify the models.

Purpose: To determine the impact of changes that affect more than one system.

Large Early Release Frequency

Description: Set split fraction ISS to one half it's normal value and reevaluate LERF.

Purpose: LERF is dominated by induced steam generator tube rupture, which is a highly uncertain phenomenon. This change considers the effect of reducing the assumed failure rate on risk ranking.

N130[illegible]tal
TAG/TPNS

Risk Ranking Sensitivity Studies (0PGP01-ZA-0304)

Avg. Model

	Level 1 Sensitivity Studies																			Level 2			
	Planned Maintenance States													Inc. Failure Rate								Comp	Final
UNIT 1 TAG/TPNS	GN1	GN2	GN3	GN4	GN5	GN6	GN7	GN8	GN9	GN10	PM1	PM2	PM3	MS2	MS5	MS10	NCC	REC	STP	LER	STPL2	Rank	Rank
2N121NPA102C	H	H	M-R	H	H	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M	M-R	M-R	T	H	H	H
2N121NTF101A	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	T	M-R	H	H
2N121TSI0011A	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	M	H	M-R	L	M-R	H	H
2N121TSI0011B	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	M	H	M-R	L	M-R	H	H
2N121TSI0011C	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	M	H	M-R	L	M-R	H	H
2N121TSI0012A	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	M	H	M-R	L	M-R	H	H
2N121TSI0012B	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	M	H	M-R	L	M-R	H	H
2N121TSI0012C	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	M	H	M-R	L	M-R	H	H
2N121TSI0013A	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	M	H	M-R	T	M-R	H	H
2N121TSI0013B	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	L	H	M-R	T	M-R	H	H
2N121TSI0013C	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	M	H	M-R	T	M-R	H	H
2N121TSI0014A	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	M	H	M-R	T	M-R	H	H
2N121TSI0014B	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	L	H	M-R	T	M-R	H	H
2N121TSI0014C	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	M-R	H	H	H	M-R	M-R	M-R	M	H	M-R	T	M-R	H	H
2N121TSI0104A	L	M	M	L	M	M	L	L	L	L	L	L	L	L	L	L	L	L	L	T	L	M	M
2N121TSI0104B	M	L	M	M	L	M	L	L	L	L	L	L	L	L	L	L	L	L	L	T	L	M	M
2N121TSI0104C	M	M	L	M	M	L	L	L	L	L	L	L	L	L	M	M	L	L	L	T	L	M	M
2N121TSI0206A	L	L	L	L	L	L	L	L	L	M	M	M	M	M	M	M	M	M	M	L	L	M	M
2N121TSI0206B	M	L	L	L	L	L	L	L	L	M	M	M	M	M	M	M	M	M	M	L	L	M	M
2N121TSI0206C	M	L	L	L	L	L	L	L	L	M	M	M	M	M	M	M	M	M	M	L	L	M	M
2N121TSI0207A	T	M	M	T	M	M	L	M	L	L	L	L	L	M	M	M	M	L	M	T	M	M	M
2N121TSI0207B	M	T	M	M	T	M	L	L	L	L	L	L	L	L	L	L	L	L	L	T	L	M	M
2N121TSI0207C	M	M	T	M	M	T	L	L	L	L	L	L	L	L	L	L	M	L	L	T	L	M	M
2N121XSI0001A	L	M	M	L	M	M	M	M	M	M	M	M	M	M	M	M	M	M	M	T	L	M	M
2N121XSI0001B	M	M	M	M	M	M	M	M	M	M	M	M	M	M	M	M	M	M	M	T	L	M	M
2N121XSI0001C	M	M	L	M	M	L	M	M	M	M	M	M	M	M	M	M	M	M	M	T	L	M	M
2N121XSI0002A	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	M	H	H	T	M-R	H	H
2N121XSI0002B	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	M	H	H	T	M-R	H	H
2N121XSI0002C	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	H	M	H	H	T	M-R	H	H
2N121XSI0004A	L	L	L	L	L	L	L	L	L	M	M	M	M	M	M	M	M	M	M	L	L	M	M
2N121XSI0004B	M	L	L	L	L	L	L	L	L	M	M	M	M	M	M	M	M	M	M	L	L	M	M
2N121XSI0004C	M	L	L	L	L	L	L	L	L	M	M	M	M	M	M	M	M	M	M	L	L	M	M

BEID	GN1	GN2	GN3	GN4	GN5	GN6	GN7	GN8	GN9	GN10	PM1	PM2	PM3	NCC	MS2	MS5	MS10	REC	STP	LER	STPL2	Final
CV_PMPS_FTR_X01A	M	M	M	M	M	M	L	L	L	L	L	L	L	M	M	L	L	L	L	L	L	M
CV_PMPS_FTS_X01A	H	H	M	H	H	H	L	L	L	L	L	L	L	M	M	L	L	L	L	L	L	H

DETERMINATIONS OF RISK

Five Critical Questions Addressed:

- **Inititating Event** - Does loss of function, in and of itself, directly cause an initiating event?
- **Fails Risk Significant System** - Does loss of function directly fail **another** risk significant system?
- **Accident/Transient** - Is function used to mitigate accidents or transients?
- **EOPs** - Is function specifically called out in EOPs and/or Emergency Response Procedures?
- **Shutdown/Mode Change** - Is loss of function safety significant for shutdown or mode change activities?

DETERMINATIONS OF RISK

Answers:

"0"	-	Negative Response
"1"	-	Positive response having insignificant impact and/or occurring very rarely
"2"	-	Positive response having minor impact and/or occurring infrequently
"3"	-	Positive response having low impact and/or occurring occasionally
"4"	-	Positive response having medium impact and/or occurring regularly
"5"	-	Positive response having high impact and/or occurring frequently

DETERMINATIONS OF RISK

Weighting Factors:

Accidents/Transients, EOPs	-	5
Fails Risk Sig. System	-	4
Initiating Event, Shutdown/Mode Change	-	3

Score Range	Risk
0 - 20	NRS
21 - 40	Low
41 - 70	Medium
71 - 100	High

SYS_ID	COMP_ID	COMP_DESC	FE_CLAS	_RISK_RA	DETERMINISTIC_INPUT	L_RISK_RANK
CV	2C091NPN053A	MECHANICAL PENETRATION|CVCS RC FILTER TO RHR PUMP	2		CONTAINMENT ISOLATION	LOW
CV	2R171NFR101A	REACTOR COOLANT FILTER 1A	2		FILTER COLLECTS CV AND BTRS DEMINERALIZER RESIN FINES AND PARTICULATES LARGER THAN 25 MICRONS. IN LETDOWN FLOWPATH. HOWEVER, REDUNDANT FILTER AVAILABLE. ALSO BYPASS LINE AVAILABLE	LOW
CV	2R171NFR101B	REACTOR COOLANT FILTER 1B	2		FILTER COLLECTS CV AND BTRS DEMINERALIZER RESIN FINES AND PARTICULATES LARGER THAN 25 MICRONS. IN LETDOWN FLOWPATH. HOWEVER, REDUNDANT FILTER AVAILABLE. ALSO BYPASS LINE AVAILABLE	LOW
CV	2R171NFR102A	SEAL INJECTION FILTER 1A	2	L	FILTER COLLECTS PARTICULATES THAT COULD BE HARMFUL TO THE RCP SEALS. REDUNDANT FILTER AVAILABLE	LOW
CV	2R171NFR102B	SEAL INJECTION FILTER 1B	2	L	FILTER COLLECTS PARTICULATES THAT COULD BE HARMFUL TO THE RCP SEALS. REDUNDANT FILTER AVAILABLE	LOW
CV	2R171NPD101A	SUCTION PULSATION DAMPER HTR PD101A	2		SUCTION STABILIZER FOR THE POSITIVE DISPLACEMENT CHARGING PUMP. ACTS TO MAINTAIN NPSH AT THE PUMP INLET TO PREVENT CAVITATION	LOW
CV	2R171NPD102A	PULSATION DAMPENER	2		ATTENUATES PRESSURE PULSATIONS AT THE DISCHARGE OF THE POSITIVE DISPLACEMENT CHARGING PUMP	LOW

SYS_ID	COMP_ID	COMP_DESC	FE_CLAS	_RISK_RA	DETERMINISTIC_INPUT	L_RISK_RA
CV	N1CVPA102A	CVCS POSITIVE DISPLACEMENT|CHARGING PUMP MOTOR|TPNS: 2R171NPA102A	7S	H	PRIMARILY USED FOR HYDROTESTING THE RCS. PROVIDES A MEANS FOR ADDING CHEMICALS TO THE RCS FOR pH AND OXYGEN CONTROL. PROVIDES SEAL INJECTION FLOW IF BOTH CCPs ARE INOPERABLE	HIGH
CV	N2CVPA202A	CVCS POSITIVE DISPLACEMENT|CHARGING PUMP MOTOR|TPNS: 2R172NPA202A	7S	H	PRIMARILY USED FOR HYDROTESTING THE RCS. PROVIDES A MEANS FOR ADDING CHEMICALS TO THE RCS FOR pH AND OXYGEN CONTROL. PROVIDES SEAL INJECTION FLOW IF BOTH CCPs ARE INOPERABLE	HIGH
RH	N1RHFY3860	RHR HEAT EXCHANGER 1A OUTLET VALVE FV-3860 CURRENT/PNEUMATIC CONVERTOR	7S		RHR HEAT EXCHANGER FLOW CONTROL: THE PNEUMATIC TRANSDUCER (FY) RECEIVES AN ANALOG ELECTRICAL SIGNAL FROM A HAND CONTROLLER IN THE CONTROL ROOM AND CONVERTS THE ELECTRICAL SIGNAL TO A PNEUMATIC SIGNAL TO PROVIDE FOR THE POSITIONING OF AN AIR OPERATEDBUTTER	HIGH
RH	N1RHFY3861	RHR HEAT EXCHANGER 1B OUTLET VALVE FV-3861 CURRENT/PNEUMATIC CONVERTOR	7S		RHR HEAT EXCHANGER FLOW CONTROL: THE PNEUMATIC TRANSDUCER (FY) RECEIVES AN ANALOG ELECTRICAL SIGNAL FROM A HAND CONTROLLER IN THE CONTROL ROOM AND CONVERTS THE ELECTRICAL SIGNAL TO A PNEUMATIC SIGNAL TO PROVIDE FOR THE POSITIONING OF AN AIR OPERATEDBUTTER	HIGH
RH	N1RHFY3862	RHR HEAT EXCHANGER 1C OUTLET VALVE FV-3862 CURRENT/PNEUMATIC CONVERTOR	7S		RHR HEAT EXCHANGER FLOW CONTROL: THE PNEUMATIC TRANSDUCER (FY) RECEIVES AN ANALOG ELECTRICAL SIGNAL FROM A HAND CONTROLLER IN THE CONTROL ROOM AND CONVERTS THE ELECTRICAL SIGNAL TO A PNEUMATIC SIGNAL TO PROVIDE FOR THE POSITIONING OF AN AIR OPERATEDBUTTER	HIGH
RH	N1RHHC0864	RHR HEAT EXCHANGER 1A CONTROL	7S		THE MANUAL CONTROL STATION PROVIDES REMOTE MANUAL CONTROL OF THE TRAIN A RHR HEAT EXCHANGER FLOW CONTROL VALVE FROM THE CONTROL ROOM OR THE AUX SHUTDOWN PANEL. THIS VALVE DOES NOT PERFORM A SAFETY FUNCTION. HOWEVER, THE VALVE IS NORMALLY OPEN AND FAILS OP	HIGH

(22)

APPENDIX K RULEMAKING

Final Rule Change
Revising the 102% Power Level Requirement

Briefing to the
Advisory Committee on Reactor Safeguards

February 4, 2000

Joe Donoghue, SRXB (301) 415-1131

Appendix K Revision

Background

- ACRS briefings on May 26 and July 14, 1999
- ACRS letter to EDO on July 22, 1999
- Staff response to ACRS on August 18, 1999
- Proposed rule change published for comment on October 1, 1999
- Public comment period ended December 15, 1999
- Concurrence process on final rule is underway

Appendix K Revision

Comments on Proposed Rule Change

- 6 responses - Caldon, NEI, and licensees
- All responses were positive
  - NRC actions when uncertainty is above 2%
  - Apparent requirement for upgraded instruments
  - §50.46 reportability of changes to ECCS analysis
- Clarifications added to Federal Register notice
  - Upgraded instrumentation not required
  - §50.46 reportability not affected
- Proposed rule change language not modified

Appendix K Revision

Conclusion

- No adverse public comments to proposed rule change
- Final rule change package will reflect public comments and will implement the proposed rule change
- Request ACRS endorsement on final rule change

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No:
469th	14	14.1

Title RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

Authors SAM DURAISWAMY

List of Documents Attached

See attached list

14

Instructions to Preparer
1. Punch holes
2. Paginate attachments
3. Place copy in file box

From Staff Person
SAM DURAISWAMY

SUBJECT	EDO LTR.	ACRS LTR.	ANALYSIS
Proposed Rev. 3 to R.G. 1.160 (DG-1082), "Assessing and Managing Risk Before Maintenance Activiat NPPs" (JJB/AS) OK	12/2/99 p.1	11/12/99 pp. 2-3	2/1/00 p.4
Resolution of Generic Issue (GI)-148, "Smoke Control & Manual Fire-Fighting Effectiveness" (JJB/AS) OK	12/15/99 pp. 5-6	11/12/99 pp. 7-8	2/1/00 p.9
Spent Fuel Fires Associated With Decommissioning(TSK/MME) OK	12/16/99 pp.10-11	11/12/99 pp.12-14	1/28/00 pp.15-16
Draft Comm. Paper Re. 120-Month Update Requirement for Inservice Inspection & Inservice Testing Programs (WJS/NFD) NO	01/13/00 pp.17-20	12/08/99 pp.21-23	1/31/2000 pp.24-25
Proposed Resolution of GSI 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life" (RLS/AS) OK	01/13/00 p.26	12/10/99 pp.27-28	2/1/00 p.29
NUREG-1624, "Rev. 1, "Technical Basis and Implementation Guidelines for a Technique for Human Event Analysis (ATHENA)" (GA/NFD) No response required; OK	01/20/00 pp.30-32	12/15/99 pp.33-37	2/1/00 p.38
Implementing a Framework for Risk-Informed Regulation in the Office of Nuclear Material Safety and Safeguards (TSK/MTM) OK	01/20/00 pp.39-41	11/17/99 pp.42-45	2/3/00 p.46
Report on the Safety Aspects of the License Renewal Application for Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (MVB/NFD) OK	01/24/00 pp.47-49	12/10/99 pp.50-53	2/1/00 P.54

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

December 2, 1999

Dr. Dana A. Powers, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PROPOSED REVISION 3 TO REGULATORY GUIDE 1.160 (DG-1082), "ASSESSING AND MANAGING RISK BEFORE MAINTENANCE ACTIVITIES AT NUCLEAR POWER PLANTS" (G19990578)

Dear Dr. Powers:

I am forwarding the staff's response to the recommendations of the Advisory Committee on Reactor Safeguards (ACRS) on the proposed draft maintenance rule guidance, DG-1082, transmitted by your letter dated November 12, 1999.

ACRS Recommendations:

1. The proposed Revision 3 to Regulatory Guide 1.160 should be issued for public comment.

   Response: The staff agrees with the recommendation. DG-1082 and NEI's Final Draft Section 11 of NUMARC 93-01 which DG-1082 endorses are being printed for public availability and comment at this writing.

2. We support the staff's endorsement of the NEI guidance for industry use when revised to incorporate the staff's comments and to provide a concise definition of unavailability.

   Response: The requested changes have been made.

Please let me know if the Committee has any further questions or comments on the proposed rule.

Sincerely,

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Diaz
Commissioner Merrifield
OGC
Commissioner Dicus
Commissioner McGaffigan
SECY
OPA
OCA

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

November 12, 1999

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

Dear Chairman Meserve:

SUBJECT: PROPOSED REVISION 3 TO REGULATORY GUIDE 1.160 (DG-1082), "ASSESSING AND MANAGING RISK BEFORE MAINTENANCE ACTIVITIES AT NUCLEAR POWER PLANTS"

During the 467th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1999, we reviewed the proposed Revision 3 to Regulatory Guide 1.160 (DG-1082), "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," and the revised draft of Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI). We previously commented on an earlier version of this guide in a report dated July 21, 1999.

Recommendations

1. The proposed Revision 3 to Regulatory Guide 1.160 should be issued for public comment.

2. We support the staff's endorsement of the NEI guidance for industry use when revised to incorporate the staff's comments and to provide a concise definition of unavailability.

Discussion

Both the staff and NEI agree that the proposed Revision 3 to Regulatory Guide 1.160 and NUMARC 93-01 provide an acceptable method for assessing and managing the increase in risk that may result from nuclear power plant maintenance activities, as required by new paragraph (a)(4) of 10 CFR 50.65. The guidance that the staff and NEI have developed resolves our concerns that we raised in our report of July 21, 1999.

There are three minor issues between the staff and NEI that we were assured would be resolved easily. In addition, the definition of unavailability in the draft Regulatory Guide needs to be clarified. The description of unavailability provided in Appendix B of the proposed

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modification to NUMARC 93-01 is not a definition. The commonly accepted definition of the unavailability of a system that is under periodic surveillance testing is simply the average fraction of time during which the system is incapable of performing its intended function. The equation in Appendix B is correct, if "required operational hours" is interpreted as the period of surveillance tests.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References :

1. Memorandum dated October 18, 1999, from Theodore R. Quay, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Request for Review of Draft Regulatory Guidance for 10 CFR 50.65, The Maintenance Rule.
2. Final Draft of Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated October 8, 1999.
3. Report dated July 21, 1999, from Dana A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, NRC, Subject: Proposed Revision 3 to Regulatory Guide 1.160 (DG-1082), "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants."

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

February 1, 2000

MEMORANDUM TO: John J. Barton, Chairman
Plant Operations Subcommittee

FROM: Amarjit Singh, P.E., Senior Staff Engineer
ACRS/ACNW Technical Support Staff

SUBJECT: ANALYSIS OF THE EDO RESPONSE TO THE ACRS REPORT REGRADING PROPOSED REVISION 3 TO REGULATORY GUIDE 1.160 (DG-1082), "ASSESSING AND MANAGING RISK BEFORE MAINTENANCE ACTIVITIES AT NUCLEAR POWER PLANTS"

The purpose of this memorandum is to provide you with an analysis of EDO response to the Committee's report dated November 12, 1999, regarding the proposed maintenance rule guidance, DG-1082, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants".

ACRS Recommendations:

1. The proposed Revision 3 to Regulatory Guide1.160 should be issued for public comment.

EDO Response:

The staff agrees with the recommendation. DG-1082 and NEI's Final Draft Section 11 0f NUMARC 93-01 which DG-1082 endorses have been issued for public comment.

2. ACRS support the staff's endorsement of the NEI guidance for industry use when revised to incorporate the staff's comments and to provide a concise definition of unavailability.

EDO Response:

The staff has made the requested change in the revised guidance document.

Analysis:

The EDO response was responsive and had made changes as recommended by ACRS. Therefore, I suggest that the Committee consider the response satisfactory.

cc: J. Larkins
H. Larson
S. Duraiswamy

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

December 15, 1999

Dana A. Powers, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESOLUTION OF GENERIC ISSUE (GI)-148, "SMOKE CONTROL AND MANUAL FIRE-FIGHTING EFFECTIVENESS"

Dear Dr. Powers:

The conclusion and recommendation in your November 12, 1999, letter on the above subject stated that (1) the ACRS concurs with the staff's proposal for resolving GI-148, and (2) Section 3 of Reference 1 should be revised to include guidance on addressing the effects of smoke on manual fire fighting. Based on (1) the staff's proposed resolution that was presented to the ACRS on September 30, 1999, (2) clarification of the content and application of existing guidance for the plant-specific IPEEE reviews, discussed below, and (3) ACRS concurrence with this resolution, GI-148 is resolved.

The ACRS recommended that Section 3 of the staff guidance on smoke control and manual fire-fighting effectiveness (Reference 1) that was developed for the plant-specific Individual Plant Evaluation of External Events (IPEEE) reviews be revised "to include guidance on addressing the effects of smoke on manual fire fighting." The staff has carefully reviewed the guidance provided in Reference 1 in light of the Committee's comment. As discussed in Reference 1, the potentially significant aspects of the effects of smoke on manual fire fighting are: (1) smoke may hamper the firefighters' effectiveness by causing access problems to the affected fire zone, or by causing difficulties in actually locating the fire within the zone, and (2) once firefighters reach the zone, misdirected suppression efforts could subsequently damage equipment not directly involved in the fire.

With regard to the first item, problems with accessing or locating the fire zone, the guidance in Section 3 of Reference 1 provides curves on the probability of successful manual suppression of fires as a function of time (measured from the time of fire ignition) for five critical plant areas. These curves, which are consistent with actual data for time to successfully extinguish a fire measured from time of ignition, implicitly take into account the probability that smoke may cause access problems or difficulty in locating the fire in the zone. The guidance in Section 3 states that the reviewer should compare the times for manual suppression used in the plant-specific IPEEEs with these generic curves. (We note that most IPEEE submittals have taken little or no credit for manual suppression efforts beyond the incipient stage of a fire.) Recognizing that current fire PRA methodology does not explicitly address the effect of smoke on manual fire fighting, these curves are reasonable to use for IPEEE fire analyses. The IPEEE reviewers are aware of this and look for instances in which excessive credit has been taken for manual suppression. In fact, the staff has requested additional information from licensees in instances where their analyses assumed unrealistically short times to successfully suppress a fire.

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With regard to item 2, collateral damage to equipment near the fire from misdirected suppression efforts, the guidance in Section 3a of Reference 1 states that the reviewer should determine if the IPEEE takes into account the possibility that fire-fighting efforts could result in firewater damage to adjacent trains of electrical equipment located near those affected by the fire. Although "the effects of smoke" is not explicitly stated in this section of the guidance, our reviews of plant specific IPEEEs have included the recognition that the effects of smoke could cause misdirected suppression.

Smoke control/removal, an item very closely related to this GI, has been addressed in previously issued staff guidance (i.e., "Fire Protection Program," Section 9.5.1 of NUREG-0800, "Standard Review Plan") and is included in the draft comprehensive fire protection regulatory guide (RG) DG-1094 that was issued for public comment in October 1999. Plant fire brigades are equipped with breathing apparatus, portable lighting and smoke removal equipment. For some plant areas, the building HVAC system has a smoke control mode. The state-of-the-art of fire PRAs is not capable of evaluating the effect of any additional mitigative systems or features to address the smoke concern. The staff believes that the guidance provided in the draft RG is adequate to provide reasonable assurance of plant safety.

In conclusion, we believe that the existing guidance in Reference 1 already includes the effects of smoke on manual fire fighting. Therefore, based on the staff's proposal, ACRS concurrence with the staff's proposal, and the clarification of existing guidance discussed above, we consider GI-148 closed.

Sincerely,

William D. Travers
Executive Director for Operations

Reference
1. "Review Guidance for Generic Issue 148, 'Smoke Control and Manual Fire-Fighting Effectiveness,'" Attachment to July 22, 1999, memorandum from T. L. King to A. C. Thadani on the subject of GI-148.

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

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

November 12, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE (GSI)-148, "SMOKE CONTROL AND MANUAL FIRE-FIGHTING EFFECTIVENESS"

During the 467th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1999, we completed our review of the proposed resolution of GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness." During our review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

Conclusion and Recommendation

- We concur with the staff's proposal for resolving GSI-148.
- Section 3 of Reference 1 should be revised to include guidance on addressing the effects of smoke on manual fire fighting.

Discussion

Smoke has a major influence on fire brigade response times and can hamper the operators' ability to shut down the plant safely. GSI-148 has been classified as a "licensing issue." The staff proposed that plant-specific reviews be performed to evaluate the significance of this issue. Such reviews have been performed as part of the Individual Plant Examination of External Events (IPEEE) program.

On the basis of IPEEE submittals and fire brigade training programs, and observations made by resident inspectors, the staff believes that smoke control and manual fire-fighting effectiveness have been adequately addressed.

In Reference 1 the staff discusses how smoke can impact plant risk, however, the effects of smoke are not addressed in Section 3 of this document that discusses review guidance for the staff. This section should be revised to include guidance for use by the staff in evaluating the impact of smoke on manual fire-fighting effectiveness.

Licensee assessments have focused on the localized effects of smoke on manual fire fighting. Smoke can spread well beyond the area of generation and create immediate and delayed

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effects on instrumentation and control circuits. These effects of smoke are not being addressed in GSI-148. The Office of Nuclear Regulatory Research is studying the effects of smoke from cable fires on digital electronic circuits. The results of this study should help to assess the potential impact of these effects.

Based on the results of the staff review of IPEEE submittals to date, anticipated revision to Section 3 of Reference 1, and the research activities in the area of smoke propagation, we agree with the staff's proposal to resolve GSI-148.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated July 22, 1999, from Thomas L. King, Office of Nuclear Regulatory Research, NRC, to Ashok C. Thadani, Office of Nuclear Regulatory Research, NRC, Subject: Staff Review Guidance for Generic Safety Issue (GSI) 148, "Smoke Control and Manual Fire-Fighting Effectiveness."
2. U. S. Nuclear Regulatory Commission, Generic Letter No. 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," dated June 28, 1991.
3. SECY-89-170, "Fire Risk Scoping Study: Summary of Results and Proposed Staff Actions," dated June 7, 1989.
4. U. S. Nuclear Regulatory Commission, NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

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

February 1, 2000

MEMORANDUM TO: John J. Barton, Acting Chairman
Fire Protection Subcommittee

FROM: Amarjit Singh, P.E., Senior Staff Engineer
ACRS/ACNW Technical Support Staff

SUBJECT: ANALYSIS OF THE EDO RESPONSE TO THE ACRS LETTER REGRADING PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-148, "SMOKE CONTROL AND MANUAL FIRE-FIGHTING EFFECTIVENESS"

The purpose of this memorandum is to provide you with an analysis of EDO response to the Committee's letter dated November 12, 1999, regarding the staff's proposed resolution of Generic Safety Issue (GSI)-148, "Smoke Control and Manual Fire-Fighting Effectiveness".

ACRS Recommendations:

1. The Committee agreed with the staff's proposed resolution of GSI-148 without any additional requirements.

EDO Response:

The staff agrees with the recommendation and have closed GSI -190 without any additional requirements.

2. ACRS also recommended that Section 3 of Reference 1 should br revised to include guidance on addressing the effects of smoke on manual fire fighting.

EDO Response:

The staff believe that the existing guidance in Reference 1 already includes the effects of smoke on manual fire fighting, but the staff has clarified the existing guidance in its response.

Analysis:

The EDO response was responsive and provided the clarification of the existing guidance as commented by ACRS. Therefore, I suggest that the Committee consider the response satisfactory.

cc: J. Larkins
H. Larson
S. Duraiswamy

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CORRECTED COPY

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

December 16, 1999

Dr. Dana Powers, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington D.C., 20555-0001

SUBJECT: SPENT FUEL FIRES ASSOCIATED WITH DECOMMISSIONING

Dear Chairman Powers:

Thank you for your letter of November 12, 1999, on the views of the Advisory Committee on Reactor Safeguards (ACRS) regarding the staff's technical study on spent fuel pool accident risk at decommissioning plants. The Office of Nuclear Reactor Regulation has reviewed your comments and recommendations related to the deterministic and probabilistic aspects of the staff's study and offers the following response.

We agree with the ACRS statements regarding the uncertainties related to oxidation kinetics and heat rejection mechanisms. Both the resolution to Generic Safety Issue 82 and the technical working group draft report acknowledge that these two modeling areas represent a significant source of uncertainty. Since we do not plan to conduct experimental research on oxidation kinetics in air or using bouyancy-driven natural circulation, we will employ conservative assumptions for decay times to account for the uncertainty.

The ACRS states that the uncertainties in the analysis for the critical temperature for the onset of runaway oxidation need to be quantified. We will perform limited sensitivity calculations to estimate the uncertainty in critical decay time and critical temperature due to uncertainties in oxidation and heat removal mechanisms. If needed, we will adjust the critical temperature and decay time to compensate for the lack of knowledge in these areas.

We agree that probabilistic risk assessments (PRAs) should be as realistic as possible in order to provide appropriate insights to the decision making process. Accordingly, we are reassessing the parameter values used in the PRA model and placing emphasis on human error assumptions. Our preliminary risk assessment was intended to scope out areas of potential concern that would then be discussed among the stakeholders. Since that time, we have continued to refine the analysis based upon receipt of additional information, including industry commitments on how spent fuel pool operations will be conducted in the future.

The ACRS suggested the use of probability distributions rather than point estimates as a better way to address the uncertainties in the risk assessment. We plan to develop an approach consistent with the acceptance guidelines in Regulatory Guide (RG) 1.174. This RG proposes that the mean values of the distributions representing the impact of the parameter uncertainties be used to compare with the acceptance guidelines.

The ACRS agreed with the choice of uncovery to the top of the fuel as an appropriate end state for the PRA consequence analysis. However, the staff remains open to alternative end states that may be more realistic and can be justified. Additionally, the ACRS suggested that an

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acceptable frequency for this end state would be the same as that for a large, early release frequency (LERF) in RG 1.174. While we have not completed our assessment on risk informed decision making, as we continue to develop our approach in accordance with RG 1.174, we will consider using LERF as you suggest.

Your letter states that this issue may be a good candidate for applying the rationalist regulatory approach as was discussed in your May 19, 1999 report. The appropriate regulatory approach will be decided when rulemaking is initiated and we will consider your suggestion at that time. For our study, we are following the guidelines in RG 1.174 with an emphasis on addressing the most safety significant aspects of the issues. Our study should be compatible with both approaches.

We appreciate the insights the ACRS has provided on the spent fuel pool accident study for decommissioning plants.

Sincerely,

William D. Travers
Executive Director for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY
OGC
OCA
OPA
CFO
CIO

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

November 12, 1999

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

Dear Dr. Travers:

SUBJECT: SPENT FUEL FIRES ASSOCIATED WITH DECOMMISSIONING

During the 467th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1999, we reviewed a draft report of a technical study prepared by the NRC staff on the spent fuel pool accident risk at decommissioning plants. During our review, we had the benefit of discussions with representatives of the NRC staff, the Nuclear Energy Institute (NEI), and two members of the public. We also had the benefit of the documents referenced.

Background

The staff discussed with us the status of its ongoing work on this issue. We appreciate the opportunity to provide our views on the direction of this effort at this interim stage.

The staff has formed a Technical Working Group with the objective of assessing the risks associated with spent fuel pools for decommissioning plants. The intent is to assist the Office of Nuclear Reactor Regulation in developing an integrated rule for decommissioning, to provide guidance for interim exemption requirements, and to identify areas where additional work is needed.

Fuel removed from a reactor must be covered with water for cooling until its decay heat generation rate falls below a critical value. Risks posed by fuel stored in a pool arise from the possibility that this water cooling may be lost. The staff has a two-fold approach to evaluating the issues of spent fuel storage: (1) develop estimates of the decay time required to avoid runaway oxidation of spent fuel clad in the event of accidental uncovery, and (2) develop a risk assessment using a broad set of initiating events and using the end-state consequence of uncovery to the top of the fuel.

NEI has interacted with the staff on this effort and has provided a review of the draft report entitled, "Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants." NEI provided us with its assessments. Our understanding of the more substantive issues raised by NEI is:

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1. Conservatism, especially in human error rates, has skewed the preliminary risk insights.

2. The choice of uncovery to the top of the fuel as the endpoint is difficult to relate to public risk. NEI believes that the analyses should be carried all the way to postulated runaway oxidation.

3. The cladding temperature used as the threshold for onset of runaway oxidation is too low.

We also had benefit of the remarks by a member of the public who expressed concern about the:

- Degree of public participation in this effort
- Acceptability to the public of PRA (probabilistic risk assessment) based regulations
- Lack of sufficient margins and defense-in-depth
- Severity of the consequences
- Vulnerability to terrorism
- Applicability of the database used for equipment failures
- Potential for recriticality

Conclusions and Recommendations

1. We agree with the general approach for determining the decay time beyond which runaway oxidation cannot occur. However, an uncertainty analysis related to the oxidation kinetics and the heat rejection mechanisms is needed. The present analysis is limited to relatively low-burnup levels and associated clad hydriding and oxidation. There are no experimental data on the behavior of realistic fuel and cladding under representative conditions. Either very conservative choices will have to be made for decay times or additional experimental research will have to be conducted.

2. We support the staff's approach to developing a decay heat critical temperature for the onset of runaway oxidation. Uncertainties in these analyses need to be quantified and factored into any decisions regarding the required decay time.

3. PRAs should be as realistic as possible. The staff should reevaluate the basis for its choices particularly for human error rates. We agree with the staff's proposal to use expert opinion to validate or modify the human reliability analyses to ensure that the analyses are not overly conservative.

4. Arguments about conservative versus realistic values are aggravated when point estimates are used for the input parameters to the risk assessments. As stated in our December 16, 1997 report, we believe that uncertainties can be best addressed by expressing the inputs as probability distributions rather than point estimates. Such distributions are easier to defend. In addition, the insights to be gained from the risk analysis would greatly benefit if the results were presented as distributions.

5. We agree with the choice of uncovery to the top of the fuel as being an appropriate end state for the PRA consequence analysis. The database on air oxidation kinetics for high-burnup fuel, subsequent fuel damage behavior, and fission product release is too sparse and the uncertainties too great to provide confidence in carrying the analyses any farther. The acceptable frequency of this end point can be based on consideration of the health consequences resulting from postulated fuel failures. Because prompt fatalities cannot be ruled out, we recommend that the acceptable frequency for this end point be the same as that for large, early release frequency in Regulatory Guide 1.174, which is a surrogate for the prompt fatality Safety Goal.

With the choice of uncovery as the end state of the analysis, the uncertainties due to model inadequacies associated with fire risk assessment are not large. We believe that the spent fuel fire issue would be a good candidate for testing the development of a rationalist regulatory approach, as discussed in our May 19, 1999 report.

We look forward to reviewing the staff's progress in this area.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Draft report entitled, "Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants," prepared by NRC Technical Working Group, June 1999.
2. A Review of Draft NRC Staff Report: "Draft Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants," prepared by ERIN Engineering and Research, Inc., for Nuclear Energy Institute, dated August 27, 1999.
3. Draft (undated) EPRI Technical Report, "Evaluation of Spent Fuel Pool Seismic Failure Frequency in Support of Risk Informed Decommissioning Emergency Planning," prepared by Duke Engineering & Services.
4. Letter dated September 3, 1999, from Mr. David A. Lochbaum, Union of Concerned Scientists, to NRC Commissioners, Subject: Inadequately Monitored Spent Fuel Pool Temperature and Operator Response Times at Permanently Closed Plants.
5. 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.
6. ACRS report dated December 16, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Treatment of Uncertainties Versus Point Values in the PRA-Related Decisionmaking Process."
7. ACRS report dated May 19, 1999, from Dana A. Powers, Chairman, ACRS to Shirley Ann Jackson, Chairman, NRC, Subject: The Role of Defense in Depth in a Risk-Informed Regulatory System.

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

January 28, 2000

MEMORANDUM TO: Thomas S. Kress, ACRS Member

FROM: Med El-Zeftawy, Sr. Staff Engineer M. Z

SUBJECT: RECONCILIATION OF ACRS REPORT ON THE "SPENT FUEL FIRES ASSOCIATED WITH DECOMMISSIONING"

The purpose of this memorandum is to provide an analysis of the letter from the Executive Director for Operations (EDO) that responded to the Committee's letter dated November 12, 1999, concerning the spent fuel fires associated with decommissioning.

ACRS Recommendation (attached)

1. An uncertainty analysis related to the oxidation kinetics and the heat rejection mechanisms is needed. There are no experimental data on the behavior of realistic fuel and cladding under representative conditions.

2. Uncertainties in the analysis to develop a decay heat critical temperature for the onset of runaway oxidation need to be quantified.

3. PRAs should be as realistic as possible (particularly for human error rates).

4. Uncertainties can be best addressed by expressing the inputs as probability distributions rather than point estimates.

5. The ACRS agreed with the choice of uncovery to the top of the fuel as appropriate end state for the PRA consequence analysis. The ACRS suggested that an acceptable frequency for this end state to be the same as that for a large, early release frequency (LERF) in RG 1.174, which is surrogate for the prompt fatality Safety Goal.

6. ACRS believes that the spent fuel fire issue would be a good candidate for testing the development of a rationalist regulatory approach.

EDO Response (attached)

I consider the EDO's response to be marginally acceptable. I do not think the staff has fully resolved the above ACRS comments and recommendations. The following are the reasons and analysis of the staff's response:

Item 1. The staff agrees with the ACRS, however, do not plan to conduct experimental research on oxidation kinetics in air or using bouyancy-driven natural circulation. Yet, it will employ "Conservative" assumptions for decay times to account for the uncertainty.

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Item 2. The staff will perform only a limited sensitivity calculations to estimate the uncertainty in critical decay time and critical temperature.

Item 3. The staff agrees with the ACRS and will reassess the parameter values used in the PRA model and place emphasis on human error assumptions. This is acceptable.

Item 4. The staff agrees with the ACRS to use probability distributions rather than point estimates, and plans to develop an approach consistent with RG 1.174. This is acceptable.

Item 5. The staff agrees with the ACRS , however, remains open to alternative end states that may be more realistic and can be justified. In addition, the staff plans to develop an approach that is consistent with RG 1.174. This is acceptable.

Item 6. The staff did not provide a convincing response to this item, and stated that the "appropriate" regulatory approach will be decided when rulemaking is initiated . No definition for the appropriate regulatory approach was given.

cc via e-mail:

ACRS Members
J. Larkins
H. Larson
R. Savio
S. Duraiswamy

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

January 13, 2000

Dr. Dana A. Powers, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: DRAFT COMMISSION PAPER REGARDING 120-MONTH UPDATE REQUIREMENT FOR INSERVICE INSPECTION AND INSERVICE TESTING PROGRAMS

Dear Dr. Powers:

This memorandum is in response to your letter of December 8, 1999, to Chairman Meserve regarding the proposed rulemaking related to the 120-month update requirement for inservice inspection (ISI) and inservice testing (IST) programs. As you are aware, since the 1970s, licensees of nuclear power plants have been required by 10 CFR 50.55a to update their ISI and IST programs every 120 months to meet the provisions of the edition and addenda of the American Society of Mechanical Engineers (ASME) Code incorporated by reference in 10 CFR 50.55a and in effect 12 months before the start of the new 120-month interval. In a proposed rule published in the *Federal Register* on April 27, 1999, the NRC suggested the replacement of the 120-month ISI/IST update requirement with a provision for licensees to update their ISI and IST programs voluntarily beyond a set of baseline ISI and IST requirements to be established in the NRC regulations. On December 2, 1999, the staff met with the Advisory Committee on Reactor Safeguards (ACRS) to discuss the public comments received on the proposed rule, the options described in a draft Commission paper on the 120-month ISI/IST update requirement, and the staff's recommendations regarding that requirement. During its December 2, 1999, meeting, the ACRS also received presentations by the ASME and the Nuclear Energy Institute (NEI) on this matter.

In the draft Commission paper, the staff identified the following three options:

(1) Replace the 120-month ISI/IST update requirement with a baseline of ISI and IST requirements and allow voluntary updating to entire subsequent NRC-endorsed ASME Code editions and addenda without prior NRC approval unless the baseline is revised in accordance with 10 CFR 50.109, where the initial baseline will consist of one of the following three possible sets of ISI and IST requirements:

(A) The 1989 Edition of the ASME *Boiler and Pressure Vessel Code* (ASME BPV Code) for ISI of ASME Code Class 1, 2, and 3 components (including supports) and for IST of ASME Code Class 1, 2, and 3 pumps and valves; the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of the ASME BPV Code for ISI of Class MC and Class CC components and their integral attachments; the 1995 Edition with the 1996 Addenda of Appendix VIII of the ASME BPV Code, Section XI, with limitations and modifications specified in 10 CFR 50.55a (as discussed in the proposed rule dated April 27, 1999);

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(B) The 1995 Edition with the 1996 Addenda of the ASME Code with the limitations and modifications specified in the NRC regulations, or

(C) A later version (e.g., the 1998 Edition) of the ASME Code with appropriate limitations and modifications.

(2) Retain the current 120-month ISI/IST update requirement and the current regulatory provision that allows licensees to use portions of NRC-endorsed ASME Code editions or addenda provided that all related requirements of the respective editions are met.

(3) Retain the 120-month ISI/IST update regulatory requirement and the current provision for use of portions of NRC-endorsed ASME Code editions or addenda, but develop explicit guidance for plant-specific alternatives to the ISI/IST update requirement.

In the NRC staff presentation to the ACRS, the staff recommended the implementation of Option 1.B.

Your letter dated December 8, 1999, to Chairman Meserve, forwarded the ACRS recommendation that the Commission adopt Option 2 as described in the draft Commission paper and retain the 120-month update requirement for ISI and IST programs. The ACRS agreed with the staff that any of the options will maintain an acceptable level of safety. Based on its review of recent analyses, the ACRS considers ISI and, on a more qualitative basis, IST to have a relatively modest impact on core damage frequency. However, the ACRS notes that, because assurance of the integrity of the reactor coolant pressure boundary and the containment is one of the cornerstones of the NRC regulatory system, ISI and IST programs have been required to provide additional assurance, through application of the defense-in-depth philosophy, of the integrity of these barriers and to compensate for uncertainties. The staff believes that defense-in-depth is maintained without requiring licensees to routinely update their ISI and IST programs. ASME Code requirements (both in ASME BPV Code, Sections III and XI) contain inherent conservatisms and margins that contribute to defense-in-depth. Also, our current practice of requiring updating of ISI and IST programs is considered inconsistent with our overall regulatory approach, in that we do not require periodic updating to new standards in other areas in order to maintain defense-in-depth unless the backfit provisions of 10 CFR 50.109 are satisfied.

The ACRS believes that the review of the past decade of experience presented by the ASME demonstrated that there were significant changes to the ISI, IST, and operations and maintenance requirements that improved the effectiveness and efficiency of these programs and that developments in technology and operating experience could lead to additional changes in the inspection programs. While the staff agrees that there continue to be improvements to the ASME Code, we believe that recently these changes are more evolutionary in nature, in many instances are relaxations to existing requirements and, while providing an overall improvement in the Code, are not necessarily justified compared to the costs imposed on licensees to implement these changes. Moreover, when scope changes are made to the ASME Code, the staff must perform (currently and under Option 1) a backfit evaluation in accordance with 10 CFR 50.109 before it can incorporate those changes into 10 CFR 50.55a.

With respect to Option 1, the ACRS does not consider 10 CFR 50.109 to be well suited to assess the appropriateness of defense-in-depth requirements, which are intended to address

uncertainties that are difficult to quantify. Until a systematic methodology is developed, the ACRS notes that decisions on defense-in-depth will have to be based on judgment. The ACRS concludes that the collective judgment of the broad-based group of experts represented by the ASME Code should be reflected in the inspection requirements. The staff considers 10 CFR 50.109 to be amenable to evaluations of the need to update the defense-in-depth requirements associated with ISI and IST programs. In particular, the NRC regulations in 10 CFR 50.109 states, in part, that the Commission will require backfitting only when it determines that "there is a substantial increase in the overall protection of the public health and safety ... and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection." In addition, the Charter of the Committee to Review Generic Requirements (CRGR) notes in Attachment 3 that the Commission has stated that the criterion in 10 CFR 50.109 regarding a substantial increase in the overall protection of the public health and safety is "flexible enough to allow for qualitative arguments that a given proposed rule would substantially increase safety." The CRGR Charter further states that "[i]ncorporation of industry standards (including revisions to existing codes and standards) into NRC rules or staff positions, as a prudent means of assuring continued conformance with currently voluntary standards and practices that provide substantial safety benefit, can provide the basis for a finding that a proposed backfit meets the 'substantial increase' standard of 10 CFR 50.109." The CRGR Charter lists the incorporation of advances in science and technology as one example of the factors that may be argued to contribute directly or indirectly to a substantial increase in safety.

The ACRS and the staff agree that the 1995 Edition with the 1996 Addenda of the ASME Code would provide a technically superior baseline for ISI and IST programs under Option 1 than the 1989 Edition of the ASME Code. As a result, the staff is recommending under Option 1.B the adoption of the 1995 Edition with the 1996 Addenda of the ASME Code as incorporated by reference in 10 CFR 50.55a as the initial baseline for ISI and IST programs. The staff believes that Option 1 will continue to rely on the judgment of the broad-based group of experts associated with the ASME Code in developing new and improved ISI and IST techniques. However, under Option 1, licensees would only be required to update their ISI and IST programs to future editions or addenda of the ASME Code when justified by the NRC as providing a safety significant enhancement to those programs and balanced against the increased burden that would be incurred by the licensees. The staff also believes that voluntary updates to more recent editions of the ASME Code will provide licensees the maximum flexibility in determining the most cost-effective approach for their facilities.

Finally, the staff points out that it will continue to review and endorse the latest editions of the ASME Code so that they can be used, on a voluntary basis, by all licensees. In the course of these reviews, the staff will assess whether the changes made to the ASME Code are of such significance as to warrant the staff backfitting the Code, either in part or in total, as a requirement. The staff concludes this approach will have two major benefits. First, it will require the staff to evaluate ASME Code improvements in accordance with the standards that are used to evaluate other potential improvements that the staff proposes to backfit on licensees, namely the 10 CFR 50.109 standard. As stated previously, the staff believes that this standard can be effectively applied through qualitative assessments. Second, the staff believes that this approach will provide an impetus for the ASME to more carefully consider those items it includes in new editions of the Code. For example, the ASME may decide to identify some items as voluntary but not mandatory or, for those items it includes as mandatory, it may evaluate them against the 10 CFR 50.109 standard.

Based on the above discussion, the staff has decided to retain its recommendation for implementation of Option 1.B in the Commission paper. If you would like to discuss this matter in more detail, please contact Jack Strosnider, Director, Division of Engineering, NRC Office of Nuclear Reactor Regulation, at 301-415-3298.

Sincerely,

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

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

December 8, 1999

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

Dear Chairman Meserve:

SUBJECT: DRAFT COMMISSION PAPER REGARDING THE 120-MONTH UPDATE REQUIREMENT FOR INSERVICE INSPECTION AND INSERVICE TESTING PROGRAMS

During the 468th meeting of the Advisory Committee on Reactor Safeguards, December 2-4, 1999, we reviewed the options proposed by the staff regarding the current requirement for licensees to update inservice inspection (ISI) and inservice testing (IST) programs every 120 months to the most recent Edition of the American Society of Mechanical Engineers (ASME) Code incorporated by reference in 10 CFR 50.55a, "Codes and Standards." Our Subcommittee on Materials and Metallurgy also reviewed this matter during its meeting on December 1, 1999. During this review, we had the benefit of discussions with representatives of the NRC staff, ASME, and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

Recommendation

We recommend that the Commission adopt Option 2 proposed by the staff and retain the 120-month update requirement for ISI and IST programs in 10 CFR 50.55a.

Background

The staff issued a proposed amendment to 10 CFR 50.55a on April 27, 1999, to solicit public comment on a proposal to eliminate the current requirement that licensees update their ISI and IST programs every 120 months to the most recent edition of the ASME Code incorporated by reference in 10 CFR 50.55a. In a letter dated April 19, 1999, we recommended against eliminating this requirement. The NRC staff held a public workshop on May 27, 1999, to discuss the update requirement. In a Staff Requirements Memorandum dated June 24, 1999, the Commission directed the staff to evaluate public comments on the update requirement and develop options and recommendations on the retention or elimination of this requirement. The Commission also directed the staff to discuss this issue further with the ACRS.

The staff has identified three options:

OPTION 1: Replace the 120-month ISI/IST update requirement with a baseline of ISI and IST requirements, and allow voluntary updating to subsequent NRC-approved Code editions and addenda unless the baseline is revised based on 10 CFR 50.109, where the initial baseline will consist of:

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Option 1.A. the 1989 Edition of the ASME Code for ISI of Code Class 1, 2, and 3 components (including supports) and for IST of Code Class 1, 2, and 3 pumps and valves; the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of the ASME Code for ISI of Class MC and Class CC components and their integral attachments; the 1995 Edition with the 1996 Addenda of Appendix VIII of the ASME Code, Section XI, with limitations and modifications specified in 10 CFR 50.55a,

Option 1.B. the 1995 Edition with the 1996 Addenda of the ASME Code with the limitations and modifications specified in the NRC regulations, or

Option 1.C. a later version (e.g., the 1998 Edition) of the ASME Code with appropriate limitations and modifications.

OPTION 2: Retain the current 120-month ISI/IST update requirement.

OPTION 3: Authorize plant-specific alternatives to the 120-month ISI/IST update requirement.

Discussion

The staff evaluated the update options in terms of the strategic goals of the Commission: (1) maintaining safety, (2) increasing public confidence, (3) reducing unnecessary regulatory burden, and (4) making NRC activities and decisions more effective, efficient, and realistic. Although the staff concludes that no particular option has an overwhelming advantage over the other options, it recommends the adoption of Option 1B, which eliminates the mandatory 120-month update. We believe that the later version of the ASME Code would provide technically superior baselines for the ISI and IST programs than the 1989 Edition, which is now over ten years old.

We agree with the conclusion of the staff that any of the options will maintain an acceptable level of safety. Each option purports to include provisions to update ISI and IST programs, although the criteria to require updating differ among the options. Furthermore, the analyses performed in support of the development of risk-informed inspections for Class 1, 2, and 3 piping and those done to support resolution of Generic Safety Issue (GSI)-190 show that ISI has a relatively modest impact on core damage frequency (CDF). We have not reviewed the analyses done to support risk-informed IST programs, but we believe that they would probably also show relatively modest impacts on CDF. This is not surprising. Because failures of these components were anticipated in the design of nuclear power plants, effective mitigation systems and procedures have been developed. However, because assurance of the integrity the reactor coolant pressure boundary and the containment is one of the cornerstones of the NRC regulatory system, ISI and IST programs have been required to provide additional assurance, through application of the defense-in-depth philosophy, of the integrity of these barriers and to compensate for uncertainties.

NEI and the staff argue in support of Option 1 that the current ASME Code requirements have reached such a level of maturity that further updating will provide little benefit. We believe that the review of the past decade of experience presented to us by the ASME demonstrated that there were significant changes to the ISI, IST, and operations and maintenance requirements that improved the effectiveness and efficiency of these programs and that developments in technology and operating experience could lead to additional changes in the inspection programs. Changes are not introduced in the ASME Code requirements frivolously. The ASME Code represents the consensus of a broad-based group of experts that includes strong utility representation (approximately 30% of the Section XI membership) as well as

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representation from manufacturers, vendors, the NRC, and other engineering and consulting organizations.

Under Option 1, any mandated updates to the ISI and IST programs would have to pass the 10 CFR 50.109 backfit criteria. The 50.109 evaluation is not well suited to assess the appropriateness of defense-in-depth requirements, which are intended to address uncertainties that are difficult to quantify. In our May 19, 1999 report, we outlined an approach for developing a systematic methodology for the evaluation of defense in depth; however, lacking such a methodology at the present time, decisions on defense in depth will have to be based on judgment. The collective judgment of the broad-based group of experts represented by the ASME Code should be reflected in the inspection requirements.

Sincerely,

Dana A. Powers
Chairman

References:

1. Memorandum dated November 18, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-99-XXX, Subject: 120-Month Update Requirement for Inservice Inspection and Inservice Testing Programs (Predecisional Draft).
2. ACRS letter dated May 19, 1999, from Dana A. Powers, Chairman, ACRS, to Honorable Shirley A. Jackson, Chairman, NRC, Subject: The Role of Defense In Depth in a Risk-Informed Regulatory System.
3. Memorandum dated June 24, 1999, from Annette L. Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - Reconsideration of SECY-99-017 (Proposed Amendment to 10 CFR 50.55a).
4. Letter dated April 19, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: SECY-99-017, "Proposed Amendment to 10 CFR 50.55a."
5. Table provided by ASME during ACRS meeting, December 2-4, 1999, "Important Section XI SG NDE Code Changes and Code Cases, 1989 Addenda through 1999 Addenda," Revision 2, 11/1/99.
6. Memorandum dated November 12, 1999, from Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Generic Safety Issue-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."

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

January 31, 2000

MEMORANDUM TO: ACRS Members

Noel Dudley

FROM: Noel Dudley, Senior Staff Engineer

SUBJECT: ANALYSIS OF THE EDO'S RESPONSE TO THE ACRS REPORT ON THE DRAFT COMMISSION PAPER REGARDING 120-MONTH UPDATE REQUIREMENT FOR ISI AND IST PROGRAMS

The purpose of this memorandum is to provide an analysis of the letter from the Executive Director for Operations (EDO) that responded to the Committee's report dated December 8, 1999, concerning the elimination of the required 120-month update in 10 CFR 50.55a, "Codes and standards."

ACRS Position

The Committee's letter recommended retaining the 120-month updated requirement for inservice inspection (ISI) and inservice testing (IST) programs in 10 CFR 50.55a. The report provided the following observations:

- ISI and IST programs provide additional assurance of the integrity of the reactor coolant pressure boundary and the containment,
- significant changes have been made to the ISI and IST programs in last 10 years,
- the backfit rule is not well suited to assess the appropriateness of defense-in-depth requirements, and
- the collective judgement of the broad-based group of experts represented by the ASME Code should be reflected in the inspection requirements.

EDO Response

The EDO response provided justification for the staff's recommendation for eliminating the 120-month update requirement. The justification included the following points.

- defense-in-depth is maintained without requiring licensees to update their ISI and IST programs,
- backfit provisions should be satisfied before requiring an update to new standards,
- ASME Code changes are more evolutionary in nature and are not necessarily justified compared to the costs,

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- the backfit rule is flexible enough to allow for qualitative agreements,
- eliminating the update requirement will provide an impetus for the ASME to more carefully consider those items it includes in new editions of the Code.

Analysis

The EDO response appears to me to be short sighted. The staff plans to expeditiously review and endorse future ASME Code changes to provide flexibility to licensees. It also plans to use qualitative assessments to justify backfitting ASME Code requirements. Up to this point, the staff has been unsuccessful in reaching either of these goals. The staff resources necessary to endorses code cases and review licensee exemption requests will probably increase, since exemptions to Code requirements will have to be submitted every time the code case is used. Due to the high threshold of the backfit rule, few if any new Code requirements will become mandatory, essentially ending the required implementation of enhanced inspection methods.

Plant life extension will increase the likelihood of identifying new and unexpected degradation mechanisms. The aging management programs that are developed for these new mechanisms will not automatically become part of licensees' ISI programs. Examples of aging mechanisms currently under review include metal fatigue after 40 years of plant operation, thermal fatigue of small bore piping, and void swelling of reactor vessel internals. The elimination of the 120-month update requirement will result in an additional burden on the staff to regulate the new aging management programs developed to address emergent aging mechanisms.

The Committee may want to consider replying to the EDO response.

cc via e-mail:
J. Larkins
S. Duraiswamy
ACRS Fellows and Staff

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

January 13, 2000

Dana A. Powers, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE 190, "FATIGUE EVALUATION OF METAL COMPONENTS FOR 60-YEAR PLANT LIFE"

Dear Dr. Powers:

Thank you for your consideration of the staff's proposed resolution of GSI-190. As per the advisory committee's recommendation, we have closed GSI-190 without any additional regulatory requirements. In addition, we will ensure that any utilities requesting license renewal consider the management of environmentally assisted fatigue degradation in their aging management programs. Again, thank you for your time and attention regarding this issue.

Sincerely,

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

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

December 10, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-190, "FATIGUE EVALUATION OF METAL COMPONENTS FOR 60-YEAR PLANT LIFE"

During the 468th meeting of the Advisory Committee on Reactor Safeguards, December 2-4, 1999, we reviewed the proposed resolution of Generic Safety Issue (GSI)-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life." During our review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

RECOMMENDATIONS

- We agree with the staff's proposal that GSI-190 be resolved without any additional regulatory requirements.
- The staff should ensure that utilities requesting license renewal consider the management of environmentally assisted fatigue in their aging management programs.

BACKGROUND

The effects of fatigue for the 40-year initial reactor license period were studied and resolved under GSI-78, "Monitoring of Fatigue Transient Limits for Reactor Coolant System," and GSI-166, "Adequacy of Fatigue Life of Metal Components."

The staff concluded that risk from fatigue failure of components in the reactor coolant pressure boundary was very small for 40-year plant life. In our March 14, 1996 letter, we agreed with the staff's conclusion.

GSI-190 was established to address the residual concerns of GSI-78 and GSI-166 regarding the environmental effects of fatigue on pressure boundary components for 60-years of plant operation. The scope of GSI-190 included design-basis fatigue transients, studying the probability of fatigue failure and its effects on core damage frequency (CDF) of selected metal components for 60-year plant life.

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DISCUSSION

Resolution of GSI-190 was based on the results of an NRC-sponsored study performed by the Pacific Northwest National Laboratory (PNNL). In that study, PNNL examined design-basis fatigue transients and the probability of fatigue failure of selected metal components for 60-year plant life and the resulting effects on CDF.

The PNNL study showed that some components have cumulative probabilities of crack initiation and through-wall growth that approach unity within the 40- to 60-year period. The maximum failure rate (through-wall cracks per year) was in the range of 10^{-2} per year, and those failures were associated with high cumulative usage factor locations and components with thinner walls, i.e., pipes more vulnerable to through-wall cracks. There was only a modest increase in the frequency of through-wall cracks in major reactor coolant system components having thicker walls. In most cases, the leakage from these through-wall cracks is small and not likely to lead to core damage. Therefore, the projected increased frequency in through-wall cracks between 40- and 60-years of plant life does not significantly increase CDF. Based on the low contributions to CDF, we agree with the proposed resolution of GSI-190.

Environmentally assisted fatigue degradation should be addressed in aging management programs developed for license renewal. Minimization of leakage is important for operational safety, occupational doses, and for continued economic viability of the plants.

Dr. William J. Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated November 12, 1999, from Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Generic Safety Issue-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."
2. Letter dated March 14, 1996, from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, to James M. Taylor, Executive Director for Operations, NRC, Subject: Resolution of Generic Safety Issue-78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System."
3. Letter dated October 16, 1995, from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, to Shirley Ann Jackson, Chairman, NRC, Subject: Fatigue Action Plan.

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

February 1, 2000

MEMORANDUM TO: Dr. Robert L. Seale, Acting Chairman
Plant Systems Subcommittee

FROM: Amarjit Singh, P.E., Senior Staff Engineer
ACRS/ACNW Technical Support Staff

SUBJECT: ANALYSIS OF THE EDO RESPONSE TO THE ACRS LETTER REGRADING PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-190, "FATIGUE EVALUATION OF METAL COMPONENTS FOR 60-YEAR PLANT LIFE"

The purpose of this memorandum is to provide you with an analysis of EDO response to the Committee's letter dated December 10, 1999, regarding the staff's proposed resolution of Generic Safety Issue (GSI)-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life".

ACRS Recommendations:

1. The Committee agreed with the staff's proposed resolution of GSI-190 without any additional requirements.

EDO Response:

The staff agrees with the ACRS recommendation and have closed GSI -190 without any additional requirements.

2. ACRS also recommended that the NRC staff should ensure that utilities requesting renewal consider the management of environmentally assisted fatigue in their aging management programs.

EDO Response:

The staff has committed that they will ensure that any utilities requesting license renewal consider the management of environmentally assisted fatigue degradation in their aging management programs.

Analysis:

The EDO response was responsive and had committed to implement the changes as recommended by ACRS. Therefore, I suggest that the Committee consider the response satisfactory.

cc: J. Larkins
H. Larson
S. Duraiswamy

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

January 20, 2000

Dr. Dana Powers, Chairman
Advisory Committee on Reactor Safeguards
United States Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: NUREG-1624, REVISION 1, "TECHNICAL BASIS AND IMPLEMENTATION GUIDELINES FOR A TECHNIQUE FOR HUMAN EVENT ANALYSIS (ATHEANA)"

Dear Dr. Powers:

On November 19, 1999, we presented a draft version of Revision 1 of NUREG-1624, "Technical Basis and Implementation Guidelines for A Technique for Human Event Analysis (ATHEANA)" to the Human Factors Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS), followed by a presentation to the ACRS full committee on December 3, 1999. On December 20, 1999, we received a letter from the ACRS containing a discussion of the ATHEANA report, as well as conclusions and recommendations. While we agree with these comments and suggestions, we plan to address and implement them as resources permit. Our responses to these comments are provided below.

Comment: The objective of ATHEANA is to develop a methodology that: (a) allows a realistic, qualitative analysis of potential accident sequences and past incidents involving human actions and (b) allows a realistic evaluation of the probabilities of unsafe human actions for inclusion in probabilistic risk assessments (PRAs). The qualitative evaluation in NUREG-1624, Revision 1, is at an advanced stage of development and is achieving its purpose. The quantitative portion still needs significant development.

Response: We agree that the quantitative process needs substantial development, and we plan to concentrate the program on this issue in FY2000.

Comment: ATHEANA's focus on the context within which the operators must act as well as on the error mechanisms is an appropriate paradigm shift away from a focus on "human error."

Response: We agree with this conclusion.

Comment: ATHEANA deals with operator actions that take place after an abnormal event has occurred, e.g., a fire or an initiating event, as defined in PRAs. Its scope should be extended to include normal activities that may cause a plant event.

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Response: We agree that this would be an important refinement to ATHEANA. Expanding the scope of ATHEANA to include normal activities will be considered for future development as resources permit.

Comment: The term "error-forcing context" is not used consistently and is misleading in some situations. An alternative, more descriptive term must be defined.

Response: While we understand your concerns with the terminology chosen, we still believe that the "error-forcing context" is appropriate in most situations. We will ensure that it is used consistently and appropriately.

Comment: The process of searching for error-forcing contexts is complex. Not all human actions require such a detailed treatment, and a screening process should be developed to identify the level of analysis that a given situation requires. The development of the screening process should be given priority.

Response: We realize that the search process for error-forcing contexts can be difficult and is not appropriate for application to all human actions. Although the development of a screening process will be addressed as soon as feasible, we will include guidance in the final version of NUREG 1624 that ATHEANA, in the present form, should not be used as a screening tool. We will also provide guidance on the role of ATHEANA, when it should be used and when other methods (e.g., THERP) should be considered.

Comment: In developing symptom-based procedures, the industry considered many deviations from expected plant behavior. The ATHEANA search process for deviations should take advantage of this experience.

Response: We agree with this suggestion, and will address this issue as resources permit.

Comment: The elements of a plant's safety culture that influence the operators when they are faced with a decision making situation should be explicitly considered when evaluating the error-forcing contexts.

Response: We agree that the plant's safety culture does influence human actions. However, we believe that this issue is best addressed by focusing on the plant's work processes and practices and the subsequent impact on risk. This modification to ATHEANA will be implemented as resources permit.

Comment: The application of ATHEANA to a fire-initiated accident scenario does not make clear its advantages over existing, less complex methods. More examples of applications need to be developed.

Response: Currently, applications to scenarios concerning pressurized thermal shock (PTS) and the use of remote shutdown panels during fire-initiated accidents are planned (in support of other staff activities). These applications, as well as others, should demonstrate the advantages of ATHEANA, and provide insights on where the process can be modified and refined.

Thank you for your comments and suggestions for refining the ATHEANA process. We would be pleased to brief the committee again on ATHEANA, after further work is completed, and look forward to continuing discussions with the ACRS.

Sincerely,

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

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

December 15, 1999

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

Dear Dr. Travers:

SUBJECT: NUREG-1624, REVISION 1, "TECHNICAL BASIS AND IMPLEMENTATION GUIDELINES FOR A TECHNIQUE FOR HUMAN EVENT ANALYSIS (ATHEANA)"

During the 468th meeting of the Advisory Committee on Reactor Safeguards, December 2–4, 1999, we reviewed Revision 1 of NUREG-1624, "Technical Basis and Implementation Guidelines for A Technique for Human Event Analysis (ATHEANA)." Our Subcommittee on Human Factors also reviewed this document on November 19, 1999. During our review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

Conclusions and Recommendations

1. The objective of ATHEANA is to develop a methodology that: (a) allows a realistic, qualitative analysis of potential accident sequences and past incidents involving human actions and (b) allows a realistic evaluation of the probabilities of unsafe human actions for inclusion in probabilistic risk assessments (PRAs). The qualitative evaluation in NUREG-1624, Revision 1, is at an advanced stage of development and is achieving its purpose. The quantitative portion still needs significant development.

2. ATHEANA's focus on the context within which the operators must act as well as on the error mechanisms is an appropriate paradigm shift away from a focus on "human error."

3. ATHEANA deals with operator actions that take place after an abnormal event has occurred, e.g., a fire or an initiating event, as defined in PRAs. Its scope should be extended to include normal activities that may cause a plant event.

4. The term "error-forcing context" is not used consistently and is misleading in some situations. An alternative, more descriptive term must be defined.

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5. The process of searching for error-forcing contexts is complex. Not all human actions require such a detailed treatment, and a screening process should be developed to identify the level of analysis that a given situation requires. The development of the screening process should be given priority.

6. In developing symptom-based procedures, the industry considered many deviations from expected plant behavior. The ATHEANA search process for deviations should take advantage of this experience.

7. The elements of a plant's safety culture that influence the operators when they are faced with a decisionmaking situation should be explicitly considered when evaluating the error-forcing contexts.

8. The application of ATHEANA to a fire-initiated accident scenario does not make clear its advantages over existing, less complex methods. More examples of applications need to be developed.

Discussion

Understanding human errors and evaluating their probability of occurrence have been active areas of research since the Three Mile Island accident. "First-generation" models, i.e., those developed in the 1970s and 1980s, varied in their depth of modeling human performance. No serious attempt was made to incorporate concepts from the behavioral and cognitive sciences into these models. The focus was on "human error" with its connotation of blame.

In the late 1980s, a need for "second-generation" models that would delve deeper into the causes of human error was recognized. Attention shifted toward an examination of contextual elements that could trigger cognitive error mechanisms which could lead to unsafe crew actions. ATHEANA is the first major effort to develop a model for human performance based on this new paradigm. We believe that this shift in paradigm is appropriate and commend the staff for carrying out this work.

ATHEANA focuses on the analysis of human performance after a plant event . This is natural, since this has been the main perceived need for improving human reliability analysis. Errors made during routine activities, such as maintenance and testing, are analyzed satisfactorily by using the methods of the human reliability handbook (NUREG/CR-1278, Revision 1). Normal plant activities that may lead to plant events, such as the reactor coolant drain-down event at the Wolf Creek Generating Station, Unit 1, on September 19, 1994, are not currently addressed by ATHEANA.

The principal premise of ATHEANA is that "plant conditions" and "performance-shaping factors" may produce an "error-forcing context" that could trigger an error mechanism such as the refusal to change an initial misdiagnosis when contradictory evidence is received. The performance-shaping factors reflect human-centered influences such as training and communications.

The search for error-forcing contexts is a major effort. A multidisciplinary team consisting of human-reliability experts, plant operators, PRA specialists, and possibly others is needed. Such an extensive effort is not appropriate for all potentially unsafe human acts. We are concerned that the amount of resources required may discourage practitioners from even attempting to use ATHEANA. We believe that a set of screening guidelines should be developed to define different levels of treatment for various unsafe human acts. The qualitative insights gained from the detailed ATHEANA investigations should form the basis for the development of simpler methods for use when appropriate.

We note that a similar situation arises when a decision must be made about the methodology to be used to elicit and utilize expert opinions in probabilistic seismic hazard analysis (NUREG/CR-6372). In some situations of great national interest in which the uncertainties are large, a very formal methodology that is implemented by a multidisciplinary team is required. In other situations, experience has shown that a single technical integrator using informal input from experts is sufficient.

The process of searching for error-forcing contexts starts with a base-case scenario that describes the expected plant and operator behavior for a given initiator. The error-forcing contexts are, then, identified by searching for deviations from the base-case scenario. A great deal of work along these lines was done when the industry developed symptom-based emergency operating procedures. We believe that ATHEANA should take advantage of this experience.

ATHEANA defines an error-forcing context as "the combined effect of PSFs [performance-shaping factors] and plant conditions that create a situation in which human error is likely." Yet, in Chapter 10 of NUREG-1624, Revision 1, it is stated that an error-forcing context may be "so noncompelling that there is no increased likelihood of the UA [unsafe act] compared with the routine PRA context." We believe that the use of clear, accurate terminology is essential, especially when concepts from the behavioral sciences are brought into the practice of engineering. We believe that an alternative terminology should be developed to replace "error-forcing context."

The error mechanisms are developed from a cognitive model that consists of detection, situation assessment, response planning, and response implementation. All of these activities involve decisions that the plant crew must make, especially in the response planning phase. Although the discussion of error mechanisms clearly assumes that decisions are being made, e.g., establishing wrong goals is identified as a possible error, no formal attempt is made to investigate either the decisionmaking process or the impact of time. The decisionmaking processes (as well as the error-forcing contexts) are expected to be different for event sequences that evolve in a relatively short time, e.g., in less than about 30 minutes, and for sequences taking place over longer periods. In addition, decisionmaking may involve balancing conflicting safety and economic objectives; therefore, the plant's safety culture is a critical element in these decisions. Safety culture should be explicitly considered when evaluating the error-forcing context.

The application of ATHEANA to a fire-initiated accident scenario failed to convince us that the results obtained were sufficiently better than those obtained through other, presumably less

resource-intensive methods to justify the use of ATHEANA . There are some inconsistencies between this application and the theoretical development in NUREG-1624, Revision 1. For example, the error-forcing contexts that the methodology claims are its foundation were not identified explicitly. We believe that a number of applications are urgently needed to convince the human reliability community and the end users that ATHEANA is a practical model that represents an improvement over existing models. These applications will also serve to guide the development of the screening process that we mentioned above.

A major motivation for the development of ATHEANA is the need for adequate models to support risk-informed regulatory applications. The guidance provided currently for evaluating the probabilities of unsafe human acts is very general. The HEART model (NUREG-1624, Revision 1, Chapter 10), whose quantitative results are proposed as one way for assessing the probability of a given error-forcing context, was developed several years before the ATHEANA project started and there is no effort to adapt it to ATHEANA. If the HEART model is to form the basis for quantifying the error-forcing context in the ATHEANA process, then ATHEANA should include sufficient information to assess the appropriateness of using this model for such purpose.

We acknowledge that any attempt at quantifying probabilities of error-forcing contexts will necessarily involve expert judgment. However, the guidance given by ATHEANA does not build on the large amount of work that has been done on the elicitation and utilization of expert opinions, e.g., in NUREG-1150, NUREG/CR-6372, and NUREG/CR-3518.

A more serious effort on probability evaluation will also help in developing the screening process that we recommended above. We expect that a lot of the details that are now investigated in the analysis of plant conditions, performance-shaping factors, and error mechanisms will not affect the quantification process, thus suggesting ways for limiting the qualitative investigation. While we recognize that the likelihood of plant conditions can be estimated, we believe that the probabilities of performance-shaping factors are much more difficult to evaluate. Thus, ATHEANA must demonstrate the feasibility of evaluating probabilities of error-forcing contexts, of which the performance-shaping factors are an important component.

We believe that the development of the screening process and the application of ATHEANA to several realistic accident scenarios are critical to its success. We look forward to working with the staff on these matters.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

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References:

1. U.S. Nuclear Regulatory Commission, NUREG-1624, Revision 1, "Technical Basis and Implementation Guidelines for a Technique for Human Event Analysis (ATHEANA)," September 1999.
2. U.S. Nuclear Regulatory Commission, NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," Final Report Prepared by Sandia National Laboratories, A. D. Swain and H. E. Guttmann, August 1983.
3. U.S. Nuclear Regulatory Commission, NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," Prepared by R.J. Budnitz, G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris, April 1997.
4. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," June 1989.
5. U.S. Nuclear Regulatory Commission, NUREG/CR-3518, "SLIM-MAUD: An Approach to Assessing Human Error Probabilities Using Structured Expert Judgment," Prepared by D.E. Embrey, P. Humphreys, E.A. Rosa, B. Kirwan, and K. Rea, Brookhaven National Laboratory, July 1984.
6. International Atomic Energy Agency, International Nuclear Safety Advisory Group (INSAG) Safety Series No. 75-INSAG-4, "Safety Culture," Vienna, 1991.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001
February 1, 2000

MEMORANDUM TO: ACRS Members

FROM: Noel Dudley, Senior Staff Engineer

SUBJECT: ANALYSIS OF THE EDO'S RESPONSE TO THE ACRS LETTER ON NUREG-1624, REVISION 1, REGARDING ATHEANA

The purpose of this memorandum is to provide an analysis of the letter from the Executive Director for Operations (EDO) that responded to the Committee's letter dated December 15, 1999, concerning the latest revision to NUREG-1624, "Technical Basis and Implementation Guidelines for a Techique for Human Event Analysis (ATHEANA)."

ACRS Position

The Committee's letter contained nine recommendations for improving ATHEANA and making it more useful.

EDO Response

The EDO response agreed with most of the recommendations and provided the following comments:

- Expanding the scope of ATHEANA to include normal activies will be considered for furture development as resources permit.
- We still believe that the "error-forcing context" is appropriate in most situations. We will ensure that it is used consistently and appropirately.
- We will include guidance in the firnal version of NUREG-1624 that ATHEANA should not be used as a screening tool. We will also provide guidance on the role of ATHEANA, when it should be used and when other methods should be considered.
- We believe that safety culture is best addressed by focusing the plant's work processes and practices and the subsequent impact on risk.
- Planned applications of ATHEANA (in support of other staff activities) should demonstrate the advantages of ATHEANA, and provide insights on where the process can be modified and refined.

Analysis

The EDO response is satisfactory. The staff agrees with all except one of the Committee's recommendations and plans to implement these recommendations as resources permit.

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

January 20, 2000

Dr. B. John Garrick, Chairman
Advisory Committee on Nuclear Waste
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Dana A. Powers, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: IMPLEMENTING A FRAMEWORK FOR RISK-INFORMED REGULATION IN THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

Dear Sirs:

I am responding to your letter dated November 17, 1999, to Chairman Meserve, regarding this subject. The Office of Nuclear Material Safety and Safeguards (NMSS) reviewed the recommendations you have provided in your letter in light of SECY-99-100, "Implementing a Framework for Risk-Informed Regulation," and its associated staff requirements memorandum (SRM). Although there are differences between these documents, we believe that there are no fundamental conflicts, and that your recommendations will be helpful as we implement the Commission's instructions.

The Commission, in approving SECY-99-100, endorsed the staff's plans to implement a 5-step process consisting of: (1) identifying candidate regulatory applications; (2) making a decision on how to modify those applications, and subsequently; (3) changing current regulatory approaches; (4) implementing the risk-informed approaches; and (5) developing or adapting traditional tools and techniques of risk analysis for application to the materials world. In addition, in its SRM on SECY 99-100, the Commission directed the staff to develop appropriate material safety goals analogous to the reactor safety goal, and include, as a goal, the avoidance of property damage. The Commission directed the staff to use an enhanced participatory process to develop these goals, and factor an Agreement State component into the decision making process to avoid duplication. The Commission also directed the staff to consider whether critical groups can be defined for classes of material use, and give due consideration to existing radiation protection standards in 10 CFR Part 20.

Your letter included two key recommendations:

1. NMSS should develop a set of principles and a safety goal approach for each of its regulated activities to guide its implementation of risk-informed and performance-based regulation.

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With respect to your recommendation on guiding principles, the staff is in the process of developing criteria for determining whether a portion of the regulatory framework (e.g., rule, regulatory guide, inspection procedure) can and should be made more risk-informed. This determination will need to be made for each type of regulatory activity conducted by NMSS. We recognize that many of NMSS' programs are already risk-informed to varying degrees, and that efficient use of resources dictates a carefully considered approach. This is step 1 of the 5 step process approved by the Commission. The staff intends to seek public participation in formulating the criteria and is developing a Federal Register Notice (FRN) and planning a public meeting to obtain public input at an early stage in this process. Our preliminary thoughts are to propose an approach where new, risk-informed activities would need to meet the following test:

1. The proposed risk-informed regulatory approach will either resolve a question with respect to maintaining safety, improve the efficiency and/or the effectiveness of NRC processes, reduce unnecessary regulatory burden, or improve public confidence,

2. Sufficient information (data), and analytical methods exist or can be developed to support risk-informing,

3. Implementation can be realized at a reasonable cost to the NRC and the regulated entity, and provide a net benefit. The net benefit will be considered to apply to the public, the applicant, and the NRC staff, and

4. The staff's work to implement subsequent steps, namely 2 through 5 of the 5-step process, will be prioritized and implemented through the NRC's Planning, Budgeting and Performance Management Process (PBPM).

In addition to seeking comment on these criteria, the staff will ask commentors to identify any specific applications or general areas where they believe NRC should focus its efforts. Our current schedule is to publish the FRN in February 2000 and hold the public meeting several weeks thereafter.

You also recommended that the staff identify analytical methods and develop a safety goal approach for its regulated activities. The staff has begun an effort to catalog and examine the risk assessment methods and the risk measures and metrics that are presently being proposed or applied in NMSS. The purpose of this initiative is to confirm that the approach being used is appropriate for the application and enhances the quality of the decision making process. This activity is focused upon the current efforts to increase the use of risk information in safety decision making involving the fuel cycle facilities, waste storage, transport and disposal, and in the industrial and medical applications of nuclear technology. Examples of these activities include the following: 1) the reassessment of the risk of transporting spent fuel by highway or railway which considers accidents beyond the licensing basis in 10 CFR 71; 2) Performance Assessment for the High and Low Level Radioactive Waste disposal sites; 3) Integrated Safety Assessments for fuel cycle facilities; and 4) the pilot PRA for dry cask storage.

Finally, I would like to bring to your attention another initiative we have underway for training staff in the methods and use of risk analysis. We plan to start a systematic training program for both Headquarters and Region based staff in the Spring of 2000. The approach being taken recognizes the knowledge and skill levels required to understand, utilize and perform risk assessments. Consequently, we are developing training in the fundamentals of risk assessment and risk management geared toward the vast majority of NMSS and Region based staff members. We also plan to offer specialized training to those staff members assigned to the Risk Group in NMSS and those involved in the current activities described above.

We propose to interact with you on all aspects of these efforts by meeting with you on specific topics. The staff is ready to brief you on its activities in these areas and will contact you to discuss your interest and availability. Our hope is to meet on topics early enough for meaningful coordination and discussion. It seems to us that the newly formed joint ACRS/ACNW subcommittee can fit our needs in just that fashion.

We are grateful for the opportunity to begin an ongoing dialog with your subcommittee.

Sincerely,

William D. Travers
Executive Director
for Operations

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

November 17, 1999

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

Dear Chairman Meserve:

SUBJECT: IMPLEMENTING A FRAMEWORK FOR RISK-INFORMED REGULATION IN THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

During the 113th meeting of the Advisory Committee on Nuclear Waste (ACNW), October 12-13, 1999, and the 467th meeting of the Advisory Committee on Reactor Safeguards (ACRS), November 4-6, 1999, the Committees considered the staff's proposed framework for risk-informed and performance-based regulation in the Office of Nuclear Material Safety and Safeguards (NMSS), as articulated in SECY-99-100 and an associated Staff Requirements Memorandum dated June 28, 1999. A meeting of the ACRS/ACNW Joint Subcommittee was held on May 11, 1999, to discuss these matters. We had the benefit of the documents referenced.

Recommendations

1. NMSS should develop a set of principles and a safety goal approach for each of its regulated activities to guide its implementation of risk-informed and performance-based regulation.

2. NMSS should identify the analytical methods to be applied to implement risk-informed and performance-based regulation on an application-specific basis.

Discussion

The NMSS staff is examining the use of risk information in four major categories of regulated activities: (1) long-term commitment of a site to the presence of nuclear material (e.g., high-level waste disposal); (2) use of engineered casks to isolate nuclear material under a variety of conditions (e.g., transportation and storage); (3) physical and chemical processing and possession of nuclear material at a large-scale facility (e.g., fuel fabrication); and (4) use of sealed or unsealed byproduct material in industrial and medical applications. The objectives of

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this examination are to focus regulatory activities on matters that are important to safety and avoid unnecessary burdens on licensees and the NRC staff.

The diversity of the four categories of activities listed above indicates that the risk assessment methods for material licensees are likely to be different from those for nuclear power plants. While quantitative risk assessment is a well-developed and utilized tool for nuclear power plant licensees, it may be unnecessarily complex for the NMSS regulated activities. The performance assessments (PAs) done for waste repositories are conceptually similar to probabilistic risk assessments (PRAs) for reactors. Recently, there have been developments for simplified approaches to quantitative risk analysis, e.g., integrated safety assessments (ISAs), that are less rigorous than PRAs or PAs.

The staff must address two crucial issues as it considers risk methods in the regulation of material licensees:

1. What criteria should be used to decide whether the regulations for a specific nuclear materials activity should be changed to a risk-informed regulation? Can the current deterministic criteria, accounting methods, or proposed approaches such as ISA accomplish risk-informed objectives?

2. What risk analysis methods (and scope) and risk acceptance criteria should be applied to the operations that merit risk-informed regulation?

To address the first question, we believe that the staff will need to develop a set of principles for risk-informed regulation. Such a set of principles is important to guide the need for and change from a prescriptive form of regulation to a less prescriptive, but risk-informed, method of regulation. In developing these principles, the staff should take full advantage of the knowledge base unique to materials and waste disposal regulation, as well as the staff's experience in developing principles for other regulatory applications, such as Regulatory Guide 1.174.

Some of the characteristics of nuclear materials regulation that differ from reactor regulation include: (1) experience in regulating to radiation exposure standards, as opposed to surrogate measures such as facility damage, (2) diversity of types of licensee activities involving major differences in materials, facilities, and practices, (3) activities not dominated by a clear-cut feature such as core damage, and (4) activities where the operational risk, as opposed to the accident risk, may be the central issue of risk regulation. Although these characteristics distinguish materials regulation from reactor regulation, the Committees believe that the approach to regulatory decisionmaking for the NMSS activities should have a basis that is consistent with the approach for reactor regulation.

An important element introduced in Regulatory Guide 1.174 and that should be investigated in the present context of materials regulation is that regulatory decisionmaking should be based on an analytic and deliberative process. Analytical results from risk assessments and other engineering analyses are only part of the input to this process. Qualitative inputs, e.g., the preservation of the defense-in-depth philosophy, may be considered by an expert panel or other decisionmaking entity. In developing the new principles, the staff should consider this approach and its applicability to the various NMSS activities. If qualitative information is to be used in the

decisionmaking process, then the reason(s) should be explained. If there is a need for an expert panel for some activities, its form and composition should be discussed.[1]

Consideration should be given to developing variations on the safety goal approach to risk acceptance. One variation may be to include uncertainty directly in the risk acceptance criteria via required confidence levels in their determination. Another may be to define acceptance criteria that are either met or not, i.e., the range of risk is partitioned into two regions, the acceptable and unacceptable regions. Another might be to adopt a three-region approach. In this concept, there is a range of acceptability with an upper and lower bound. The lower bound constitutes the level below which no further action is required. The upper bound constitutes a level above which definitive action to control the risk is required. The middle region is the region in which cost-benefit tradeoffs can be made. These are a few concepts that should be investigated by the staff for materials regulation. There may be others.

The Committees believe that, just as "guiding principles" are important to establishing a well-founded philosophy of risk-informed regulation, so are certain risk assessment concepts. The representation of risk as a triplet set is such a guiding concept. The triplet consists of accident scenarios (what can go wrong?), probabilities of these scenarios (how likely is each scenario?), and the consequences (what are the consequences?). We view the various risk (or safety) assessment methods that exist in the literature as dealing with these three elements of the risk triplet in different ways. PRAs for reactors and PAs for HLW repositories offer the most complete treatment of the triplet, and they require the most resources. We believe that the staff should clarify how any chosen method deals with the risk triplet (either quantitatively or qualitatively) and justify the appropriateness of the selected scopes as differentiated among the four major categories of NMSS licensees. If methods that are less rigorous than PRAs or PAs are judged to be appropriate for certain applications, their treatment of the triplet should be explicitly identified. The reasons for resorting to these less rigorous methods should be carefully justified. We are especially concerned about the completeness of the scenario list and the analysis of uncertainties.

We look forward to reviewing staff activities on these matters during future meetings.

Sincerely,

B. John Garrick
Chairman, ACNW

Dana A. Powers
Chairman, ACRS

[1] This concept of an expert panel refers to the discussion on integrated decisionmaking in Regulatory Guide 1.174. The purpose of such an expert panel is to evaluate multiple sources of information to make decisions in an integrated manner. This is different from the guidance in the "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program," NUREG-1563, that refers to a specific formalized process for developing information and "data" to be used in a performance assessment.

References:

1. Memorandum dated June 28, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, and John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: Staff Requirements - SECY-99-100 - Framework for Risk-Informed Regulation in the Office of Nuclear Material Safety and Safeguards.
2. SECY-99-100, Memorandum dated March 31, 1999, from William D. Travers, Executive Director for Operations, NRC, to the Commissioners, Subject: Framework for Risk-Informed Regulation in the Office of Nuclear Material Safety and Safeguards.
3. Memorandum dated February 24, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-144 - White Paper on Risk-Informed and Performance-Based Regulation.
4. Report dated April 19, 1999, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to Shirley Ann Jackson, Chairman, NRC, Subject: Status of Efforts on Revising the Commission's Safety Goal Policy Statement.
5. 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.
6. U.S. Nuclear Regulatory Commission, NUREG-1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program," November 1996.

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February 3, 2000

NOTE TO: Dr. Thomas Kress
Dr. George Apostolakis

FROM: Mike Markley

SUBJECT: RECONCILIATION DURING 469th ACRS MEETING: CONCERNING IMPLEMENTING A FRAMEWORK FOR RISK-INFORMED REGULATION IN THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

The purpose of this note is to provide an analysis of the EDO's January 20, 2000 response, to te ACRS report dated November 17, 1999, concerning implementing a framework for risk-informed regulation in the Office of Nuclear Material Safety and Safeguards (NMSS).

In general, the EDO's response agrees with the Committee's comments and recommendations. However, instead of developing principles as was recommended by the joint Committees, the staff has offered a new approach for screening/qualifying potential risk-informed elements of the regulatory framework (rules, regulatory guides, and inspection procedures) for each type of activity regulated by NMSS. The staff plans to seek public comment on this revised approach. With respect to the joint Committee recommendation to develop analytical methods, risk criteria, and safety goals, the EDO states that an initiative has begun to examine the risk methods that are being applied presently to NMSS activities and to consider how the decisionmaking process can be enhanced. The EDO also highlights the training being provided to the staff and offers to discuss these issues during future meetings.

From my view, the EDO's response should be acceptable to the Committee, in part, because the Commission disapproved the staff's recommendation to proceed with a study of developing overarching principles (SECY-99-191) concerning modifications to the Safety Goal Policy Statement. An appropriate reconciliation might be to indicate, in the Summary Report to the Commission, that the Committees plan to continue their review of this matter during future meetings.

This item will, of course, still need to be reconciled by the ACNW

cc: J. Garrick, ACNW
G. Hornberger, ACNW
ACRS reconciliation package

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

January 24, 2000

Dr. Dana A. Powers, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

Dear Mr. Powers:

In your letter to Chairman Meserve on December 10, 1999, you provided the results of the review by the Advisory Committee on Reactor Safeguards of Baltimore Gas and Electric Company's (BGE's) license renewal application for Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2, and the related final safety evaluation report prepared by the staff. On the basis of its review, the Committee concluded that all open and confirmatory items were resolved and that there is reasonable assurance that CCNPP can be operated safely in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The Committee's timely review and preparation of its letter was important in maintaining the schedule established for the review of the CCNPP renewal application. The staff is currently providing the Commission with its recommendation regarding issuance of the CCNPP renewed license.

The staff would like to provide the Committee with additional information regarding several topics discussed and found acceptable in your letter, which also have further generic implications.

1. Effects of the Reactor Coolant Environment on Fatigue Life

BGE committed to implement a plant-specific monitoring program using correlations published in NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999, to calculate the effects of the reactor coolant environment on the fatigue life of components and piping. The Committee agreed with BGE's approach and commented that BGE's commitments may have generic implications for other applications for license renewal. The evaluation of environmental effects on the fatigue life of reactor coolant components and piping is a plant-specific, time-limited aging analysis that each license renewal applicant must conduct. The approach used by BGE is one method the staff finds acceptable. Another approach acceptable to the staff is the one Duke Energy is using for renewal of the Oconee Nuclear Station licenses. Duke committed to monitor the design transients, taking into account the environmental effects of the reactor coolant. With both the BGE and Duke approaches, corrective actions will be required if acceptance limits are exceeded.

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As the Committee is aware, the staff completed on December 26, 1999, resolution of Generic Safety Issue (GSI) 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life." The conclusion to close out this issue is based upon the low core damage frequencies from fatigue failures of metal components estimated by technical studies making use of recent fatigue data developed on test specimens. The results of these probabilistic analyses and associated sensitivity studies led the staff to conclude that no generic regulatory action is required. However, calculations including environmental effects, that were performed to support resolution of this issue, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The staff is considering the BGE and Duke approaches in developing the guidance for inclusion in the License Renewal Standard Review Plan for future renewal applicants to address environmental effects on fatigue life.

2. Timing of One-Time Inspections

The use of one-time inspections by BGE to be conducted before the end of the current license term was found acceptable to the Committee and the staff in specific cases to confirm the expectation that degradation is not occurring or that the effects are not significant. In your letter you commented on the importance of performing these inspections late in the current license term.

As you noted in your letter, BGE has indicated that it intends to perform most of the one-time inspections late in the current license term. The staff is including a condition in the renewed license that requires BGE to complete the inspections by the end of the current license term or request an amendment to its license. However, the staff determined that it is not necessary to establish a restriction on how early the one-time inspections can be performed. When the Commission established the license renewal rule in 1991, it determined that renewal applications could be submitted as early as 20 years before expiration of the current operating license because that would be sufficient operating experience to disclose plant-specific, age-related degradation. Therefore, if an aging effect is occurring, performance of the inspection after 20 years of operation but before the end of the current term should identify the aging effect. Also, it should be noted that these one-time inspections are intended to confirm that aging effects are not occurring. In those cases for which the staff had concerns regarding whether an aging effect was occurring at CCNPP or whether a one-time inspection was sufficient, the one-time inspections originally proposed by BGE were converted into periodic inspections. Additionally, if operating experience reveals an emerging concern, whether before or after the one-time inspection is performed, the licensee must investigate and take any required corrective action in accordance with the requirements of Appendix B to 10 CFR Part 50.

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3. ASME Class 1 Small-Bore Piping

BGE resolved an issue regarding cracking of ASME Class 1 small-bore piping at CCNPP by committing to perform a one-time inspection. Similar to other one-time inspections, this inspection is intended to verify that cracking is not occurring within the subject pipe. If cracking is found, appropriate corrective action will be required. Because cracking of this piping involves plant-specific conditions, BGE's commitment to perform the inspection does not generically resolve this aging concern for other plants. Future renewal applicants will need to address management of this aging effect for their plants.

In addition, the Materials Reliability Project, a consortium of representatives of pressurized water reactor owners, vendors, the Nuclear Energy Institute, and the Electric Power Research Institute, have undertaken a comprehensive voluntary initiative to address thermal fatigue issues. While the focus of this initiative is to address small diameter piping, the results may also have implications for larger piping sizes. The goal of this program is to develop guidance for licensees regarding thermal fatigue management through the use of operational controls, monitoring, inspections, and assessment. This industry initiative could provide a generic approach to addressing this issue and will be of interest to the staff both during the current licensing period and into the period of extended operation for license renewal.

The staff expects to continue discussions with the Committee on generic implementation of license renewal and is available to discuss further with the Committee the implications of these or other topics.

Sincerely,

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

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

December 10, 1999

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

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

During the 468th meeting of the Advisory Committee on Reactor Safeguards, December 2-4, 1999, we completed our review of the Baltimore Gas and Electric Company's (BGE's) application for license renewal of the Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2 and the related Final Safety Evaluation Report (FSER). Our review included four meetings with the staff and the applicant concerning the license renewal of CCNPP and two meetings with the staff and the Nuclear Energy Institute concerning generic license renewal issues. During this review, we had the benefit of discussions with representatives of the NRC staff and BGE. We also had the benefit of insights gained from our review of another license renewal application and of the documents referenced. We provided an interim letter, dated May 19, 1999, concerning the BGE application.

Conclusion

On the basis of our review of BGE's application, the FSER, and the resolution of the open and confirmatory items identified in the Safety Evaluation Report (SER), we conclude that BGE has properly identified the structures, systems, and components (SSCs) that are subject to aging management programs. Furthermore, we conclude that the programs instituted to manage aging-related degradation of the identified SSCs are appropriate and provide reasonable assurance that Calvert Cliffs Nuclear Power Plant, Units 1 and 2 can be operated in accordance with their current licensing basis for the period of the extended license without undue risk to the health and safety of the public.

Background and Discussion

This report is intended to fulfill the requirement of 10 CFR 54.25 that each license renewal application be referred to the ACRS for a review and report. BGE requested renewal of the operating licenses for the CCNPP, Units 1 and 2 for a period of 20 years beyond the current license term. The FSER documents the results of the staff's review of information submitted by

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BGE, including those commitments that were necessary to resolve open and confirmatory items identified by the staff in its SER. The staff's review included the verification of the completeness of the identification and categorization of the SSCs considered in the application; the validation of the integrated plant assessment process; the identification of the possible aging mechanisms associated with each passive long-lived component; and the adequacy of the aging management programs. The staff also conducted onsite inspections to verify the implementation of these programs.

The staff's SER identified a number of open and confirmatory items. The staff and BGE have now resolved all the open and confirmatory items, in part, through additional commitments made by BGE. The BGE commitments to be added to its Final Safety Analysis Report (FSAR) will become a part of the plant's licensing basis and are enforceable.

The commitments made by BGE are adequate to resolve the open and confirmatory items. Several of the open items such as the effects of the reactor coolant environment on fatigue life and the thermal fatigue of American Society of Mechanical Engineers (ASME) Class 1 small-bore piping may have generic implications for other applications for license renewal.

BGE committed to the implementation of a plant-specific monitoring program in which it will use correlations published in NUREG/CR-5704 to calculate the effects of the reactor coolant environment on fatigue life of components and piping. The correlations reflect data developed to resolve Generic Safety Issue (GSI)-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life." We concur with the staff's conclusion that BGE's proposed program is an acceptable plant-specific approach for the resolution of GSI-190 concerns.

BGE resolved an open item concerning cracking of ASME Class 1 small-bore piping by including small-bore piping in the CCNPP's age-related degradation inspection (ARDI) program. Under the ARDI program, inspections of small-bore piping will be performed during the last five years of the current license term. The timing of these inspections is appropriately set late in the current licensing period so that they will be most useful for assessing the need for additional requirements. We concur with the resolution of this open item.

Another open item concerned the adequacy of the bases provided to justify the use of one-time inspections to resolve some potential aging issues. The staff has accepted one-time inspections prior to the end of the current license term, rather than regular, periodic inspections, in those cases in which age-related degradation is not expected to occur. In such cases, the one-time inspection is intended to confirm the expectation that age-related degradation is not occurring, or that its effects are insignificant. We agree that this is an appropriate approach to address such aging issues. We reviewed the basis for the staff's acceptance of one-time inspections in individual cases (SER open Item 3.1.6.3-1) and concur with the staff's determination.

During our meeting, BGE informed us that it expects to conduct most of the one-time inspections after 30 years of plant operation. We believe that it is important that these one-time inspections be performed late in the current license term (the last ten years).

After the SER was issued, the staff identified void swelling as a potential mode of degradation for pressurized water reactor vessel internals. BGE committed to participate in the industry

programs to address the significance of void swelling and to develop an inspection program if needed.

As CCNPP, Units 1 and 2 age, inspection and operating experience may prompt significant adjustments to their aging management programs. BGE is required to document in its FSAR that the 10 CFR Part 50 Appendix B quality assurance program also applies to those nonsafety-related SSCs which are subject to an aging management review. Furthermore, the staff has required that BGE include in its FSAR the license renewal application commitments that the staff relied on to conclude that aging effects will be adequately managed for the period of extended operation. These steps ensure that future changes can be controlled under the 10 CFR 50.59 process. Future schedule changes will require license amendments if the schedules are delayed.

The staff has performed a comprehensive and thorough review of the BGE application. The additional programs required by the staff are appropriate and sufficient. Current regulatory requirements and existing BGE programs provide adequate management of aging-induced degradation for those components within the scope of the license renewal rule.

We believe that the applicant and the staff have identified possible aging mechanisms associated with passive long-lived components. Adequate programs have been established to manage the effects of aging so that CCNPP, Units 1 and 2 can be operated safely in accordance with their licensing basis for the period of the extended license.

Dr. William J. Shack did not participate in the Committee's deliberations on aging-induced degradation.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated November 16, 1999, from Christopher I. Grimes, Office of Nuclear Reactor Regulation, NRC, to Charles H. Cruse, Baltimore Gas and Electric Company, Subject: Final Safety Evaluation Report.
2. Letter dated May 19, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Interim Letter on the Safety Aspects of the Baltimore Gas and Electric Company's License Renewal Application for Calvert Cliffs Nuclear Power Plant, Units 1 and 2.
3. U. S. Nuclear Regulatory Commission, NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.
4. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2," March 1999.
5. Letter dated April 8, 1998, from Charles H. Cruse, Baltimore Gas and Electric Company, to U. S. Nuclear Regulatory Commission Document Control Desk, Subject: Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2, Application for License Renewal.

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6. U. S. Nuclear Regulatory Commission, Code of Federal Regulations, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
7. U. S. Nuclear Regulatory Commission, Code of Federal Regulations, 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

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

February 1, 2000

MEMORANDUM TO: ACRS Members

FROM: Noel Dudley, Senior Staff Engineer

SUBJECT: ANALYSIS OF THE EDO'S RESPONSE TO THE ACRS REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR CALVERT CLIFFS NUCLEAR POWER PLANT

The purpose of this memorandum is to provide an analysis of the letter from the Executive Director for Operations (EDO) that responded to the Committee's report dated December 10, 1999, concerning the license renewal application for Calvert Cliffs Nuclear Power Plant. The Committee concluded that the programs instituted to manage aging-related degradation provide reasonable assurance that the Plant can be operated without undue risk to the public.

The EDO response provided additional information on the generic implications of the following three topics:

Effects of the reactor coolant environment on fatigue life: The evaluation of the effects is a plant-specific time-limited aging analysis that each licencee renewal applicant must conduct. BGE will calculate the effects of the the reactor coolant environment on the fatigue life of compnents and piping. Duke will monitor the design transients. Both approaches were approved by the staff.

Timing of one-time inspections: The staff determined that it is not necessary to establish a restriction on how early the one-time inspections can be performed. The staff decided that if an aging effect is occurring, performance of the inspection after 20 years of operation, but before the end of the current term of the license, should identify the aging effect.

ASME Class 1 small-bore piping: The industry has undertaken a comprehensive voluntary initiate to address thermal fatigue issues. The goal of this initiative is to develop guidance for licensees reguarding thermal fatigue management through the use of operational controls, monitoring, inspections, and assessment. The industry intiative could provide a generic approach to addressing this issue.

Analysis

The EDO response is satisfactory.

cc via e-mail:
J. Larkins
S. Duraiswamy
ACRS Fellows and Staff

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
469th	15	15-1

Title

FUTURE ACRS ACTIVITIES

Authors

John T. Larkins

List of Documents Attached

Memorandum dated February 1, 2000, Subject: Future ACRS Activities - 470th ACRS Meeting, March 2-4, 2000

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

From Staff Person

John T. Larkins

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

February 2 , 2000

MEMORANDUM TO: ACRS Members

FROM: John T. Larkins, ACRS

SUBJECT: FUTURE ACRS ACTIVITIES, 470th ACRS MEETING
MARCH 2-4, 2000

Attached is a proposed list of topics for the 470th March 2-4, 2000, ACRS meeting and beyond. The Planning and Procedures Subcommittee has reviewed this proposed list of topics and its recommendations are included.

Attachment:
Future Agenda Items
dated February 1, 2000

cc w/attachments:
ACRS Staff
ACRS Fellows

ANTICIPATED WORKLOAD
FEBRUARY 3-5, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Markley	Subcommittee Report on Risk-Informed Tech. Specs. and Programs for Risk-Based Analysis of Reactor Operating Experience	--	RPRA 12/15-16	ACRS/ACNW Subc. 1/13-14 PO 1/20 Retreat 1/27-29
		Markley	Lower-Power and Shutdown Operations Risk Insights Report (staff presentation completed at the December meeting)	*Report		P&P 2/2
	Kress	Singh	Impediments to the Increased Use of Risk-Informed Regulation and Use of Importance Measures in Risk-Informing 10 CFR Part 50	Report	--	
Barton	Sieber	Markley	Reactor Oversight Process (Significant Determination Process and Associated Performance Indicators)	*Interim Report	PO 1/20	Retreat 1/27-29
Bonaca	--	Dudley	Proposed Final Amendment to 10 CFR 50.72 and 50.73 Regarding Event Reporting System	Report	--	RPRA 12/15-16 PO 1/20 Retreat 1/27-29
	Seale	Dudley	License Renewal Process	*Report	--	P&P 2/2

ANTICIPATED WORKLOAD
FEBRUARY 3-5, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Kress	Apostolakis	Boehnert	Proposed Revision of the Commission's Safety Goal Policy Statement for Reactors	*Report **(Tentative)**	ACRS/ACNW Subc. 1/13-14	RPRA 12/15-16 PO 1/20
	Apostolakis	Markley	Report of the ACRS/ACNW Joint Subcommittee	--	--	Retreat 1/27-29
Powers	--	Larkins	Follow-up Items Resulting from the January 27-29, 2000 ACRS Retreat	--	Retreat 1/27-29 P&P 2/2	PO 1/20
		Duraiswamy	Response to Commissioner's Questions	Report		
Shack	--	Dudley	120-Month ISI/IST Update Requirement	Report	--	Retreat 1/27-29
Sieber	Barton	El-Zeftawy	Proposed Regulatory Guide and Associated NEI Document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluation" - **Status Report**	*Report (to be completed in March)	--	RPRA 12/15-16 PO 1/20 Retreat 1/27-29
Wallis	--	Boehnert	Proposed Final Revision of Appendix K to 10 CFR Part 50	*Report	--	Retreat 1/27-29
		El-Zeftawy	Research Report to the Commission	Final Report	--	--

*Time permitting, the Committee will complete this report during this meeting.

ANTICIPATED WORKLOAD
MARCH 1(P.M.)-4, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	Kress	Markley	Proposed Risk-Informed Revisions to 10 CFR Part 50 - **Status Report**	Report **(tentative)**	--	PLR 2/23-24 P&P2/29 ADAMS 3/1
		Dudley	Latest Revision to Human Performance Program Plan	Report		
Barton	Sieber	Singh	Proposed Final Revision 3 to Reg. Guide 1.160 on Maintenance	Report	--	ADAMS 3/1
		Markley	Proposed Commission Paper on Revised Reactor Oversight Process	Report		
Bonaca	Seale	Dudley	License Renewal Application for Oconee	Final Report	PLR 2/23-24	P&P2/29 ADAMS 3/1
			Proposed Final Amendment to 10 CFR 50.72 and 50.73	Report		
Kress	Apostolakis	Boehnert	Proposed Revision of Commission's Safety Goal Policy Statement for Reactors (staff presentation completed at the February meeting)	Report	--	PLR 2/23-24 ADAMS 3/1
Powers	--	El-Zeftawy	PIRT for High Burnup Fuel	Report **(tentative)**	P&P2/29	PLR 2/23-24 ADAMS 3/1
Seale	--	Boehnert	Proposed Resolution of GSI B-17, Criteria for Safety-Related Operator Actions	Report	--	PLR 2/23-24 ADAMS 3/1

ANTICIPATED WORKLOAD
MARCH 1(P.M.)-4, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
			Meeting with the Commission March 2, 2000 (9:30-11:30am)	--	--	--
Powers		El-Zeftawy	Overview	--	--	--
Kress	--	Markley	**I. Impediments to the further development of risk-informed regulations:**			
	Apostolakis		a. Plant-Specific Risk Acceptance Criteria			
	Apostolakis		b. ANS/ASME Standards for PRA			
	Bonaca		c. PRA Capabilities within the Industry and Within the Staff			
	Powers		d. Fire-Risk Analysis and the Gerry-Rigged Model to be Used in the Significant Determination Process Versus that Called for in the NFPA 805 Standard			
	Powers		e. Shutdown and Low Power Risk Findings of Analyses Versus Conventional Wisdom			
	Kress		f. Role of Defense-in-Depth in Risk-Informed Regulatory System			
	Shack		g. Future Development of PRA Capabilities Including the Evaluation of the Risk Significance of QA/QC			

ANTICIPATED WORKLOAD
MARCH 1(P.M.)-4, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Singh	**II. Importance Measures, Conservatisms, Uncertainties, and the Establishment of Thresholds**	--	--	--
Barton	Bonaca	Dudley	**III. Technical Foundations of the Performance Indicators**	--	--	--

ANTICIPATED WORKLOAD
APRIL 6-8, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Markley	Proposed Final ASME Standard for PRA[1] Quality - Phase 1	Report	--	M&M 3/16 P&P4/5 NR 4/4
	--		Proposed White Paper on Risk-Based Performance Indicators	Report		
Barton	--	Markley	Reactor Trip and Partial Loss of AC Power at Indian Point Unit 2	--		NR 4/4
Bonaca	--	Markley	Special Studies for Risk-Based Analysis of Reactor Operating Experience	--	--	P&P 4/5 M&M 3/16 NR 4/4
Kress	--	El-Zeftawy	Spent Fuel Pool Accident Risk for Decommissioning Plants	Report	--	THP 3/14-15 M&M 3/16 NR 4/4
			Proposed Resolution of GSI-173A, "Spent Fuel Pool Cooling Issue for Operating Plants"	Report	--	
Shack	--	Dudley	Proposed Approach for Revising 10 CFR 50.61, PTS Rule - **Subcommittee Report**[2]	--	M&M 3/16	NR 4/4
Sieber	Barton	El-Zeftawy	Proposed Reg. Guide and Associated NEI Document 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations"	Report	--	NR 4/4
Uhrig	--	Singh	Proposed Digital I&C Research Plan	Report	--	NR 4/4

[1]Assignments to review individual chapters to be made.

[2]Dr. Shack to recommend the need for presentation to the full Committee.

ANTICIPATED WORKLOAD
APRIL 6-8, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Wallis	--	Boehnert	TRACG/RETRAN-3D Subcommittee Report	--	THP 3/14-15	NR 4/4

2/4/00

ACRS AGENDA ITEMS

I. 470TH ACRS MEETING (MARCH 2-4, 2000)

1. Meeting with the NRC Commissioners (Open)(DAP et.al./JTL et.al.) ESTIMATED TIME: 2 hours

   Purpose: Periodic Meeting

   PRIORITY: HIGH

   The ACRS is scheduled to meet with the NRC Commissioners between 9:30 and 11:30am on Thursday, March 2, 2000 to discuss risk-informing 10 CFR Part 50 and related matters.

2. Final SER for Oconee License Renewal (Open)(MVB/RLS/NFD) ESTIMATED TIME: 2 hours

   Purpose: Review and Comment

   PRIORITY: HIGH

   Review schedule specified in the Chairman's Tasking Memorandum (CTM). The staff requested that the ACRS complete its review of the final safety evaluation report (SER) regarding Oconee license renewal application and issue its final report at the March 2000 ACRS meeting. The Plant License Renewal Subcommittee plans to review this item during a meeting in the vicinity of the plant on February 23-24, 2000, at Clemson, South Carolina.

3. Proposed Final Revision 3 to Regulatory Guide 1.160, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants" (Open)(JJB/AS) ESTIMATED TIME: 1 ½ hours

   Purpose: Review and Comment

   PRIORITY: HIGH

   Review requested by the NRC staff. During the 467th meeting of the ACRS, the Committee reviewed the proposed revision 3 to Regulatory Guide 1.160 (DG-1082), "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," and the revised draft of Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The Committee wrote a report to the Commission on November 12, 1999, recommending that the proposed revision 3 to Regulatory Guide 1.160 be issued for public comment. The Committee also supported the staff's endorsement of the NEI guidance. The staff issued this regulatory guide for public comment on December 10 1999, and the comment period ended on

January 10, 2000. The staff requests ACRS review of the proposed final revision 3 to Regulatory Guide 1.160 during the March 2000 ACRS meeting.

4. Proposed Resolution of Generic Safety Issue (GSI) B-17, "Criteria for Safety-Related Operator Actions" (Open)(RLS/AS) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

PRIORITY: HIGH

Review requested by the NRC staff. Generic Safety Issue (GSI) B-17, "Criteria for Safety-Related Operator Actions," relates to the current plant designs which rely on the operator actions in response to certain transients. GSI B-17 called for the development of criteria for safety-related operator actions that included a methodology for determining whether automatic actuation would be needed to mitigate a design basis event. This GSI originally was identified in June 1978 and was prioritized, "MEDIUM," in November 1983.

During the 426th meeting of the ACRS, the Committee reviewed the proposed final Regulatory Guide 1.164, which was developed by the NRC staff to resolve GSI B-17. The staff proposed to endorse ANSI/ANS 58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions," to resolve GSI B-17. This Standard establishes criteria and simplifies the process for calculating the minimum allowable response times for manual operator actions to stabilize the plant during a design basis event. The Committee provided a letter dated November 14, 1995, recommending that the staff not endorse the ANSI/ANS 58.8-1994 in Regulatory Guide 1.164, and did not believe that endorsement of the Standard was the appropriate way to resolve GSI B-17. The Committee did not find adequate technical basis for the estimates of minimum times for operator actions in ANSI/ANS 58.8-1994. Additionally, the proposed Standard did not address operator response times for advanced nuclear power plants.

The staff plans to provide the resolution package to ACRS by the first week of February 2000 and requests to brief the Committee during the March 2000 meeting.

5. Phenomena Identification and Ranking Tables (PIRTs) for High Burnup Fuel (Open)(DAP/MME) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

PRIORITY: HIGH

Review Requested by the NRC Staff. On March 24, 1999, the ACRS issued its letter on the PIRT process for high burnup fuel endorsing the Office of Nuclear Regulatory Research (RES) approach on this matter. Since then, RES has made progress and held two public workshops to solicit expert opinions

regarding the PIRT for high burnup fuel. RES representatives would like to brief the ACRS on this issue at the March 2000 ACRS meeting.

6. Status Report on Development of Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Open) (GA/TSK/MTM) ESTIMATED TIME: 2 hours

Purpose: Information Briefing/Possible Comment

PRIORITY: HIGH

Briefing proposed by the NRC staff. In a Staff Requirements Memorandum (SRM) dated June 8, 1999, the Commission approved most Committee recommendations related to SECY-98-300. The Commission directed the staff to pursue Option 2 to develop risk-informed definitions for "safety-related" and "important to safety," and modify the scope of the Maintenance Rule. Also, the Commission approved Option 3 to study other areas of 10 CFR Part 50 that may be risk informed, and approved policy issues related to voluntary provisions and the use of exemptions for pilot plants.

The Committee last reviewed proposed risk-informed revisions to 10 CFR Part 50 (SECY-99-256) during its September 30-October 2, 1999 ACRS meeting, and provided a report to the Commission dated October 12, 1999. In that report, the Committee agreed with staff proposal to develop new regulatory section 10 CFR 50.69 and associated Appendix T and hat the current terminology of SSCs should be preserved and that additional terminology referring to safety significance be considered. The Committee also noted that the determination of safety significance of SSCs relies heavily on importance measures which are strongly affected by the scope and quality of PRA. The Committee stated that Appendix T should clarify the proper role of importance measures and that guidance should be provided in Appendix T for the expert panel on the use of importance measures. Although the Commission has not yet voted on SECY-99-256, the staff has requested to brief the Committee on the status of its efforts to develop risk-informed revisions to 10 CFR Part 50.

The Committee plans to meet with the NRC Commissioners on March 2, 2000, to discuss risk-informing 10 CFR Part 50 and related matters.

7. Proposed Draft Regulatory Guide and Associated Guidance in NEI 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations" (Open) (JDS/JJB/MTM) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

PRIORITY: HIGH

Review schedule specified in the CTM. The ACRS reviewed proposed changes to 10 CFR 50.59 (Changes, Tests and Experiments) in May 1999 and

provided a report to Chairman Jackson dated May 17, 1999, recommending issuance of the proposed final rule and conforming changes. In a Staff Requirements Memorandum (SRM) dated June 22, 1999, the Commission approved the proposed final rule.

The staff briefed the Committee on the status of proposed draft regulatory guide and the resolution of issues associated with the NEI 96-07 document during the February 3-5, 2000 ACRS meeting. The staff plans to provide these documents to the Committee in early February 2000 and brief the Committee during its March 2-4, 2000 meeting. The staff plans to submit these documents to the Commission in March 2000, requesting approval for issuance to solicit public comment.

8. Revised Reactor Oversight Process - Technical Components (Open) (JJB/MTM)
ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

PRIORITY: HIGH

Review requested by the Commission and NRR Director. In a letter dated November 23, 1999, the NRR Director requested the Committee to review selected technical components of the reactor oversight process. In particular, the NRR Director requested that the Committee review the updated significance determination process (SDP) and plant performance indicators (PIs). The staff has made changes to the SDP and PIs as a result of insights from the pilot plant program and expects to continue to make changes in the coming months as more experience is gained. In a Staff Requirements Memorandum dated December 17, 1999, the Commission requested the ACRS to review the technical adequacy of the performance indicators (current and proposed) for the new reactor oversight process, which includes an assessment of the extent to which the performance indicators, collectively, provide meaningful insights into those areas of plant operations that are most important to safety. ACRS response to the Commission is due March 15, 2000.

The staff and NEI met with the Plant Operations Subcommittee on January 20, 2000. During that meeting, the staff informed the Subcommittee that the proposed Commission paper associated with the reactor oversight process will not be available for ACRS review until mid-February 2000. The Subcommittee identified a number of issues for the staff to address during the February 3-5, 2000 ACRS meeting.

Mr. Barton recommends that the Committee prepare an interim letter to the EDO during the February ACRS meeting with a final report to the Commission during the March 2-4, 2000 ACRS meeting.

9. Proposed Final Amendment to 10 CFR 50.72 and 50.73 (Open) (MVB/NFD)
ESTIMATED TIME: ½ hour

Purpose: Review and Comment

Priority: HIGH

Review requested by the ACRS. The staff briefed the ACRS on the proposed final amendment to 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," and 50.73, "Licensee Event Report Systems," during the February 3-5, 2000 ACRS meeting. NEI also provided its views regarding this proposed final amendment. The Committee plans to continue its discussion of this matter during the March meeting, especially the intent of the requirement for reporting degraded components.

10. Human Performance Plan (Open)(GA/NFD) ESTIMATED TIME: 1 hour

Purpose: Review and Comment

PRIORITY: MEDIUM

Review requested by the NRC staff. The staff is preparing a Commission paper that provides the status of the Human Performance Plan. The staff has requested an extension on the paper until late January 2000, due to comments made by NRR. The staff plans to provide the ACRS with a draft of this Commission paper in early February and brief the Committee at the March 2000 meeting.

II. ITEMS REQUIRING COMMITTEE ACTION

11. NRC Research Plan for Digital Instrumentation and Control (Open) (REU/AS) ESTIMATED TIME: 1 hour

Purpose: Decide on a course of action

Review requested by the NRC staff. The Office of Nuclear Regulatory Research (RES) is in the process of developing a digital instrumentation and control (I&C) research plan as recommended in the National Academy of Sciences, National Research Council Study on, "Digital Instrumentation and Control Systems." The purpose of this research plan is to address the limitations in the current method and tools used to assess digital I&C capability for application in nuclear power plants and the understanding of technical issues associated with emerging technology. The intent of the digital I&C research plan is to support the regulatory review of digital I&C systems, including software quality, environmental qualification of digital systems, and emerging technology.

The RES staff plans to submit the I&C research plan to the Commission in Mid-April 2000. The staff plans to provide a draft version of the plan no later than March 17, 2000 for ACRS review. The staff requests ACRS review during the April meeting.

The Planning and Procedures Subcommittee recommends that Dr. Uhrig propose a course of action, including the need for a consultant to support Committee review of this matter.

12. Proposed Revision to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (open) (GA/MTM) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the NRC staff. The staff is proposing to revise Regulatory Guide 1.174 and associated Office Letter 803 for risk-informed changes to the licensing basis. The staff plans to incorporate guidance provided in SECY-99-246 concerning the use of risk information in license amendment reviews that were not submitted for risk-informed decisionmaking in accordance with Regulatory Guide 1.174. The staff may also consider possible endorsement of industrial standards for PRA quality (e.g., ASME) in the proposed revision to Regulatory Guide 1.174, if appropriate. The staff proposes to provide the draft documents in early June 2000 and requests to brief the Subcommittee on Reliability and PRA in late June 2000 and the full Committee during its July 12-14, 2000 meeting. The staff plans to issue the documents for public comment in late July 2000.

The Planning and Procedures Subcommittee recommends that Dr. Apostolakis propose a course of action.

13. Proposed Final Regulatory Guide DG-1053, "Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Open) (WJS/NFD) ESTIMATED TIME: 1 ½ hours

Purpose: Decide on a Course of Action

Review requested by the NRC staff. The ACRS reviewed the draft Regulatory Guide, DG-1025, at the July 1993 meeting and requested an opportunity to review the proposed final version of the guide after the public comments have been reconciled. The staff held a public meeting on September 18, 1996, to discuss the reconciliation of public comments. The Materials and Metallurgy Subcommittee heard a briefing regarding the status of DG-1025 on April 16, 1997. Subsequently, the staff added a section to this Guide for using Monte Carlo methodologies. DG-1025 is now designated as DG-1053.

The staff held another public meeting concerning DG-1053 in September 1999, to discuss issues associated with the use of the proposed draft regulatory guide. The staff plans to meet with NEI to address administrative type concerns before reconciling public comments. The staff plans to provide a copy of the revised draft Regulatory Guide to ACRS in May 2000 and brief the full Committee in June 2000. In the interim, the staff has provided the ACRS with a copy of the

latest revision of DG-1053 in early January 2000. A copy of the latest draft of this Guide has been sent to Dr. Shack.

The Planning and Procedures Subcommittee recommended previously that Dr. Shack provide his views on the need for the Committee to review this Guide. **Dr. Shack recommends that the Committee review the proposed final version of this Guide at the June ACRS meeting and that a consultant be invited to provide support to the Committee's review of this matter.**

14. Spent Fuel Pool (SFP) Accident Risk for Decommissioning Plants (Open)(TSK/MME) ESTIMATED TIME: 2 hours

Purpose: Decide on a Course of Action

Review Requested by the Commission. During the 467th meeting of the ACRS, November 4-6, 1999, the Committee reviewed a draft report of a technical study prepared by the NRC staff on the spent fuel pool accident risk at decommissioning plants. The Committee also discussed this issue with the Nuclear Energy Institute representatives and two members of the public.

The Committee issued a letter on November 12, 1999, and provided its views on the direction of this effort. Currently, the staff plans to modify the draft report and provide the ACRS with a revised draft by mid-February, 2000. The staff will issue the revised draft for public comment (possibly for 45 days) in February 2000. In an SRM dated December 21, 1999, the Commission requested that the ACRS perform a technical review of the validity of the draft study and risk objectives, and any available public comments, and provide its views to the staff and the Commission by March 15, 2000. Subsequently, the staff has requested an extension of 30 days with a completion of the ACRS review by April 15, 2000.

The staff plans to provide the proposed final report and any available public comments to the ACRS in March 2000 and brief the ACRS at the April 2000 meeting.

The Planning and Procedures Subcommittee recommends that a "team," consisting of Drs. Powers, Kress, and Shack and ACRS Fellow Cronenberg be established to perform the technical review requested by the Commission.

15. ABB/CE Topical Report on "Common Qualified Platform" (Open) (REU/AS) ESTIMATED TIME: 1 ½ hours

Purpose: Decide on a Course of Action

Review requested by the ACRS. ABB-CE (now known as British Nuclear Fuels (BNF)) submitted a topical report for the Common Qualified Platform that is the physical realization of the design that was proposed in the ABB-CE System 80+. The proposed digital instrumentation and control (I&C) system would replace the

existing reactor protection system, engineered safety features system, and post accident monitoring system. The NRC staff plans to issue an interim safety evaluation report (SER) by 2/29/00. The staff expects to issue the final SER by end of July 2000.

The staff also plans to issue a draft final SER on Siemen's topical report by January 31, 2000 and final SER by March 15, 2000.

The staff has suggested that the ACRS also hear a briefing on Siemen's topical report either prior to the Subcommittee meeting in mid-April proposed on ABB-CE topical report or during that meeting.

Please note, these topical reports have been submitted by individual vendors. The Westinghouse and Triconex plan to submit topical reports for NRC staff review in the near future.

The Planning and Procedures Subcommittee recommends that Dr. Uhrig propose a course of action regarding the staff's suggestion to brief the ACRS on the Siemen's topical report. Dr. Uhrig recommends that the Committee hear a briefing regarding SERs on ABB-CE and Siemens topical reports at the June ACRS meeting.

III. UPDATED INFORMATION

16. Management Directive 6.4 to Address ACRS Concerns Associated with the Generic Safety Issue Process (RLS/AS) ESTIMATED TIME: 1 hour

Purpose: Review and Comment

Review requested by the NRC staff. During the 449th and 456th meetings of ACRS, the Committee reviewed matters associated with the generic safety issue (GSI) process. In letters to the Executive Director for Operations dated March 16, 1998 and October 16, 1998, the Committee raised concerns about the GSI process. In response, RES had performed a reevaluation of the GSI process and briefed the ACRS on the results of the reevaluation during the March 1999 meeting. The staff stated that ACRS concerns will be addressed through the Management Directive (MD) 6.4. The Committee issued a letter to the EDO dated April 19, 1999, recommending that the staff conduct a pilot study to evaluate the effectiveness of using MD 6.4 and the associated handbook. The staff is in the process of performing the pilot study. The staff expects to complete the pilot study and MD 6.4 during December 2000 and brief the ACRS at the February 2001 meeting.

17. Initiatives Related to Risk-Informed Technical Specifications (Open) (JDS/GA/MTM) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the NRC staff. The NRC staff requested to brief the Committee during the November 2-4, 1999 ACRS meeting, concerning initiatives related to risk-informed technical specifications (TS). The staff has worked closely with the industry Risk-Informed Technical Specifications Task Force (RITSTF). RITSTF has proposed a list of seven issues to pursue in the Phase 1 effort: 1) define HOT SHUTDOWN as the preferred end state for TS actions (as opposed to COLD SHUTDOWN) , 2) increase time allowed to delay entry into action when a surveillance is missed, 3) modify existing Mode restraint logic to allow greater flexibility (e.g., use risk assessments for entry into higher Mode LCOs), 4) develop a risk-informed extension of current allowed outage times (AOTs) based on the Configuration Risk Management Program (CRMP), 5) optimize surveillance requirements, 6) modify TS 3.0.3 LCO actions and timing by extending minimum LCO shutdown time from 1 hour to 24 hours, and 7) define required actions when equipment is functional but not operable.

The Reliability and Probabilistic Risk Assessment Subcommittee held a meeting on December 16, 1999, to discuss this matter. Dr. Apostolakis is scheduled to provide a report to the Committee, recommending a course of action, at the February 2000 ACRS meeting.

18. Committee Visit to DOE/DOD Naval Reactors Facilities Pursuant to Review of VIRGINIA Class Submarine Design (RLS/PAB) ESTIMATED TIME: 4 hours

Purpose: Site Visit/Information Gathering

ACRS Initiative The Naval Reactors (NR) Organization will be submitting its new submarine design (VIRGINIA Class, successor to the LOS ANGLES Class) to the NRC and ACRS for review in mid-2001. The Committee last reviewed an NR reactor plant design (SEAWOLF) in 1994. Only three of the current ACRS members were on the Committee at the time of that review.

Dr. Powers has suggested that the Committee interact with NR, early on, to become familiar with its organization, history, and approach. Discussions with NR representatives have lead to the scheduling of a visit by the Committee to NR Headquarters Offices for a ½ -day introductory briefing on the Naval Reactors program. This visit is scheduled for the morning of April 4, 2000. In the future, the Committee is expected to visit the NR training complex located at the Charleston SC Naval Base. This complex is comprised of the Moored Training Ships and the Nuclear Power Training School. Discussions are still under way with NR, and it appears that this visit may be timely later this year.

19. Proposed Final Regulatory Guide which Endorses NEI 97-04 Document on Design Bases Information (Open)(JJB/MTM) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review schedule specified in the CTM. In October 1999, the Committee reviewed draft Regulatory Guide DG-1093, "Guidance and Examples for

Identifying 10 CFR 50.2 Design Bases." The Committee also discussed Appendix B of NEI 97-04, "Design Bases Program Guidelines," which the staff proposes to endorse in DG-1093 as an acceptable method to meet NRC requirements. The Committee issued a letter to the EDO dated November 12, 1999, recommending that DG-1093 be issued for public comment.

The staff has not yet issued DG-1093 for public comment, in part, due to objections raised by the Office of the General Counsel (OGC). OGC has informed the staff that the proposed changes would require rulemaking. The staff and NEI are working to resolve the issues raised by OGC and suggest that a September 2000 ACRS briefing on the proposed final version of the Regulatory Guide would be appropriate. The staff plans to update the Chairman's Tasking Memorandum to reflect the revised schedule.

20. Control Room Habitability (Open) (TSK/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the ACRS. The NRC staff has initiated an effort to resolve issues associated with control room habitability, primarily due to problems with uncontrolled inleakage. The NEI had developed a draft guidance document (NEI 99-03: "Control Room Habitability Assessment Guidance") in August 1999, and the staff had identified significant concerns with the NEI guidance document. The Severe Accident Management Subcommittee reviewed this matter during its September 16-17, 1999 meeting, and the Subcommittee Chairman provided a report to the ACRS during the October meeting.

Given the significant staff concerns with the initial draft of 99-03, NEI has now formed a Task Force to resolve this issue. NEI and the NRC staff held a meeting on January 13, 2000 to kick off this resolution effort. A set of NRC/Industry Subgroups have been formed and will meet over the next 5-6 months to resolve all open issues and provide a revised 99-03 document for staff review by December of this year. Based on discussions with NRC, NEI, and Dr. Kress (cognizant Subcommittee Chairman), a meeting of the Severe Accident Management Subcommittee has been scheduled for April 25-26, 2000 with ACRS review during its May meeting.

IV. POTENTIAL ITEMS FOR THE 471st, APRIL 6-8, 2000 ACRS MEETING

21. Proposed Resolution of Generic Safety Issue (GSI) 173A: Spent Fuel Storage Pool for Operating Facilities (Open) (TSK/MME) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the NRC staff. This GSI was identified in August 1994 and was assigned a HIGH priority ranking in June 1996. This issue involved the potential for a sustained loss of spent fuel pool (SFP) cooling capability, which

was identified through the report filed with the NRC relating to Susquehanna, and the potential for substantial loss of SFP coolant inventory. Postulated adverse conditions that may develop following a LOCA or a sustained loss of power to SFP cooling system components could prevent restoration of SFP decay heat removal. The staff is in the process of revising the guidance documents for spent fuel storage design (i.e., portions of SRP 9.1.3 and Regulatory Guide 1.13). Currently, the staff is working with the industry to revise ANSI/ANS -57.2 Standard that contains guidance for SFP design for operating plants. The staff plans to incorporate the changes from the Standard, as appropriate, into SRP and Regulatory Guide. The staff plans to provide the resolution package to the ACRS in early March 2000 and requests to brief the ACRS in April 2000.

22. Special Studies for Risk-Based Analysis of Reactor Operating Experience (Open) (MVB/GA/MTM) ESTIMATED TIME: 1 hour

Purpose: Information Briefing

Briefing Requested by the ACRS. The ACRS last heard a briefing on NRC programs for risk-based analysis of reactor operating experience (performance indicators, accident sequence precursors, common-cause failures, special studies, etc) in November 1997. The Subcommittee on Reliability and PRA was last briefed by the staff on November 20, 1998, concerning the use of performance indicators associated with proposed improvements to NRC inspection and assessment programs.

The staff recently issued reports on several special studies and has offered to brief the Committee on the following studies: D.C. Cook risk-assessment of high-pressure safety injection system reliability, Westinghouse and General Electric reactor protection systems, and component performance study of turbine driven pumps. Reports of these studies were forwarded to the Committee on August 20, 1999. The staff also expects to discuss future plans regarding the risk-based analysis of reactor operating experience programs. The staff requests to brief the Committee during the April 6-8, 2000 meeting.

As recommended by the Planning and Procedures Subcommittee, Dr. Apostolakis agreed to propose a course of action following the meeting of the Subcommittee on Reliability and Probabilistic Risk Assessment on December 15, 1999.

23. TRACG Best-Estimate Thermal-Hydraulic Code (Open) (GBW/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review Requested by the NRC Staff. Generic Electric has recently submitted documentation associated with its best-estimate large-break LOCA code, TRACG, to the NRC staff for review and approval. The initial application will be

for modeling of "Anticipated Operational Occurrences" only. NRR has begun an acceptance review of the GE submittal. The T/H Phenomena Subcommittee is scheduled to be briefed on the status of NRR's review during its March 14-15, 2000 meeting. Dr. Wallis will report on the results of this meeting to the Committee during its April meeting.

The Planning and Procedures Subcommittee recommends that Dr. Wallis propose a course of action. Dr. Wallis will propose a course of action subsequent to the March 14-15, 2000 Subcommittee meeting.

24. Reactor Trip and Partial Loss of AC Power at Indian Point Unit 2 (Open) (JJB/MTM) ESTIMATED TIME: 1 hour

Purpose: Information Briefing

Briefing requested by the ACRS. On August 31, 1999, Indian Point Unit 2 tripped from full power when all offsite power was lost to the plant's vital 480VAC Buses. All three EDGs started and loaded to the safety Buses. However, EDG-23 output breaker tripped on overcurrent resulting in loss of the 6A Bus. The subject Bus remained deenergized for an extended period of time which resulted in the loss of one 125VDC Bus and one 120VAC instrument Bus. Associated equipment losses included one train of motor-driven auxiliary feedwater (AFW), turbine-driven AFW due loss of DC flow control valve, one power-operated block valve thereby degrading feed-and-bleed capability, one EDG and vital battery, and most annunciators. The cause of this event was an uncorrected but known vulnerability of the $OT_{\Delta}T$ (over-temperature delta temperature) trip channel to noise spikes. Another channel was already in the trip position for maintenance (e.g., bistable replacement). Failure to recover the 6A Bus was attributed to deficiencies in command and control by the control room operators and the lack of reliable battery depletion time estimates. The NRC dispatched an Augmented Inspection Team to the site. The AIT report was forwarded to the Committee on November 1, 1999.

During the November 1999 ACRS meeting, the Committee decided to hear a briefing on this event at a future ACRS meeting. Time permitting, this briefing will be scheduled for the April 2000 ACRS meeting.

25. Proposed Final ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications - Phase 1 (Open) (GA/MTM) ESTIMATED TIME: 2 hours

Purpose: Review and Comment

Review schedule specified in CTM. The Committee met with representatives of ASME and the NRC staff regarding the proposed ASME Standard for probabilistic risk assessment for nuclear power plant applications (Phase 1). The Committee provided a report to the EDO on March 25, 1999, supporting this

initiative and requesting to review the proposed final Standard following the reconciliation of public comments.

ASME issued the Phase 1 Standard for public comment in parallel with the March 1999 ACRS review. A large number of public comments were received and ASME has made substantial changes to the proposed Standard. ASME has held a workshop in October 1999. It plans to resolve public comments and develop a proposed final version of the Standard and submit the Standard for consensus review in January-February 1999.

ASME has offered to brief the ACRS Subcommittee on Reliability and PRA in late March 2000 and brief the Committee during its April 6-8, 2000 ACRS meeting. Dr. Apostolakis has requested that technical experts from ASME be invited to participate in the Subcommittee deliberations.

26. Proposed White Paper on Risk-Based Performance Indicators (Open) (GA/MTM)
ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the NRC staff. As part of the proposed improvements to the NRC inspection, assessment, and enforcement programs, the staff is developing risk-based performance indicators (PIs). This includes the trial application of PIs and the identification of thresholds for regulatory action. The staff has started trial application of risk-based PIs through the pilot program associated with the revised inspection, assessment, and enforcement programs in June 1999. The staff plans to hold a public workshop in February 2000, to discuss candidate PIs, brief the Commission in October 2000, and implement approved PIs in January 2001.

The Committee reviewed the staff's development of risk-based performance indicators during the June 1999 meeting and provided a letter to the EDO dated June 10, 1999, concerning this matter and pilot applications of the revised inspection and assessment program, performance-based initiatives, and related matters. During the October 1999 ACRS meeting, the Committee decided to consider possible future meetings regarding this matter only if requested by the Commission. Subsequently, the EDO's staff has discussed this matter with Commissioner's Technical Assistants, who agreed that the Commission would be interested in having ACRS advice on technical issues (e.g., risk-based PIs) associated with the new reactor oversight process.

During the November meeting, the Committee decided to hear a briefing on this matter, and subsequently decide whether there are any technical issues that should be brought to the attention of the Commission. The staff expects to provide its draft White Paper in early February 2000 and requests to brief the Subcommittee on Reliability and PRA in late March 2000 with full Committee review during its April 6-8, 2000 meeting.

27. Proposed Approach for Revising 10 CFR 50.61, Pressurized Thermal Shock Rule (Open) (WJS/NFD/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment.

Review Requested by the NRC Staff. The NRC staff last briefed the Committee on this subject on July 14, 1999. The NRC staff is continuing with its development of a technical basis for promulgating a risk-informed revision to 10 CFR 50.61. The staff requests to brief the Committee at the April 2000 meeting on interim results and progress achieved by the RES on (1) the PTS analysis methodology development (PFM computer code, PRA uncertainty analyses and acceptance criteria development, and probabilistic fracture mechanics analysis method/input/issues), (2) the flaw distribution experts' elicitation process, and (3) thermal-hydraulic analysis (issues and status). A joint meeting of the Materials and Metallurgy and Thermal Hydraulic Phenomena Subcommittees will be held on March 16, 2000 to discuss this matter. A Subcommittee report is scheduled for the April ACRS meeting.

28. RETRAN-3D Transient Analysis Code (Open) (GBW/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the NRC staff. As part of its Thermal-Hydraulic (T/H) Code Review Action Plan, NRR initiated review of the EPRI RETRAN-3D thermal-hydraulic transient analysis code. The code is designed for analysis of FSAR Chapter 15 transients (excluding Appendix K LOCA analysis), and plant events. The T/H Phenomena Subcommittee began its review of this code during its December 16-17, 1998 meeting.

NRR had developed a detailed schedule for reviewing the RETRAN-3D code. In accordance with this review schedule, the T/H Phenomena Subcommittee met on March 23, 1999. A Subcommittee report was provided to the Committee during its April 1999 meeting.

Dr. Wallis conducted a detailed review of portions of the RETRAN code documentation. He has identified several issues of a significant nature with the models and correlations used in the "3D" version of the code. NRR has also identified a number of significant issues regarding the code modeling. In addition, EPRI was required to modify its "five equation" flow model to correct known errors. A meeting was held on June 29, 1999 between NRR and EPRI to address these matters. The outcome of the meeting gave indication that a significant amount of work remains before completing the review of this code.

Dr. Wallis provided a report to the Committee during the July 1999 ACRS meeting regarding his concerns. The Committee considered a draft letter to the EDO on this matter, but the letter was tabled. During the September meeting the Committee discussed the direction to be taken by the ACRS regarding future

review of the RETRAN-3D Code. It was agreed that the Committee would defer further action on this matter, pending receipt of the staff's review document.

The staff plans to develop a review document subsequent to receiving the EPRI response to a set of RAIs that include questions from Dr. Wallis and the T/H Phenomena Subcommittee. NRR plans to report the status of the RETRAN review to the T/H Phenomena Subcommittee during its March 14-15, 2000 meeting. Dr. Wallis will report the results of this briefing to the ACRS during its April meeting and propose a course of action.

V. POTENTIAL ITEMS FOR FUTURE ACRS MEETINGS

29. Steam Generator Tube Integrity Issues (Open) (RLS/NFD) ESTIMATED TIME: 2 hours

Purpose: Review and Comment

Review requested by the NRC staff. The proposed generic letter and regulatory guide associated with steam generator tube integrity was reviewed by the Committee to Review Generic Requirements on July 21, 1998. In a memorandum dated September 11, 1998, to the EDO, the staff proposed to delay issuance of the proposed generic letter for three months while it worked with industry to reach agreement on the content of industry guidelines. The staff issued the draft regulatory guide, "Steam Generator Tube Integrity," and the differing professional opinion (DPO) response for public comment on January 20, 1999. The Materials and Metallurgy Subcommittee heard a status briefing on this issue at its March 24-25, 1999 meeting. The staff plans to resolve GSI-163, "Multiple Steam Generator Tube Leakage," on the basis of the information contained in the DPO resolution package.

The Committee issued a letter on April 22, 1999, regarding the status of resolution of steam generator tube integrity issues. In a Larkinsgram dated July 22, 1999, the ACRS informed the EDO about its decision not to review the latest draft of the DPO consideration document.

The staff and NEI met on August 27, 1999, to discuss NEI's proposed changes to the Standard Technical Specifications and Technical Requirements Manual. These changes would require licensees to have a steam generator tube inspection program. They also discussed the use of NEI 97-06, "Steam Generator Guidelines" in the regulatory process. NEI is expected to submit these documents for NRC review in February 2000. The staff plans to provide the ACRS with draft safety evaluations related to these documents in early April 2000 and brief the full Committee at the May 2000 ACRS meeting. Subsequently, the staff plans to develop a Regulatory Issues Summary that endorses the use of NEI 97-06.

30. Fire Protection Issues (Open) (DAP/AS) ESTIMATED TIME: 2 hours

Purpose: Review and Comment:

Review requested by the Staff. The Fire Protection Subcommittee was briefed on January 20-21, 1999, on a draft NFPA 805 Standard, outline for comprehensive Regulatory Guide on Fire Protection, and the status of research program on fire risk. The Fire Protection Subcommittee plans to hold a meeting in May 2000 to discuss the revised NFPA 805 Standard, the draft comprehensive Regulatory Guide, post-fire safe shutdown circuit analysis, and other fire protection related issues. These matters will be scheduled for the June 2000 meeting for full Committee discussion.

31. Development of ACRS Code Review Guidelines (Open) (GBW/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Structured Presentation

ACRS Initiative. During the February 1999 ACRS meeting, the Committee discussed the future activities of the Thermal-Hydraulic (T/H) Phenomena Subcommittee. The ACRS Chairman suggested that the T/H Phenomena Subcommittee reevaluate the Committee's approach to review the best-estimate codes. He suggested that the Subcommittee recommend criteria and bases to be applied to the Committee's review of these codes. The Subcommittee briefly discussed this suggestion during its February 23, 1999 meeting, and Dr. Wallis is expected to develop a review plan to address this matter. An ACRS structured discussion by Dr. Wallis was to be held during the November ACRS meeting, but was postponed due the press of higher priority business. Dr. Wallis is scheduled to lead a Committee discussion on this matter during the January 27-29, 2000 Planning and Procedures (P&P) Subcommittee meeting. Future action by the Committee will pend the results of the P&P meeting discussions.

32. Proposed Update to 10 CFR Part 52 (Open) (REU/MME) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the NRC staff. Based on the insights gained from the review of the ABWR, CE-System 80+, and AP600 designs, and the comments received on 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants," the staff is in the process of updating 10 CFR Part 52.

This update is expected to be of an administrative and procedural nature. The staff has received a 6-months extension form EDO for completing the proposed update to 10 CFR Part 52. The staff expects to provide the document to the ACRS in May 2000 and brief the full Committee in June 2000.

The Planning and Procedures Subcommittee recommends that Dr. Uhrig provide his views on the need for the Committee to review this matter after receiving the document.

33. Use of Voluntary Initiatives In the Regulatory Process (Open)(TSK/NFD) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the ACRS. In SECY-99-063, dated March 2, 1999, the staff proposed that guidelines for crediting voluntary initiatives be developed with contributions from affected stakeholders. The Commission, in a Staff Requirements Memorandum dated May 27, 1999, directed the staff to move forward, working with industry and other stakeholders, in the development of the process and guidelines for use of industry initiatives in the regulatory process.

The staff held a meeting for internal stakeholders on September 8, 1999. The staff held a meeting for external stakeholders on October 27, 1999, in Chicago, Illinois. The staff plans to develop a Commission Paper on the process for voluntary industry initiatives during May 2000.

The Committee decided to review this matter at a future meeting when the proposed Commission paper is available.

34. Proposed Regulatory Guide and Standard Review Plan Section Associated with NRC Code Reviews (Open) (GBW/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review Requested by the NRC Staff. The NRC Staff has been developing a regulatory guide and a Standard Review Plan (SRP) Section to document a set of general principles applicable to the staff's review of all analytical computer codes. This effort was driven primarily by problems identified by the NRC staff (Maine Yankee Lessons Learned) and the ACRS during its review of the AP600 passive plant design.

Draft (pre-decisional) versions of the proposed regulatory guide and SRP Section have been made available to the T/H Phenomena Subcommittee Members and Consultants. The Subcommittee held a meeting on November 17, 1999 to begin review of this matter. Dr. Wallis reported the results of this meeting during the December ACRS meeting.

The NRC staff intends to issue the guideline documents for public comment during January. Review by the ACRS has, therefore, been postponed. NRR will provide a status report on this matter during the March 14-15, 2000 T/H Phenomena Subcommittee meeting. Review of these documents by the ACRS now appears timely during its May meeting.

35. Siemens S-RELAP-5 Best-Estimate Large-Break LOCA Code (Open/Closed) (GBW/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review Requested by the NRC Staff Siemens Corporation plans to submit its version of a large-break LOCA best-estimate code for NRC review around April 2000. This code will "compete" with the Westinghouse best-estimate large-break LOCA code, which has been approved by the NRC staff.

Dr. Wallis has agreed to propose a course of action after receiving the documents.

36. Revised Source Term - Proposed Final Regulatory Guide and Standard Review Plan Section (Open) (TSK/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review schedule specified in CTM. The ACRS reviewed the final version of the revised source term rule and the draft regulatory guide and Standard Review Plan Section during its September 1999 meeting. The draft Guide and SRP Section were subsequently issued for public comment.

Based on the current schedule for receipt and reconciliation of public comments, review of the proposed final versions of these documents by the ACRS appears timely during its May 2000 meeting.

37. Proposed Revision 1 to Regulatory Guide (RG) 1.78 (DG-1087), Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (Open) (TSK/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review Requested by the NRC Staff. This RG is being revised with the goal of reducing regulatory burden. Changes proposed include replacing the concentration limits for toxic gases with the limits specified by the Occupational Safety and Health Administration. Updated data on transportation statistics and spills were reviewed for inclusion in the Regulatory Guide.

The ACRS reviewed the proposed version of RG 1.78 during its September meeting. The Committee made three recommendations: (1) the Guide should be redrafted to facilitate risk-informed license amendment requests to eliminate technical specification requirements for toxic gas monitoring systems, (2) the staff should consider providing performance-based guidance to licensees rather than prescriptive guidance in the proposed Regulatory Guide, and (3) the staff should document evidence of the validity and the capability of computer codes endorsed in regulatory guides such as the HABIT code endorsed in this

proposed Regulatory Guide. The staff plans to revise this RG in response to Items (1) and (2) above and issue it for public comment this spring.

Based on the above staff schedule, ACRS review of the final version of RG 1.78 looks timely during the September meeting.

38. Proposed Refinements to Standard Review Plan Section 3.9.8 and Regulatory Guide 1.178 for Risk-Informed Inservice Inspection of Piping (Open) (GA/MTM)
ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review Requested by the NRC Staff. The Committee reviewed proposed Standard Review Plan (SRP) Section and draft Regulatory Guide 1.178 (DG-1063) for risk-informed inservice inspection of piping in June 1998 and issued a report to Chairman Jackson dated June 12, 1998, recommending that the subject documents (SECY-98-139) be issued in final form rather than for "trial use." The Commission subsequently approved the staff's proposal to issue these documents for trial use.

The staff expects to provide proposed final refinements to SRP Section 3.9.8 and Regulatory Guide 1.178 in early May 2000 and requests to brief the Subcommittee on Reliability and PRA in late May 2000 and the full Committee during the June 7-9, 2000 ACRS meeting.

39. Standard Development for PRA Quality - Phase 2 (Open) (GA/MTM)
ESTIMATED TIME: 2 hours

Purpose: Review and Comment

Review schedule specified in CTM. The NRC staff previously requested the American Society of Mechanical Engineers (ASME) to develop a Standard for use by industry in standardizing and upgrading their PRAs to facilitate risk-informed decisionmaking. ASME created a Task Force which includes representatives of the American Nuclear Society (ANS), the Nuclear Energy Institute (NEI), individual utilities, and the NRC staff. The ASME Task Force briefed the Committee on the draft Phase 1 Standard during the March 10-13, 1999 ACRS meeting. The Committee provided a letter to the EDO, dated March 25, 1999, on this matter.

The Task Force plans to pursue Phase 2 Standard development. On February 18, 1999, ANS sent a letter to ASME informing them that the ANS is preparing to begin the Phase 2 initiative. This letter was forwarded to the Committee on February 23, 1999. A definitive schedule for developing Phase 2 Standard has not been established. As soon as the draft Phase 2 Standard is made available to the Committee, it will be scheduled for ACRS review.

40. Proposed Revisions to Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI Division 1," to Endorse Risk-Informed Code Cases (Open) (WJS/GA/MTM) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review schedule specified in CTM. The American Society of Mechanical Engineers (ASME) expects to complete its risk-informed Code Cases for inservice inspection of piping. The NRC staff plans to endorse this Code Cases via a proposed revision to Regulatory Guide 1.147, with appropriate exceptions and clarifications. The staff plans to discuss the proposed revision to Regulatory Guide 1.147 with the joint Subcommittees on Reliability and Probabilistic Risk Assessment and on Materials and Metallurgy in August 2000 and brief the full ACRS in September 2000.

41. Generic Safety Issue-168, "Environmental Qualification of Electrical Equipment" (Open) (REU/AS) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the NRC staff. During the April 1998 ACRS meeting, the Committee decided not to hear a briefing on the draft Commission paper regarding the environmental qualification (EQ) task action plan. The staff provided the draft Commission paper to the ACRS on February 18, 1998. This information paper provides a status update of the staff's progress in resolving Generic Safety Issue (GSI)-168. This GSI involves an assessment of risk impact on EQ, a programmatic review of EQ, and research to investigate uncertainties associated with accelerated aging and cable condition monitoring. In a memorandum dated April 9, 1998, the ACRS Executive Director informed the EDO that the ACRS plans to review the resolution of GSI-168 when it is available. The low voltage cable testing was completed by December 1999. Additional test for 60-year plant life is expected to be completed in March 2000. GSI-168 is scheduled for resolution in September 2000. The staff plans to provide the resolution package to the ACRS in August 2000 and brief the full Committee in September 2000.

During the March 1998 ACRS meeting, Dr. Kress suggested that members also review a recent Information Notice 97-45, Supp. 1, "Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails." The subject Information Notice was provided to the members during the March 1998 ACRS meeting.

42. Reactor Water Cleanup System Line Break Outside the Containment (Open) (JJB/MME) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

This issue was raised by the ACRS in its report on the ABWR design. The NRC staff performed a study regarding the effects of the Reactor Water Cleanup (RWCU) System line break outside the primary containment for three representative BWR plants, each with a different containment design. During the June 1997 ACRS meeting, the staff briefed the Committee regarding the results and conclusion of such study.

During the June 1997 meeting, the Committee expressed concerns regarding the scope of the study, the inclusion of the resolution of Generic Issue 87, "Failure of HPCI Steam Line Without Isolation," effects of break location and size, water cascades and water droplets, and the effects of alternate plant configurations. The staff is in the process of addressing the ACRS concerns.

The Committee decided to discuss this matter after the staff has resolved the ACRS concerns.

43. Proposed Regulatory Guide for Lightning Protection at Nuclear Power Plants (Open) (REU/AS) ESTIMATED TIME: 1 hour

Purpose: Review and Comment

Review requested by the NRC staff. On October 8, 1996, the ACRS Subcommittee on I&C Systems and Computers was briefed on the status of staff actions to develop a regulatory guide for equipment protection against lightning and other transients. NRR prepared a user-need request on this matter and RES prepared a request to procure contract technical assistance. At this time, no work is being performed on this issue and it is on hold indefinitely.

44. Extended Shutdown of Millstone Units 1, 2, and 3 (Open) (JJB/MTM) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the ACRS. Millstone Unit 1 is a 670 Gross MWe General Electric boiling water reactor (BWR) with isolation condensers. It was shut down in November 1995 and remains defueled. Millstone Unit 2 is a 930 Gross MWe Combustion Engineering pressurized water reactor (PWR). It was shut down in February 1996 and remains in Mode 6, REFUELING. Millstone Unit 3 is a 1184 Gross MWe Westinghouse PWR with large, dry containment. It was shut down in March 1996. In June 1996, Millstone was designated a Category 3 facility (shutdown requiring Commission approval to operate).

The licensee has undergone significant management and organizational changes. In November 1996, the NRC has established a Special Projects Office (SPO) and an associated Restart Action Plan. SECY-97-003, "Millstone Restart Preview Process," was forwarded to the Committee on January 13, 1997. Millstone Unit 3 restarted on July 30, 1998. Unit 2 is resumed full power

operations in May 1999. The licensee has decided not to restart Unit 1 and has begun decommissioning activities.

During the April 1999 ACRS meeting, the Committee suggested that the Plant Operations Subcommittee recommend a course of action with regard to ACRS review of the plants that have been shut down for more than a year.

45. Use of MAAP Code for Severe Accidents (Open/Closed) (TSK/PAB)
ESTIMATED TIME: 2 hours

Purpose: Review and Possible Comment

Review requested by the ACRS. Dr. Kress recommended a presentation by industry on the MAAP code in order to understand how the code handles containment heatup, reactor coolant boil-off, and transport of fission products, given that this code is in wide use by the nuclear power industry. The present version of the code (Version 4) has undergone limited review by NRR in conjunction with the AP600 design review.

Subsequent discussions with Dr. Kress have led to a request for relevant documentation on the details of the code models. This material has been provided to Dr. Kress. After reviewing this information Dr. Kress will propose a course of action.

46. Proposed Revision of Appendix K to Allow Modification of the ANS Decay Heat Standard (Open) (GBW/PAB)

Purpose: Review and Comment

Review proposed by the ACRS. By a letter from A. Thadani, RES, to S. Collins, NRR, RES is proposing that the two Offices discuss reducing the conservatism inherent in the use of the 1971 ANS Decay Heat Standard, dictated by Appendix K, by use of a later version of the Standard (e.g., 1979, 1994). The modification is being proposed to reduce licensee burden. RES sees the proposal as relatively straightforward, since the Standard and associated uncertainty evaluation exist in the form of a "package". The proposal would be to effect a rule change to replace the 1971 Standard and would include a requirement for an upper uncertainty bound similar to the "ANS-plus-20% requirement that currently exists in Appendix K. It is estimated that this action could allow each nuclear power plant licensee to increase core power by at least 1%.

During its June meeting, the Committee stated that it would wait for NRR's reaction to the RES proposal before taking action. The Committee also said that NRR should evaluate the impact of use of mixed-oxide fuel on this proposal.

NRR has indicated that this issue has low priority at this time, given a lack of industry interest. It will be considered as part of the "Option 3" effort, pursuant to the SECY-98-300 paper regarding options for risk-informing 10 CFR Part 50.

47. WCOBRA/TRAC Small-Break LOCA Best-Estimate Code (Open/Closed) (GBW/PAB) ESTIMATED TIME: 2 Hours

Purpose: Review and Comment

Review requested by NRC staff. Westinghouse has submitted an application to apply its best-estimate methodology to small-break LOCA analyses, using its WCOBRA/TRAC code. A meeting of the Thermal-Hydraulic Phenomena Subcommittee was held on November 19, 1998 to be briefed on this matter, and for Westinghouse to obtain the Subcommittee's initial reaction.

The Subcommittee made a number of comments/requests with regard to the modeling approach proposed and the details of specific code models. Westinghouse noted that the date for submittal of its code documentation has slipped to the March time frame.

Based on the above schedule change, the next meeting of the Thermal-Hydraulic Phenomena Subcommittee to review this matter appears timely in April/May 2000.

48. Safety Culture (Open) (GA/JDP/JS) ESTIMATED TIME: 1 ½ hours

Purpose: Structured Presentation

ACRS Initiative. The staff prepared a Commission Paper related to assessing the safety culture at operating nuclear power plants. In the Commission paper, the staff included five options and recommended discontinuing any further agency efforts. In a Staff Requirements Memorandum (SRM) dated September 1, 1998, the Commission approved the continuation of the current policy that safety culture be evaluated only on a for-cause basis.

On November 19, 1999, Mr. Sorensen, ACRS Senior Fellow, briefed the Human Factors Subcommittee on this subject.

During the October 1999 ACRS meeting, the Committee decided to discuss this item at the January 2000 ACRS retreat.

49. ASME Piping Code (Open) (WJS/NFD) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Possible Comment

Review requested by then Commissioner Rogers. During a meeting with Dr. Kress on February 8, 1995, commented on the appropriateness of the 1994 Addenda for Class 1, 2, and 3 piping systems to the ASME Boiler and Pressure Vessel Code Section III, concerning piping supports.

Mr. Russell, then NRR Director, and Mr. Beckjord, then RES Director, sent a letter to the ASME Board on Nuclear Codes and Standards taking strong exception to this Addenda. Mr. Sheron, NRR, sent a second letter to the Board on May 24, 1995. During the ASME Code Committee meeting in September 1995, the ASME Section III Special Working Group on Seismic Design Rule (SWG-SR) was established to review the 1994 Addenda.

The Materials and Metallurgy Subcommittee discussed this issue with representatives of the NRC staff, ASME, and Japan during a meeting on June 1, 1998, and decided to review this issue after the SWG-SR reaches a consensus and the Japanese compete their analyses of piping fatigue failure tests. The Subcommittee heard a briefing on the status of the SWG-SR activities at the March 24-25, 1999 Materials and Metallurgy Subcommittee meeting.

50. Proposed Revisions to SRP Chapter 18, Human Factors Engineering (Open) (GA/NFD) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

Review requested by the NRC staff. The NRC staff has offered to brief the Committee on the proposed revisions to Standard Review Plan Chapter 18. These revisions would reflect the changes in the requirements for Human Factors Engineering programs. Previously, the staff planned to provide a copy of the proposed revisions to the ACRS in September 1999 and brief the Committee at the November 1999 ACRS meeting. Based on recent discussions with the staff, it seems that completion of these revisions will be postponed to FY 2000.

Attachments:

A - pp. Memorandum dated January 27, 2000, from James L. Blaha, OEDO, Subject: Proposed Agenda Items for the ACRS and the ACNW.

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

January 27, 2000

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

FROM: James L. Blaha
Assistant for Operations
Office of the Executive Director for Operations

SUBJECT: PROPOSED AGENDA ITEMS FOR THE ACRS AND THE ACNW

Attached is a listing of proposed agenda items for the ACRS for the months of March 2000 - June 2000. Also, attached is a listing of the proposed ACNW agenda items for February 2000 - May 2000.

We have attached an annotated copy of our Work Items Tracking System (WITS) for the upcoming three month period. In addition, we have provided a projection of office originated Commission papers that may also be of interest to the ACRS/ACNW. If there are particular items identified out of the field of projected Commission papers that we had not planned to bring to the ACRS/ACNW for formal review or briefing, but that are of Committee(s) interest, we would appreciate receiving timely feedback on such preferences.

Attachments: As stated

ACRS MEETING - MARCH 2-4, 2000

TITLE/ISSUE	PURPOSE	PRIORITY	DOCUMENTS
Supplemental SER for Oconee Licensee Renewal (C. Grimes, NRR)	Review and Comment	High	To be provided 2/12/00.
GSI 173A Spent Fuel Pool Cooling Issue for Operating Plants (D. Diec, NRR)	Review and Comment	High	Expected: 2/3/00. Response due: Within 4 weeks of meeting.
GSI B-17, "Criteria for Safety Related Operator Actions (P. Lewis, RES)	Review and Comment	High	To be provided. Response due from ACRS: 3/17/00.
Phenomenon Identification and Ranking Tables (PIRTs) for High Burnup Fuel (R. Meyer, RES)	Review and Comment	High	www.NRC.gov/RES/PIRT
Final Regulatory Guide for 10 CFR 50.65(a)(4) (W. Scott, NRR)	Review and Comment	High	Document expected to be provided 2/3/00.
Draft Regulatory Guide on and inspection guidance on 10CFR 50.59 (E. McKenna, NRR)	Review and Comment	High	To be provided 2/15/00. Response by 4/15/00.

ACRS MEETING - APRIL 6-8, 2000

TITLE/ISSUE	PURPOSE	PRIORITY	DOCUMENTS
Guidance (SRP/REG. Guide) for Analytical Code Reviews (R. Caruso, NRR)	Review and Comment	High	To be provided 3/1/00. Response by 5/1/00.
SFP Accident Risk for Decommissioning Plans (D. Jackson, NRR)	Review and Comment	High	Expected 2/15/00. Response by 4/15/00.
NEI 97-06, Steam Generator Program Guidelines (J. Andersen, NRR)	Review and Comment	High	March 2000
Digital I&C Research Plan (S. Arndt, RES)	Review and Comment	High	To be provided 3/15/00. Response by 4/21/00.
Risk-Based Performance Indicator Development (H. Hamzehee, RES)	Review and Comment	Medium	White Paper overview 2/15. Response by 5/1/00.
Performance based risk informed fire protection standard for LWRs and related issues (E. Weiss, NRR)	Review and Comment	Medium	To be provided 2/1/00.
Special Studies for risk-based analysis of reactors operating experience (Baranoski, RES)	Information Briefing	Medium	

ACRS MEETING - MAY 11-13, 2000

TITLE/ISSUE	PURPOSE	PRIORITY	DOCUMENTS
Proposed Refinements to SRP Section 3.9.8 and RG 1.178 for Risk-informed ISI of Piping (S. Dinsmore, NRR)	Review and Comment	Unknown	To be provided 4/7/00. Response by 6/15/00.

ACRS MEETING - JUNE 7-9, 2000

TITLE/ISSUE	PURPOSE	PRIORITY	DOCUMENTS
DG-1053, "Dosimetry and Neutron Transport Calculations" (E. Hackett, RES)[1]	Review and Comment	High	To be provided 1 month before meeting
Digital I&C Draft Research Plan (N. Chokshi, RES)	Review and	High	To be provided 1 month before meeting

[1] This is being re-scheduled for mid-2000.

Performance Base Risk-Informed Fire Protection.

(25)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No:
469th	16	16.1

Title **MINUTES OF PLANNING & PROCEDURES SUBCOMMITTEE MEETING - FEBRUARY 2, 2000**

Authors **JOHN T. LARKINS**

List of Documents Attached

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Instructions to Preparer
1. Punch holes
2. Paginate attachments
3. Place copy in file box

From Staff Person

JOHN T. LARKINS

February 4, 2000

MINUTES OF THE
PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
WEDNESDAY, FEBRUARY 2, 2000

The ACRS Subcommittee on Planning and Procedures held a meeting February 2, 2000, in Room 2B1, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 1:00 p.m. and adjourned at 4:30 p.m.

ATTENDEES

D. A. Powers, Chairman
M. Bonaca, Member-at-Large

ACRS STAFF

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

NRC STAFF

Robert Jasinski, OPA (part-time)

DISCUSSION

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

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

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the February 2000 ACRS meeting be as shown in the handout. The Committee should complete its response to the issues raised by the Commission in the SRM dated December 17, 1999, regarding: Impediments to the Increased Use of Risk-Informed Regulation; Use of Importance Measures in Risk-Informing 10 CFR Part 50; and Technical Components of the Revised Reactor Oversight Process, including the Technical Adequacy of the current and proposed performance indicators. Since the Commission is awaiting ACRS comments on EDO response regarding the 120-month ISI/IST update requirement, the

Committee should complete a report on this matter. In addition, the Committee should complete its annual report to the Commission on the NRC Safety Research Program and response to questions raised by individual Commissioners following the ACRS meeting with the Commission on November 4, 1999.

Time permitting, the Committee should try to complete other reports and letters. The Committee should try to consider the proposed ACRS reports in the order listed on the board.

2) Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through April 2000 is included in a separate handout. The objectives are: (1) to review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate, (2) to manage the members' workload for these meetings, and (3) to plan and schedule items for ACRS discussion of topical and emerging issues.

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate.

3) Meeting with the Commission

The ACRS is scheduled to meet with the NRC Commissioners between 9:30 and 11:30 a.m. on Thursday, March 2, 2000 to discuss risk-informing 10 CFR Part 50 and related matters. Proposed subtopics for the meeting have been sent to the Office of SECY, requesting Commissioners' feedback (p. 1).

RECOMMENDATION

The Subcommittee recommends that the members with lead responsibility prepare proposed slides with input from other members, as needed, on topics assigned to them. Subsequent to receiving slides on all topics, the ACRS staff should send them to all members for review and comment. After incorporating the comments, as appropriate, and with the approval of the ACRS Chairman, slides should be sent to the Commission by February 18, 2000.

4) Schedule for the March ACRS Meeting

The March ACRS meeting is currently scheduled for March 2-4, 2000. It is anticipated that several letters scheduled for the February meeting will be deferred to the March meeting. In addition, the Committee is scheduled to meet with the Commission on March 2. There are other matters that should be reviewed by the Committee during the March meeting. In view of the heavy workload for the March meeting, the Committee should consider extending this meeting.

RECOMMENDATION

The Subcommittee recommends that after completing the ADAMS training on March 1, 2000, the Committee begin the meeting at 1:00 or 1:30 p.m. on Wednesday, March 1. On Wednesday, the Committee should first hear a presentation from the staff on the status of the proposed risk-informed revisions to 10 CFR Part 50, then complete as many reports and letters as possible that were deferred from the February meeting, and prepare for the meeting with the Commission scheduled for Thursday, March 2.

The Subcommittee recommends also that the next Planning and Procedures meeting be held on Tuesday, February 29 starting at 8:30 a.m. After completing the scheduled items, the Subcommittee and other members should discuss the matters scheduled for the meeting with the Commission. All the members are invited to attend the afternoon session of the Planning and Procedures Subcommittee meeting on February 29, 2000.

5) Follow-up Items Resulting from the January 27-29, 2000 ACRS Retreat

The positions agreed to during the ACRS retreat on several matters and the follow-up items resulting from the retreat were discussed.

RECOMMENDATION

The Subcommittee recommends that the Committee adopt the positions agreed to during the retreat, and take necessary steps to bring closure to the follow-up items expeditiously.

6) Status of Selecting Candidates for Potential ACRS Membership

In response to solicitation of candidates for ACRS membership, we have received about 20 applications. The ACRS Member Candidate Screening Panel has reviewed all of the applications. It has selected four best-qualified candidates for interview by the Panel and the ACRS members. We are in the process of establishing a schedule for the members and the Panel to interview three of these candidates during the March ACRS meeting.

RECOMMENDATION

The Subcommittee recommends that the ACRS Member Candidate Screening Panel provide the Committee the names and résumés of the candidates it has selected along with the interview schedule. The members should interview the candidates and provide feedback to the Panel for use in formulating recommendations to the Commission.

7) ACRS Self-Assessment Matrix

In an SRM dated August 6, 1999, the Commission stated that "the periodic self-assessment report and the ACRS and ACNW Operating Plans can be combined into one annual report to the Commission that should include self-assessment summary matrices." In order to prepare the matrix, the ACRS staff needs to summarize the comments and recommendations included in the ACRS reports, which may result in interpreting the Committee positions. The Committee should approve the matrix and

- motion made to grant the Executive Director authority to interpret letters reports into the self-assessment matrix; voted & agreed by Committee

self-assessment report to preclude any ambiguities and delegate the ACRS Executive Director to authorize or interpret Committee comments and recommendations.

RECOMMENDATION

The Subcommittee recommends that the Committee authorize the ACRS staff to prepare the self-assessment matrix, summarizing the Committee reports and recommendations and also delegate the ACRS Executive Director to authorize or interpret Committee comments and recommendations. (Note: The Committee needs to vote on this recommendation.) agreed!

8) Change in Travel Requirements for Federal Employees

Effective March 1, 2000, all Federal employees (including members) will be required to use their government issued credit card for all government travel expenses exceeding $75.

RECOMMENDATION

The Subcommittee recommends that members ensure that they use their NRC-issued government credit card for all travel expenses exceeding $75 (e.g., hotel rooms, plane tickets, rental cars).

9) Member Issues

(a) In response to an invitation from the RSK, the following members have requested ACRS endorsement to visit Siemens, in Germany (pp. 2-5):

R. Uhrig
G. Wallis
J. Sieber
M. Bonaca

agreed!

RECOMMENDATION

The Subcommittee recommends that the Committee approve the above travel requests.

(b) Dr. Uhrig plans to travel to South Africa in March 2000 on both personal and professional business (pp. 6-7) and as a part of that trip he will be meeting with ESCOM to discuss the Modular Pebble Reactor Design. He is requesting ACRS approval for the trip, without any financial support. (Note: We are obligated to pay for his time on official Committee business), so that his time with ESCOM is covered as official government business.

agreed!

RECOMMENDATION

The Subcommittee recommends that the Committee approve the above request.

(c) Dr. Kress is planning on attending a meeting in Marseilles, France, to discuss the results of the recent Phebus tests. He will be replacing Dr. Powers and representing the ACRS. The meeting will be in mid-March 2000 and he requested ACRS approval (pp. 8-9). agreed

(d) Dr. Powers is planning on attending for the Materials Research Society Annual meeting, on April 23-26, 2000, in San Francisco, California, regarding materials issues which arise in ACRS reviews of fuel, piping systems, corrosion, life extension, and accident phenomena (p. 10). agreed

RECOMMENDATION

The Subcommittee recommends that the Committee approve this request.

(e) Dr. Powers suggested that the ACRS discuss whether or not the Committee should be briefed by the NRC and DOE staffs on their current safeguards programs, particularly:

- DOE activities associated with terrorist actions involving chemical and biological agents (p. 11).

- NRC plans, if any, to prevent terrorist action at NRC-licensed facilities.

RECOMMENDATION:

The Subcommittee recommends that a Subcommittee meeting be held to discuss this matter with representatives of DOE and NRC.

5) Interviews of Prospective ACRS Members During the March ACRS Meeting

The proposed interview schedule is provided on pp. 12-13.

(10) Future Activities

See separate handout

G:\PlanPro\P&P464.WPD

Feb 4, 2000

motion was

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

February 1, 2000

MEMORANDUM TO: William M. Hill, Jr.
Office of the Secretary

FROM: Sam Duraiswamy, ACRS

SUBJECT: ACRS MEETING WITH THE NRC COMMISSIONERS - MARCH 2, 2000

The ACRS is scheduled to meet with the NRC Commissioners on March 2, 2000, between 9:30 and 11:30 a.m., to discuss risk-informing 10 CFR Part 50 and related matters. The following subtopics are proposed by the ACRS for discussion during this meeting.

a. Impediments to the further development of risk-informed regulation.

b. Importance measures, conservatisms, uncertainties, and the establishment of thresholds.

c. Technical foundations of the performance indicators.

We would appreciate the Commissioners' feedback on these topics as soon as possible.

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: Robert L. Davis Date Submitted: Dec 30, 1999

Dates of Planned Trip: Jun 24, 2000 to Jul 1, 2000

Destination Erlangen Germany

Meeting or Facility to be Visited: Meeting with Siemens Corp and German Nuclear Regulatory Authorities and Visit to Nuclear Plant with Siemens Digital I&C System

Purpose/Relevance to ACRS Business: ACRS continues to have concerns about hardware-software interactions, common mode failures, and human factors/human machine interface issues associated with digital I&C systems

Participation (Invited Speaker, paper presented, etc.): Meeting of selected ACRS members reviewing technical and safety aspects of digital I&C systems

Justification (Foreign Travel Only): ACRS has accepted the invitation of Siemens Corp to informally pre-establish communications with German regulatory authorities and to discuss regulatory aspects of digital I&C systems and human factors/human machine interface issues

NRC SUPPORT REQUESTED

Air Fare Yes ✓ No _____ Per Diem: Yes ✓ No _____ Days 6

Registration $ None Compensation: Yes ✓ No _____ Days 6

2

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: GRAHAM WALLIS Date Submitted: Jan 5, 2000

Dates of Planned Trip: June 25, 2000 to June 30, 2000

Destination: ERLANGEN, GERMANY

Meeting or Facility to be Visited: Siemens Corp, German Nuclear Regulatory Authority, Nuclear Plant Neckarwestheim

Purpose/Relevance to ACRS Business: Liaison with German NRC. Update on technical aspects of digital I&C, human factors, German policies and safety regulations

Participation (Invited Speaker, paper presented, etc.): Invitation from Siemens

Justification (Foreign Travel Only): Liaison with German nuclear industry and regulators. I used to be pretty fluent in German

NRC SUPPORT REQUESTED

Air Fare: Yes ✓ No ____ Per Diem: Yes ✓ No ____ Days 6

Registration: $ none Compensation: Yes ✓ No ____ Days 6

ACRS SPECIAL TRAVEL ENDORSEMENT FORM

THIS FORM IS TO BE USED TO REQUEST ACRS ENDORSEMENT OF SPECIAL TRAVEL REQUESTS BY MEMSERS 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 SUBCOMMITEE AT LEAST 60 DAYS PRIOR TO THE MEETING, IF POSSIBLE SUPPLEMENTAL INFORMATION MAY BE ADDED AS DETAILS DEVELOP

Member Name, John D. Sieber **Date Submitted:** ~~January 2, 2000~~ 01/04/00 JS.

Date of Planned Trip: **From:** June 24, 2000 **To:** July 1, 2000

Destination: Erlingen, Germany, and return.

- **Meeting or Facility to be visited:** To attend a meeting with German Nuclear Regulatory Authorities and the Siemens Company, related to digital I & C controls and the Operator interface with such controls.
- **Purpose/Relevance to ACRS Business:** The ACRS continues to have concerns about the reliability of digital I & C controls and associated software interactions, as well as the Operator/ Instrument interface that occurs in the Control Rooms of Nuclear Power Plants. We appreciate the advances in the human/machine interface, which has been brought about by digital I & C, but we need to investigate our concerns about the safety aspects of such integrated technology.
- **Participation (Invited Speaker, paper presented, etc.).** None
- **Justification** (Foreign Travel Only):

  The ACRS has accepted the invitation of the Sieman's Corp. to informally re-establish Communications with German Regulatory Authorities. We plan to discuss the regulatory aspects of digital I & C and human factors interaction with advanced digital control systems.

NRC SUPPORT REQUESTED

Air Fare: YES: X NO: ____.

Per Diem: YES X NO ____, Days: 6.

Registration: $ N/A.

Compensation: YES X NO ____, Days: 6

Signed: John D. Sieber ACRS Member

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Janet/Patty Disk Travel Form
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: MARIO V. BONACA Date Submitted: JAN 20, 2000

Dates of Planned Trip: JUNE 24, 2000 to JULY 1, 2000

Destination: ERLANGEN, GERMANY

Meeting or Facility to be Visited: MEETING WITH SIEMENS Co. & GERMAN NUCLEAR REGULATORY AUTHORITIES; VISIT TO NUCLEAR PLANT WITH SIEMENS DIGITAL I&C SYST

Purpose/Relevance to ACRS Business: ACRS HAS EXPRESSED CONCERNS WITH CERTAIN ISSUES ASSOCIATED WITH DIGITAL I&C SYSTEMS (HARDWARE/SOFTWARE INTERACTION, COMMON MODE FAILURES). THE MEETING WILL HELP IMPROVE ACRS UNDERSTANDING OF ISSUES, AND SOLUTIONS PROVIDED BY SIEMENS TECHNOLOGY.

Participation (Invited Speaker, paper presented, etc.): A LIMITED NUMBER OF ACRS MEMBERS

Justification (Foreign Travel Only): ACRS HAS ACCEPTED THE INVITATION OF SIEMENS TO INFORMALLY RE-ESTABLISH COMMUNICATIONS WITH GERMAN REGULATORS, AND TO UNDERSTAND THE GERMAN REGULATORY PERSPECTIVE ON THE USE OF DIGITAL I&C SYSTEMS IN NUCLEAR APPLICATIONS

NRC SUPPORT REQUESTED

Air Fare Yes ✓ No ____ Per Diem: Yes ✓ No ____ Days 6

Registration: $ NONE Compensation: Yes ✓ No ____ Days 6

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From: Robert Uhrig <ruhrig@utk.edu>
To: TWFN_DO.twf1_po(JTL)
Date: Thu, Dec 30, 1999 1:11 AM
Subject: Proposed Visit to South Africa

ROBERT E. UHRIG
5221 N. W. 44TH PLACE
GAINESVILLE, FL 32606-4328
December 30, 1999

Dr. John Larkins, Director
Advisory Committee for Reactor Safeguards
Two White Flint North
11542 Rockville Pike
Rockville, MD 20852

Dear John:

I respectfully request that the attached letter dealing with my proposed travel to South Africa be submitted to the Planning and Procedures Committee for consideration at their next meeting.

If I can provide any additional information, please let me know.

Sincerely yours,

Robert E. Uhrig
ACRS Member

ROBERT E. UHRIG
5221 N. W. 44TH PLACE
GAINESVILLE, FL 32606-4328
December 30, 1999

MEMORANDUM

To: ACRS P&P Committee

From: Robert E. Uhrig

Subject: Visit to South Africa for a Meeting with ESCOM and PBMR on Design and Safety Analysis of Modular Pebble Bed Reactor Design

I will be traveling to South Africa leaving March 7, 2000 to participate in a 14-day historical educational program sponsored by Elderhostel entitled "South Africa, Old and New: A Miracle in the Making." While there, I plan to spend some time, probably two days and possibly more, meeting with ESCOM, the national South African utility. As you know, ESCOM is proceeding with the design and is now committed to construction of the first of ten modular nuclear power plants of the pebble bed type. This 110 MWe plant is

based on the pebble bed experimental reactor developed by Germany almost two decades ago. I have been in contact with Mr. David Nicholls, Chief Executive Officer of PBMR, the ESCOM subsidiary that is actually designing and scheduled to build this plant, regarding the meeting arrangements. On the basis of our discussions so far, I anticipate that I will be given a thorough review of the design and safety analysis of this plant. The design is being optimized to make it competitive or better with fossil plants throughout the world. Clearly, this plant is unique in that it does not have a containment vessel. I feel that such a visit would be beneficial to me in my ACRS work. PBMR has an aggressive commercialization plan that envisions literally hundreds of these modules in the 2015-2020 time frame. Should this rapid development come to pass, it seems inevitable that someone will request NRC to review such a plant for installation in the United States.

My interest in the pebble bed graphite reactor system is a serious one. While with Florida Power and Light, I was the company representative to the Gas Cooled Reactor Associates (GCRA), a utility funded group promoting this gas cooled reactor technology. I was also Chairman of GCRA's Technical Advisory Committee and visited the prototype pebble bed test facility and THTR reactor in Germany about two decades ago.

I also feel that other ACRS members would also be interested in the design and safety analysis of this "next generation" plant. I would be pleased to share the results of this meeting with ACRS members and staff. Indeed, I would be willing to write a trip report on this visit with whatever distribution within NRC that P&P Committee deems appropriate. In turn, I am asking for some sort of formal ACRS recognition that participation in such a meeting with ESCOM and PBMR is of interest to ACRS. I am NOT asking for any financial support from ACRS. The only tangible benefit to me of such recognition would be a rationale for using my Delta Frequent Flyer miles (some of which may be due to ACRS travel) so that I can travel Business Class on South African Airlines, which is affiliated with Delta Air Lines on travel between the U.S. and South Africa. It is a long trip--15 hours nonstop from Atlanta to Cape Town and 17 hours from Johannesburg back to Atlanta with a refueling stop due to the Westerly winds.

I ask your favorable consideration of this request.

CC: GATED.nrcsmtp("apoatola@mit.edu","dapower@sandia.g...

Janet/Patty Disk:Travel.Frm
9/9/94

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Member Name: I. Kress Date Submitted: January 7, 2000

Dates of Planned Trip: March 18 to March 23

Destination: Marseille, France

Meeting or Facility to be Visited: Fourth Phebus FP Technical Seminar

Purpose/Relevance to ACRS Business: See attached

Participation (Invited Speaker, paper presented, etc.):

Justification (Foreign Travel Only): See attached

NRC SUPPORT REQUESTED

Air Fare: Yes X No ____ Per Diem: Yes X No ____ Days 6

Registration: $ No Compensation: Yes X No ____ Days 4

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T. S. Kress
Fax: (423)4827548

facsimile transmittal

To:	Tanya Winfrey	Fax:	301-415-5589
From:	T S Kress	Date:	01/21/00
Re:	Justification for 445 Form	Pages:	1 (including cover)
CC:	none		

☐ Urgent ☐ For Review ☐ Please Comment ☐ Please Reply ☐ Please Recycle

Tanya: Below is the information you need for the 445 form.

Travel to: Marseille, France
Leave: March 18, 2000
Return: March 23, 2000
Purpose: To attend the 4 th Phebus FP Technical Seminar

Justification:
The Phebus FP program is an international research program being participated in by NRC. It is the only current severe accident program designed to evaluate and provide validation to severe accident codes that deal with fission product release and transport in RCS and containment under core melt conditions.

Two members of ACRS (Dr. Powers and Dr. Kress) have been involved in the Phebus FP program since its inception – helping to evaluate NRCs participation, setting the program objectives, establishing the experiment matrix, and evaluating the results. The program has now arrived at its midpoint: three experiments have been successfully performed and the preparation for the next two experiments is at an advanced stage. Valuable scientific results have been obtained on core degradation and fission product behavior. This seminar will be a good opportunity to review the main results and future plans and keep ACRS abreast of this important area of reactor safety.

Cheers,
Tom Kress

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Member Name: Dana Powers Date Submitted: 2/Feb/00

Dates of Planned Trip: 23/Apr/00 to 26/Apr/00

Destination: San Francisco, Ca.

Meeting or Facility to be Visited: Materials Research Society annual mtg.

Purpose/Relevance to ACRS Business: Materials issues arise in ACRS reviews of fuel, piping systems, corrosion, life extension and accident phenomena

Participation (Invited Speaker, paper presented, etc.): Invited speaker

Justification (Foreign Travel Only):

NRC SUPPORT REQUESTED

Air Fare: Yes ✓ No ____ Per Diem: Yes ✓ No ____ Days 4

Registration: $ ____ Compensation: Yes ✓ No ____ Days 4

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From: John Larkins
To: "dapower@sandia.gov"@GATED.nrcsmtp
Date: Fri, Jan 14, 2000 5:33 PM
Subject: Re: Terrorist Plans at NRC and Licensees

No, we do not have an idea of current NRC staff thinking on this subject and it has been quite a while since we've had a briefing on this subject. I suggest we put this on the P&P Agenda to see if there is interest in having a briefing on current safeguards activities.

>>> "Powers, Dana A" <dapower@sandia.gov> 01/14 4:31 PM >>>

John,

I continue to see DOE doing a lot of planning for responses to terrorists actions involving chemical and biological agents. Do we know what NRC thinking on this is for its own facilities and for licensee facilities? It is very pertinent to the issue of defense and depth and people that have the hubris to think they can calculate everything or account for it with an uncertain parameter!

Dana

CC: SXD1

SCHEDULE FOR INTERVIEWING CANDIDATES
FOR ACRS MEMBERSHIP

THURSDAY, MARCH 2, 2000

CANDIDATE	SCREENING PANEL INTERVIEW ROOM 2E 2	ACRS MEMBERS INTERVIEW	
	TIME	TIME	MEMBERS
Dr. Robert A. Bari BNL - Upton, NY 516-344-2629	1:30 - 2:30 p.m.	12:00 - 12:30 12:30 - 1:00 LUNCH	GROUP 1* GROUP 2* (CAUCUS ROOM)
Mr. Louis F. Storz Public Service Electric & Gas 302-376-0329	2:45 - 3:45 p.m.	12:00 - 12:30 12:30 - 1:00	GROUP 2* GROUP 1* (CAUCUS ROOM)

GROUP 1

G. Apostolakis
J. J. Barton
M. V. Bonaca
T. S. Kress
R. L. Seale

GROUP2

D. A. Powers
G. B. Wallis
W. J. Shack
J. D. Sieber
R. E. Urich

DOCUMENT NAME: G:\INTERV.SD1

SCHEDULE FOR INTERVIEWING CANDIDATES
FOR ACRS MEMBERSHIP

FRIDAY MARCH 3, 2000

CANDIDATE	SCREENING PANEL INTERVIEW ROOM 2E 2	ACRS MEMBERS INTERVIEW	
	TIME	TIME	MEMBERS
Mr. Graham M. Leitch Baltimore Gas & Electric Co. Calvert Cliffs 610-344-2904	9:00 - 10:00 a.m.	12:00 - 12:30 12:30 - 1:00 Lunch	GROUP 1* GROUP 2* (CAUCUS ROOM)

GROUP 1

G. Apostolakis
J. J. Barton
M. V. Bonaca
T. S. Kress
R. L. Seale

GROUP 2

D. A. Powers
G. B. Wallis
W. J. Shack
J. D. Sieber
R. E. Urich

DOCUMENT NAME: G:\INTERV.SD2