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

July 3, 2001

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

Dear Chairman Meserve:

SUBJECT: SUMMARY REPORT - 483rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, JUNE 6-8, 2001 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 483rd meeting, June 6-8, 2001, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report and letters. In addition, the Committee authorized Dr. John T. Larkins, Executive Director, ACRS, to transmit the memoranda noted below:

REPORT

— Response to Your May 7, 2001 Memorandum Regarding Differing Professional Opinion on Steam Generator Tube Issues (Letter to Chairman Meserve, NRC, from George E. Apostolakis, Chairman, ACRS, dated June 14, 2001)

LETTERS

— Risk-Based Performance Indicators: Phase 1 Report (Letter to William D. Travers, Executive Director for Operations, NRC, from George E. Apostolakis, Chairman, ACRS, dated June 19, 2001)

— Response to Your April 12, 2001 Letter on Issues Raised by ACRS Pertaining to Industry Use of Thermal-Hydraulic Codes (Letter to William D. Travers, Executive Director for Operations, NRC, from George E. Apostolakis, Chairman, ACRS, dated June 19, 2001)

MEMORANDA

— Proposed Final Regulatory Guide, 1.52, Revision 3, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated June 11, 2001)

— Proposed Revision 1 to Risk-Informed Regulatory Guide 1.174 and Standard Review Plan Chapter 19 (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated June 12, 2001)

HIGHLIGHTS OF KEY ISSUES CONSIDERED BY THE COMMITTEE

1. Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50

The Committee heard presentations by and held discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) concerning the status of proposed risk-informed revisions to 10 CFR 50.46 for emergency core cooling systems (ECCS) and proposed revision to the framework for risk-informing the technical requirements of 10 CFR Part 50. The Committee considered the staff's preliminary views and schedule for completing its Phase 1 feasibility study for developing risk-informed alternative ECCS requirements. The Committee discussed candidate options to improve the realism of large-break loss-of-coolant accident (LBLOCA) analysis including possible LBLOCA redefinition. The Committee discussed LBLOCA phenomena and frequency; demonstration of functionality and performance-based acceptance criteria; realism of ECCS evaluation models; credible break sizes; and uncertainty propagation. The Committee also discussed possible Phase II technical work and policy issues, e.g., single-failure criterion and selective implementation. The staff plans to provide its draft Commission paper for consideration by the Committee in late-June 2001.

Committee Action

A joint meeting of the ACRS Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and PRA is scheduled for July 9, 2001, on this matter. The Committee plans to continue its review of this matter during the July 11-13, 2001 ACRS meeting, subject to the availability of the staff's proposed Commission paper.

2. Potential Margin Reductions Associated with Power Uprates

The Committee was briefed by the ACRS Senior Fellow Dr. A. W. Cronenberg on his views with regard to the adequacy of the staff's review process for power uprates and potential safety margin reductions associated with power uprates. Key points noted by Dr. Cronenberg were:

- The NRC's General Design Criteria do not explicitly address how much design margin is required. Typically, the words "sufficient margin" are used. Margin requirements are more explicit in the Standard Review Plan and in such documents as industry Codes and Standards (e.g.,ASME and ANSI).

- An investigation of the impact of power uprates for the Hatch plant shows that, in general, design margins are reduced for the uprates, but no design limits were exceeded.

- Licensee Safety Analysis Reports and NRC Staff Safety Evaluation Reports do not appear to be of sufficient detail or consistency to conduct an assessment of the margin impact for multiple licensing actions (e.g., power uprates/life extension/higher fuel burnups, etc.).

- The staff should consider: (1) development of a Standard Review Plan Section to address power uprate requests, (2) development of Legacy Tables to track the impact of successive licensing actions on such parameters as plant operations, structures, systems, and components (SSCs), and plant margins, and (3) performance of risk assessments for significant power uprate applications.

Committee Action

No Committee action on this matter was taken at this time. Further discussion of this issue will be held during the July 2001 meeting, following a June 12, 2001 Subcommittee meeting to continue discussion of issues pertaining to core power uprates.

3. Draft Final Safety Evaluation Report for the South Texas Project Nuclear Operating Company (STPNOC) Request to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations

The Committee heard presentations by and held discussions with representatives of the NRC staff on the preliminary safety evaluation for the STP request for exemptions from

certain special treatments requirements contained in 10 CFR Parts 21, 50, and 100 that impose controls to ensure the quality of SSCs that are within the scope of the regulations.

The NRC staff review assumes that the design basis would not change, that the functional capability of SSCs would be maintained for design basis conditions, and, that the FSAR (the licensing basis for exemptions) would include high level descriptions of programs on treatment of low safety significance SSCs.

In the preliminary safety evaluation, it is concluded that the categorization process is acceptable and that the alternative treatment program, if effectively implemented by the licensee, can result in SSCs remaining capable of performing their safety functions under design basis conditions. Thirteen exemptions are recommended to be granted and six exemptions are recommended not to be granted.

Committee Action

The ACRS plans to complete a report to the Commission on this matter during its July 13-14, 2001 Committee meeting.

4. Discussion of General Design Criteria

The Committee was briefed by the ACRS Senior Fellow J. N. Sorensen on his views regarding risk informing the General Design Criteria (GDC), Appendix A to 10 CFR Part 50. The GDC were incorporated into Part 50 in 1971, and reflect the state of the art in light water reactor safety design at that time. The safety standard addressed is reasonable assurance that a facility can be operated without undue risk to the health and safety of the public, rather than quantitative risk metrics derived from safety goals. There are three approaches that can be taken to making the GDC "risk-informed" as the term is currently used. The first is to revise the scope of the GDC to address structures, systems and components important to risk, using metrics such as core damage frequency and large early release frequency. The second is to examine individual criteria and modify each, as necessary, to address risk as the appropriate measure of safety. The third approach is to replace the GDC within the regulatory structure with a statement of regulatory objectives and risk acceptance criteria for each objective. Current NRC staff activities associated with risk-informing the special treatment requirements in 10 CFR Part 50 (Option 2) and risk-informing the technical requirements in 10 CFR Part 50 (Option 3) are examples of the first and second approaches, respectively. The Nuclear Energy Institute (NEI) is developing a proposed set of general design criteria that will be applicable to all reactor designs not just LWRs.

Committee Action

This was an information briefing only. Further discussions will be held during the December 2001 meeting, when additional information is available from both the NRC staff work on Options 2 and 3, and the NEI effort.

5. Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"

The Committee held a discussion concerning the need to revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," on the basis of experience gained by three license renewal applications and the generic guidance documents associated with the license renewal process.

Committee Action

The Committee decided to hear a briefing from the staff and prepare a report on this matter at the July 11-13, 2001 ACRS meeting.

6. Regulatory Challenges for Advanced Power Reactors

Dr. Thomas Kress, Chairman of the ACRS Subcommittee on Advanced Reactors, provided a report on the results of the June 4-5, 2001 Subcommittee meeting, concerning regulatory challenges for future nuclear power plants. He noted that Commissioner Nils Diaz provided an outstanding start to the meeting as the keynote speaker. Dr. Kress stated that the Subcommittee heard presentations by and held discussions with a broad range of personnel from government, industry, universities, and concerned citizen groups concerning these matters. He noted that the discussion covered a broad range of issues including: NRC and industry infrastructure needed to support a new generation of plants, defense-in-depth features including provisions for containment and emergency preparedness, risk assessment for new plant designs and human performance elements, and consideration of the Commission's Safety Goals for multiple-modular reactors.

Committee Action

The Committee plans to continue its review of this matter during future meetings. The Committee requests to be kept informed of staff schedules related to possible future nuclear power plants and requests the opportunity to review these matters during the

early stages of development.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee discussed the response from the NRC Executive Director for Operations (EDO) dated May 17, 2001, to the ACRS comments and recommendations included in the ACRS report dated April 13, 2001, concerning the proposed final license renewal guidance documents.

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

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from May 10, 2001, through June 6, 2001, the following Subcommittee meetings were held:

- Advanced Reactors Workshop on Regulatory Challenges for Future Nuclear Power Plants - June 4-5, 2001

  The Subcommittee discussed matters related to regulatory challenges for future nuclear power plants. The meeting was conducted as a workshop, with presentations, panel discussions, and participation by the workshop attendees.

- Planning and Procedures - June 6, 2001

  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 MATTERS FOR THE ATTENTION OF THE EXECUTIVE DIRECTOR FOR OPERATIONS

- The Committee plans to continue its review of matters related to possible future nuclear power plants. The Committee requests to be kept informed of staff schedules regarding industry initiatives in this area and requests the opportunity to review these matters during the early stages of development.

- The Commission plans to work with the staff in the development of risk-based performance indicators.

- As requested by the staff, the Committee plans to review the proposed final revision to Regulatory Guide 1.174 to address PRA quality in risk-informed activities after reconciliation of public comments.

PROPOSED SCHEDULE FOR THE 484th ACRS MEETING, JULY 11-13, 2001

The Committee agreed to consider the following topics during the 484th ACRS Meeting, July 11-13, 2001:

Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50
Briefing by and discussions with representatives of the NRC staff regarding proposed risk-informed revisions to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and proposed revisions to the framework for risk-informing the technical requirements of 10 CFR Part 50.

SECY-01-0100, "Policy Issues Related to Safeguards, Insurance,and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools"
Briefing by and discussions with representatives of the NRC staff regarding SECY-01-0100 and related matters.

Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"
Briefing by and discussion with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding the need to revise 10 CFR Part 54.

Control Rod Drive Mechanism (CRDM) Cracking
Briefing by and discussions with representatives of the NRC staff and NEI regarding the staff and industry proposals for dealing with CRDM cracking.

Draft Individual Plant Examination of External Events (IPEEE) Insights Report
Briefing by and discussions with representatives of the NRC staff regarding the draft IPEEE Insights Report (NUREG-1742).

Proposed Resolution of Generic Safety Issues (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Pump Performance"
Briefing by and discussions with representatives of the NRC staff regarding the status of resolution of GSI-191.

Potential Margin Reductions Associated with Power Uprates
Discussions with representatives of the NRC staff regarding ongoing or proposed staff activities related to the development of a Standard Review Plan for use in the review of power uprate applications.

Reactor Oversight Process
Discussion of proposed response to the following items in the April 5, 2000 Staff Requirements Memorandum: (1) Review the use of performance indicators (PIs) in the Revised Reactor Oversight Process (RROP) to ensure that the PIs provide meaningful insight into aspects of plant operation that are important to safety; (2) Review the initial implementation of the significance determination processes (SDPs) and assess the technical adequacy of the SDP to contribute to the RROP.

Sincerely,

George E. Apostolakis
Chairman

CERTIFIED

Date Issued: July 17, 2001
Date Certified: July 26, 2001

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JUNE 6-8, 2001

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MEMORANDA

— Proposed Final Regulatory Guide, 1.52, Revision 3, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated June 11, 2001)

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APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
IV. Future Agenda and Subcommittee Activities
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MINUTES OF THE 483rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS JUNE 6-8, 2001 ROCKVILLE, MARYLAND

The 483rd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on June 6-8, 2001. Notice of this meeting was published in the *Federal Register* on May 23, 2001 (65 FR 28567) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

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

ATTENDEES

ACRS Members: Dr. George Apostolakis (Chairman), Dr. Mario V. Bonaca (Vice Chairman), Dr. F. Peter Ford, Dr. Thomas S. Kress, Mr. Graham M. Leitch, Dr. Dana A. Powers, Dr. William J. Shack, 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. George E. Apostolakis, Committee Chairman, convened the meeting at 8:30 a.m. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee.

II. Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (Open)

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

Dr. William J. Shack, Chairman of the ACRS Subcommittee on Materials and Metallurgy introduced the topic to the Committee. He stated that the purpose of this meeting was to discuss the status the status of proposed risk-informed revisions to 10 CFR 50.46 for emergency core cooling systems (ECCS) and proposed revision to the framework for risk-informing the technical requirements of 10 CFR Part 50.

NRC Staff Presentation

Mr. Mark Cunningham, Office of Nuclear Regulatory Research (RES), led the presentation for the NRC staff. Mr. Alan Kuritsky provided supporting discussion. The staff provided an overview of existing 10 CFR 50.46 requirements and discussed options for developing risk-informed alternative requirements. Significant points made during the presentation include:

- Improved realism of large-break loss-of-coolant accident (LBLOCA) analysis, including possible LBLOCA redefinition, may involve evaluation of credible break sizes and uncertainty propagation through associated ECCS models. An option might also be to develop design- or plant-specific definitions of LBLOCA.

- Options for revising ECCS criteria should consider development of requirements such that ECCS reliability is commensurate with LOCA frequency. A performance-oriented option would replace single-failure criterion and LOCA/LOOP criterion with ECCS functional reliability. A prescriptive-oriented option would revise LOCA/LOOP to apply to the more likely pipe breaks and retain single failure criterion.

- Options for revising ECCS acceptance criteria should consider development of criteria such that ECCS performance is shown to maintain coolable core geometry for the duration of the accident. A performance-oriented option would show that fuel cladding integrity is maintained through tests. A prescriptive-oriented option would limit peak cladding temperature and maximum oxidation.

- Staff expectations for possible LBLOCA redefinition would include technical justification of LBLOCA frequency including service experience, estimates of pipe failure frequency, and analysis of other failures as surrogates. The staff would expect analysis to be at least as rigorous as that required for pressurized thermal shock.

- Policy issues for the Phase II technical work include single-failure criterion and selective implementation.

- The staff informed the Committee that this was a status briefing and that they were not requesting an ACRS report or letter at this time. The staff plans to provide its draft Commission paper for consideration by the Committee in late June 2001.

Industry Presentation

Messrs. Anthony Pietrangelo and Adrian Heymer of the Nuclear Energy Institute (NEI) led the industry presentation. Mr. Robert Osterider of the Westinghouse Owners Group provided supporting discussion. Significant points made during the presentation include:

- Industry proposes a "simple rule change" to allow the NRC to consider alternative break sizes based on classes of NSSS designs and plant-specific features. Alternative break sizes would be approved by the Commission or Director of the Office of Nuclear Reactor Regulation (NRR).

- NEI representatives expressed concern over the schedule for completing the proposed rulemaking for 10 CFR 50.46. NEI stated that the 10 CFR 50.46 effort should build on past experience, e.g., leak-before-break (LBB) criteria (GDC-4) and probabilistic fracture analysis (PFM), and expressed concern that progress on risk-informing 10 CFR 50.46 might slow in seeking perfection. NEI proposed that to separate criteria for decay heat peak clad temperature be partitioned into a separate rulemaking.

Dr. Powers questioned why 10 CFR 50.46 was amenable to risk-information. The staff stated that the January 2000 NEI letter had highlighted 10 CFR 50.46 as a priority candidate for risk-informing. The staff noted that there was a significant amount of service information available on large diameter pipe performance and that there was also a fair amount of available risk information. Dr. Shack suggested that other safety enhancements, e.g., increased emergency diesel generator reliability, could also be achieved.

Drs. Kress and Powers questioned what actual changes would be made to the regulations if it was decided that risk-informing the regulations in 10 CFR 50.46 is feasible. The staff stated that they did not bring a list of changes to discuss but noted that a lot of smaller regulatory changes would also be needed if 10 CFR 50.46 is revised.

Dr. Powers noted that the oxidation criteria in the 10 CFR 50.46 is related mostly to embrittlement and is not, in fact, a hydrogen generation criterion. He questioned whether the staff had considered revising the rule to include an actual embrittlement criterion. The staff agreed and discussed detailed plans to pursue such changes.

Dr. Apostolakis questioned which accident sequences were affected by risk-informing 10 CFR 50.46. Dr. Kress stated that he would like to see an evaluation of the sequences beyond that needed to calculate core damage frequency (CDF) and large, early release frequency (LERF) as the basis for selecting the categorization of equipment. Dr. Kress further suggested that this should apply to both the current generation of nuclear plants and future reactors and stated that embarking on a change based on risk without some understanding of the uncertainties and rigor of the analysis is a mistake . Drs. Powers and Kress expressed the view that defense in depth should be appropriately considered, particularly with regard to the containment integrity. The staff stated that they plan to use risk tools to the extent practicable and suggested that the quantitative risk information should help in making better qualitative judgments. The staff further stated that containment integrity would be appropriately considered with other important defense-in-depth issues. Dr. Wallis expressed the view that changing the rule and associated regulations should have a safety benefit and not serve merely for the relaxation of requirements.

Committee Action

A joint meeting of the ACRS Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and PRA is scheduled for July 9, 2001, on this matter. The Committee plans to continue its review of this matter during the July 11-13, 2001 ACRS meeting, subject to the availability of the staff's proposed Commission paper.

III. Potential Margin Reductions Associated with Power Uprates (Open)

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

The Committee was briefed by the ACRS Senior Fellow Dr. A. W. Cronenberg on his views with regard to the adequacy of the staff's review process for power uprates and potential safety margin reductions associated with power uprates. Key points noted by Dr. Cronenberg were:

- The NRC's General Design Criteria do not explicitly address how much design margin is required. Typically, the words "sufficient margin" are used. Margin requirements are more explicit in the Standard Review Plan and in such documents as industry Codes and Standards (e.g.,ASME and ANSI).

- An investigation of the impact of power uprates for the Hatch plant shows that, in general, design margins are reduced for the uprates, but no design limits were exceeded.

- Licensee Safety Analysis Reports and NRC Staff Safety Evaluation Reports do not appear to be of sufficient detail or consistency to conduct an assessment of the margin impact for multiple licensing actions (e.g., power uprates/life extension/higher fuel burnups, etc.).

- The staff should consider: (1) development of a Standard Review Plan Section to address power uprate requests, (2) development of Legacy Tables to track the impact of successive licensing actions on such parameters as plant operations, structures, systems and components (SSCs), and plant margins, and (3) performance of risk assessments for significant power uprate applications.

Committee Action

No Committee action on this matter was taken at this time. Further discussion of this issue will be held during the July 2001 meeting, following a June 12, 2001 Subcommittee meeting to continue discussion of issues pertaining to core power uprates.

IV. Draft Final Safety Evaluation Report for the South Texas Project Nuclear Operating Company (STPNOC) Request to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations (Open)

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

Mr. John D. Sieber, Chairman of the ACRS Subcommittee on Plant Operations, introduced this topic to the committee. He stated that this was last in a series of meetings on the South Texas Project (STP) Exemption Request. This presentation focused on the preliminary safety evaluation on exemptions requested from special treatment requirements. Previous meetings were held on the draft safety evaluation, categorization, and treatment.

NRC Staff Presentation

The staff presentation on the South Texas Project exemption request was made by Mr. John Nakoski, NRR, along with Mr. Jack Strosnider, NRR, and Mr. Gary Holahan, NRR.

NRC regulations in 10 CFR Parts 21, 50, and 100 contain special treatment requirements that impose controls to ensure the quality of SSCs that are within the scope of the regulations.

A risk informed request for exemptions from the special treatment requirements of 10 CFR Parts 21, 50, and 100 was submitted by STPNOC on July 13, 1999, and supplemented on October 14 and 22, 1999; January 26 and August 31, 2000; January 15, 18, and 23, March 19, and May 8 and 21, 2001. The submittal sought approval of processes for the categorization and treatment of SSCs consistent with their categorization as the principal basis for granting the exemptions.

The NRC review of STPNOC's treatment processes assumes that the design basis would not change, that the functional capability of SSCs would be maintained for design basis conditions, and, that the FSAR (the licensing basis for exemptions) would include high level descriptions of the programs on treatment of LSS SSCs.

In the preliminary safety evaluation, the NRC staff concluded that the categorization process is acceptable to reduce the scope of SSCs subject to special treatment and for defining SSCs for which exemptions can be granted. The staff also concluded that the alternative treatment program includes the necessary elements that <u>if effectively implemented by the licensee,</u> can result in SSCs remaining capable of performing their safety functions under design basis conditions. Thirteen exemptions are recommended to be granted and six exemptions are recommended not to be granted.

<u>Committee Action</u>

The ACRS will complete a report to the Commission at the July Committee meeting.

V. <u>Discussion of General Design Criteria</u> (Open)

[Note: Mr. Howard J. Larson was the Designated Federal Official for this portion of the meeting.]

The Committee was briefed by the ACRS Senior Fellow J. N. Sorensen on his views regarding risk informing the General Design Criteria (GDC), Appendix A to 10 CFR Part 50. The GDC were incorporated into Part 50 in 1971, and reflect the state of the art in light water reactor safety design at that time. The safety standard addressed is reasonable assurance that a facility can be operated without undue risk to the health and safety of the public, rather than quantitative risk metrics derived from safety goals. There are three approaches that can be taken to making the GDC "risk-informed" as the term is currently used. The first is to revise the scope of the GDC to address SSCs important to risk, using metrics such as core damage frequency and large early release frequency. The second is to examine individual criteria and modify each, as necessary,

to address risk as the appropriate measure of safety. The third approach is to replace the GDC within the regulatory structure with a statement of regulatory objectives and risk acceptance criteria for each objective. Current NRC staff activities associated with risk-informing the special treatment requirements in 10 CFR Part 50 (Option 2) and risk-informing the technical requirements in 10 CFR Part 50 (Option 3) are examples of the first and second approaches, respectively. NEI is developing a proposed set of general design criteria that will be applicable to all reactor designs not just LWRs.

Committee Action

This was an information briefing only. Further discussions will be held during the December 2001 meeting, when additional information is available from both the NRC staff work on Options 2 and 3, and the NEI effort.

VI. Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Open)

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

Dr. Mario Bonaca, Chairman of the Plant License Renewal Subcommittee, stated that the Committee had completed its review of the Calvert Cliffs, Oconee, and Arkansas Nuclear One license renewal applications. He noted that the Commission in the Staff Requirements Memorandum dated August 28, 1999, requested the staff to prepare a detailed analysis and provide recommendations on whether it would be appropriate to resolve generic technical issues when more data and experience had been accumulated from the license renewal applications.

The ACRS members discussed the lessons learned from its review of the license renewal applications and the generic documents associated with the license renewal process. They discussed the importance of the results of the scoping process and the Committee's previous recommendation that this information be included in future license renewal applications.

Committee Action

The Committee decided to hear a briefing from the staff and prepare a report on this matter at the July 11-13, 2001 ACRS meeting.

VII. Regulatory Challenges for Advanced Power Reactors (Open)

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

Dr. Thomas Kress, Chairman of the ACRS Subcommittee on Advanced Reactors, provided a report on the results of the June 4-5, 2001 Subcommittee meeting, concerning regulatory challenges for future nuclear power plants. He noted that Commissioner Nils Diaz provided an outstanding start to the meeting as the keynote speaker. Dr. Kress stated that the Subcommittee heard presentations by and held discussions with a broad range of personnel from government, industry, universities, and concerned citizen groups concerning these matters. He noted that the discussion covered a broad range of issues including: NRC and industry infrastructure needed to support a new generation of plants, defense-in-depth features including provisions for containment and emergency preparedness, risk assessment for new plant designs and human performance elements, and consideration of the Commission's Safety Goals for multiple-modular reactors.

Committee Action

The Committee plans to continue its review of this matter during future meetings. The Committee requests to be kept informed of staff schedules related to possible future nuclear power plants and requests the opportunity to review these matters during the early stages of development.

VIII. 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 May 17, 2001, to the ACRS comments and recommendations included in the ACRS report dated April 13, 2001, concerning the proposed final license renewal guidance documents.

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

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

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

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

— Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through September 2001 was discussed. The objectives were:

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

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

— Quadripartite Meeting Update

During the April meeting, the Committee was informed that Lothar Hahn, Chairman of the RSK, was preparing for Germany to host the next Quadripartite meeting, possibly later this year. The French GPR confirmed their participation and the RSK is currently working to confirm the participation of the Japanese NSC.

Subsequent to the April meeting, Mrs. Waldorf from GRS in Germany said that the RSK planned to meet on May 3, 2001 to discuss the next Quadripartite meeting. Mr. Lathar Hahn attended this meeting. According to Mrs. Waldorf, consideration is being given to hold the next Quadripartite meeting in November 2001.

During the May 2001 meeting, the Committee proposed November 13-15, 2001 as possible dates for this meeting and suggested that Dr. Larkins get feedback from RSK on these dates. Also, the Committee suggested that the members propose topics for this meeting.

Dr. Larkins spoke with Mr. Kersting, Vice-Chairman of the RSK, about a meeting in November and a format for the meeting. During that discussion, Mr. Kersting noted that the RSK would probably propose that the meeting be held in May of

2002 because there was not enough time to plan and orchestrate a meeting in November 2001.

Dr. Larkins volunteered to help the RSK develop the format for the next Quadripartite meeting. An issue which needs to be resolved is the participation of the Japanese Advisory Committee, the NSC, in this next Quadripartite meeting.

During the May meeting, the Committee suggested the following topics for this meeting.

- Risk-informed regulation
- Thermal hydraulic analysis and code issues
- Revised reactor oversight process
- High burnup fuel
- Fire protection and insights from the IPEEE process
- Risk analysis of spent fuel storage
- Quality of data and the predictive capabilities for environmentally assisted cracking
- Defense-in-depth needs and the safety criteria for future reactors
- Human performance and safety culture
- Level of inherent safety needed to eliminate the need for emergency response and evacuation

— Annual Report to the Commission on the NRC Safety Research Program

Advanced copies of the ACRS report for 2001 was provided to the Commissioners and is expected to be published as NUREG-1635, Vol. 4 in the near future. The 2001 ACRS report is currently available on the ACRS web site.

The 2002 report is due to the Commission in March 2002. During previous meetings, the Committee discussed briefly whether to write a comprehensive report every year, but did not make a decision. In view of the fact that the NRC research programs may not change significantly within a year, a comprehensive report may not be needed each year.

— RSK Workshop on Risk Informed Decisionmaking in Nuclear Safety

During the May 2001 ACRS meeting, the Committee was informed about the RSK Workshop on Risk-Informed Decisionmaking in Nuclear Safety that is scheduled to be held on June 11-12, 2001, in Germany. Lothar Hahn stated that

since several anticipated participants were having difficulty attending the Workshop on June 11-12, it would be postponed and a future date to be discussed.

— Items of Interest

- The ACRS/ACNW self-assessment paper (SECY-01-0092) was completed on May 21, 2001, satisfying a Commission milestone. We are currently in the process of revising the Operating Plan, which will be issued by the end of June 2001.

- The ACRS Action Plan has been placed on the web site and is expected to be published as a brochure within the next week or two.

- In response to an inquiry from former ACRS Member Hal Lewis regarding an ACRS reunion for the Committee's 50th Anniversary, Dr. Larkins forwarded an e-mail to Chairman Meserve that the Committee has had some preliminary discussions, but has not planned anything specific at this time.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 484th ACRS Meeting, July 11-13, 2001.

The 483rd ACRS meeting was adjourned at 12:00 p.m. on June 8, 2001.

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

July 26, 2001

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

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

SUBJECT: CERTIFIED MINUTES OF THE 483rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), JUNE 6-8, 2001

I certify that based on my review of the minutes from the 483rd ACRS full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

George E. Apostolakis, Chairman

July 26, 2001
Date

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

July 17, 2001

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 483rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - June 6-8, 2001

Enclosed are the proposed minutes of the 483rd meeting of the ACRS. This draft is being provided to give you an opportunity to review the record of this meeting and provide comments. Your comments will be incorporated into the final certified set of minutes as appropriate.

Attachment:
As stated

Safety and Health, directed the preparation of this notice. The authority for this notice is the Paperwork Reduction Act of 1995 (44 U.S.C. 3506) and the Secretary of Labor's Order No. 3–2000 (65 FR 50017).

Dated: Signed at Washington, DC on May 18, 2001.

R. Davis Layne,

Acting Assistant Secretary of Labor.

[FR Doc. 01–13043 Filed 5–22–01; 8:45 am]

BILLING CODE 4510–26–M

NATIONAL FOUNDATION ON THE ARTS AND THE HUMANITIES

National Endowment for the Arts Combined Arts Advisory Panel

Pursuant to section 10(a)(2) of the Federal Advisory Committee Act (Public Law 92–463), as amended, notice is hereby given that two meetings of the Leadership Initiatives Advisory Panel to the National Council on the Arts will be held at the Nancy Hanks Center, 1100 Pennsylvania Avenue, NW., Washington, DC 2506 as follows:

Visual Arts (Creativity and Organizational Capacity categories): June 19–21, 2001, Room 716. A portion of this meeting, from 1:30 a.m. to 2:30 p.m. on June 21st, will be open to the public for policy discussion. The remaining portions of this meeting, from 9 a.m. to 5:30 p.m. on June 19th and 20th, and from 9 a.m. to 11 a.m. and 12 p.m. to 5 p.m. on June 21st, will be closed.

Opera (Creativity and Organizational Capacity categories): June 26–27, 2001, Room 714. A portion of this meeting, from 1 p.m. to 2 p.m. on June 27th, will be open to the public for policy discussion. The remaining portions of this meeting, from 10 a.m. to 6:15 p.m. on June 26th, and from 9 a.m. to 1 p.m. and 2 p.m. to 4:30 p.m. on June 27th, will be closed.

The closed portions of these meetings are for the purposes of Panel review, discussion, evaluation, and recommendation on applications for financial assistance under the National Foundation on the Arts and the Humanities Act of 1965, as amended, including information given in confidence to the agency by grant applicants. In accordance with the determination of the Chairman of May 12, 2000, these sessions will be closed to the public pursuant to (c)(4)(6) and (9)(B) of section 552b of Title 5, United States Code.

Any person may observe meetings, or portions thereof, of advisory panels that are open to the public, and, if time allows, may be permitted to participate in the panel's discussions at the discretion of the panel chairman and with the approval of the full-time Federal employee in attendance.

If you need special accommodations due to a disability, please contact the Office of AccessAbility, National Endowment for the Arts, 1100 Pennsylvania Avenue, NW., Washington, DC 20506, 202/682–5532, TDDY–TDD 202/682–5496, at least seven (7) days prior to the meeting.

Further information with reference to this meeting can be obtained from Ms. Kathy Plowitz-Worden, Office of Guidelines & Panel Operations, National Endowment for the Arts, Washington, DC 20506, or call 202/682–5691.

Dated: May 17, 2001.

Kathy Plowitz-Worden,

Panel Coordinator, Panel Operations, National Endowment for the Arts.

[FR Doc. 01–12911 Filed 5–22–01; 8:45 am]

BILLING CODE 7537–01–M

NATIONAL INSTITUTE FOR LITERACY

Notice of Meeting

AGENCY: National Institute for Literacy (NIFL).

SUMMARY: This notice sets forth the schedule and proposed agenda of a forthcoming meeting of the National Institute for Literacy Board (Advisory Board). This notice also describes the function of the Advisory Board. Notice of this meeting is required under Section 10(a)(2) of the Federal Advisory Committee Act. This document is intended to notify the general public of their opportunity to attend the meeting.

DATE AND TIME: June 7, 2001 from 10:00 am to 5:00 pm.

ADDRESSES: National Institute for Literacy, 1775 I Street, NW., Suite 730, Washington, DC 20006.

FOR FURTHER INFORMATION CONTACT: Shelly Coles, Executive Assistant, National Institute for Literacy, 1775 I Street, NW., Suite 730, Washington, DC 20006. Telephone number (202) 233–2027, email scoles@nifl.gov.

SUPPLEMENTARY INFORMATION: The Advisory Board is established under the Workforce Investment Act of 1998, Title II of Public Law 105–220, Sec. 242, the National Institute for Literacy. The Advisory Board consists of ten individuals appointed by the President with the advice and consent of the Senate. The Advisory Board is established to advise and make recommendations to the Interagency Group, composed of the Secretaries of Education, Labor, and Health and Human Services, which administers the National Institute for Literacy (Institute). The Interagency Group considers the Advisory Board's recommendations in planning the goals of the Institute and in the implementation of any programs to achieve the goals of the Institute. Specifically, the Advisory Board performs the following function (a) Makes recommendations concerning the appointment of the Director and the staff of the Institute; (b) provides independent advice on operation of the Institute; and (c) receives reports from the Interagency Group and Director of the Institute. In addition, the Institute consults with the Advisory Board on the award of fellowships. The National Institute for Literacy Advisory Board meeting on June 7, 2001, will focus on future and current NIFL programs activities, and other relevant literacy activities and issues. Records are kept of all Advisory Board proceedings and are available for public inspection at the National Institute for Literacy, 1775 I Street, NW., Suite 730, Washington, DC 20006, from 8:30 am to 5 pm.

Dated: May 14, 2001.

Carolyn Y. Staley,

Acting Executive Director.

[FR Doc. 01–12994 Filed 5–22–01; 8:45 am]

BILLING CODE 6055–01–M

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 June 6–8, 2001, in Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Friday, November 17, 2000 (65 FR 69578).

Wednesday, June 6, 2001

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.–11:20 a.m.: Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding proposed risk-informed revisions to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and proposed revisions to the framework for risk-informing the technical requirements of 10 CFR Part 50.

11:20 a.m.–12:15 p.m.: Potential Margin Reductions Associated with Power Uprates (Open)—The Committee will hear a presentation by and hold discussions with ACRS Senior Fellow, Dr. A. W. Cronenberg, regarding his

views on the adequacy of the staff's review process for power uprates and potential safety margin reductions associated with power uprates.

1:15 p.m.–3:15 p.m.: Draft Final Safety Evaluation Report for the South Texas Project Nuclear Operating Company (STPNOC) Request to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final Safety Evaluation Report on the STPNOC exemption request.

3:35 p.m.–4:30 p.m.: Discussion of General Design Criteria (Open)—The Committee will hear a presentation by and hold discussions with Mr. J. N. Sorensen, ACRS Senior Fellow, regarding his views on risk-informing General Design Criteria included in Appendix A to 10 CFR Part 50.

4:50 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, as well as proposed ACRS reports on: Risk-Based Performance Indicators; Response to the EDO's letter of April 12, 2001 on Topics Raised by the ACRS Pertaining to Issues Associated with Industry Use of Thermal-Hydraulic Codes; and Response to NRC Chairman Meserve's May 7, 2001 Memorandum Regarding Differing Professional Opinion on Steam Generator Tube Issues.

Thursday, June 7, 2001

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

8:35 a.m.–9:45 a.m.: Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Open)—The Committee will discuss the need for revising 10 CFR Part 54.

10:00 a.m.–11:00 a.m.: Regulatory Challenges for Advanced Power Reactors (Open)—The Committee will discuss follow-up matters resulting from the June 4–5, 2001 Workshop of the ACRS Subcommittee on Advanced Reactors and develop a course of action for dealing with these matters in the future.

11:00 a.m.–11:45 a.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

11:45 a.m.–12:00 Noon: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

1:00 p.m.–1:30 p.m.: Arrangements for the Visit to the Waterford Plant and Region IV Office (Open)—The Committee will discuss arrangements and logistics for the June 27–28, 2001, visit to the Waterford Plant and NRC Region IV Office.

1:30 p.m.–7:00 p.m.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Friday, June 8, 2001

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open).

8:35 a.m.–9:00 a.m.: Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

9:00 a.m.–10:00 a.m.: Meeting with Commissioner Dicus (Open)—The Committee will meet with NRC Commissioner Dicus to discuss items of mutual interest.

10:15 a.m.–2:30 p.m.: Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

2:30 p.m.–3:00 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 11, 2000 (65 FR 60476). In accordance with these procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify Mr. James E. Lyons, ACRS, five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting Mr. James E. Lyons prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with Mr. James E. Lyons if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting Mr. James E. Lyons (telephone 301–415–7371), between 7:30 a.m. and 4:15 p.m., EDT.

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

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

Dated: May 17, 2001.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 01–13016 Filed 5–22–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

The ACRS Subcommittee on Planning and Procedures will hold a meeting on June 6, 2001, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

APPENDIX II

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

May 16, 2001

SCHEDULE AND OUTLINE FOR DISCUSSION
483rd ACRS MEETING
JUNE 6-8, 2001

WEDNESDAY, JUNE 6, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - ~~11:20~~ 10:20 A.M. Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (Open) (WJS/GBW/MTM/RBE)

(~~10:00-10:20~~ A.M. BREAK) 10:20-10:40

- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff regarding proposed risk-informed revisions to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and proposed revisions to the framework for risk-informing the technical requirements of 10 CFR Part 50.

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

10:40-11:20 Discuss Thermal Hydraulic Codes letter

3) 11:20 - 12:15 P.M. Potential Margin Reductions Associated with Power Uprates (Open) (GBW/MVB/AWC/PAB)
- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with the ACRS Senior Fellow, Dr. A. W. Cronenberg, regarding his views on the adequacy of the staff's review process for power uprates and potential safety margin reductions associated with power uprates.

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

4) ~~1:15~~ 1:20 - ~~3:15~~ 2:55 P.M. Draft Final Safety Evaluation Report for the South Texas Project Nuclear Operating Company (STPNOC) Request to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations (Open) (JDS/GEA/MWW)
- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final Safety Evaluation Report on the STPNOC exemption request.

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

2:55- ~~3:15~~ - **3:35 P.M.** *****BREAK*****

5) 3:35 - 4:55 ~~4:30~~ P.M. Discussion of General Design Criteria (Open) (GEA/JNS)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with Mr. Sorensen, ACRS Senior Fellow, regarding his views on risk-informing General Design Criteria included in Appendix A to 10 CFR Part 50.

4:55-5:10 ~~4:30~~ - **4:50 P.M.** *****BREAK*****

6) 4:50 - 7:00 P.M. Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6:00-7:00 6.1) Risk-Based Performance Indicators (GEA/MTM)

6.2) Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (WJS/GBW/MTM/RBE)

5:10-6:00 6.3) Response to Chairman Meserve's May 7, 2001, memorandum, "Differing Professional Opinion On Steam Generator Tube Issues" (DAP/SD)

6.4) Response to the Executive Director for Operations' Letter of April 12, 2001 on Topics Raised by ACRS Pertaining to Issues Associated with Industry Use of Thermal-Hydraulic Codes (GBW/PAB)

6.5) South Texas Project Exemption Request (JDS/GEA/MWW)

6.6) Potential Margin Reductions Associated with Power Uprates (Tentative) (GBW/PAB/AWC)

THURSDAY, JUNE 7, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GEA/JTL/SD)

8) 8:35 - 9:35 ~~9:45~~ A.M. Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Open) (MVB/SD/RBE)

8.1) Remarks by the Subcommittee Chairman

8.2) Discussion by the ACRS members on the need for revising 10 CFR Part 54.

9:35- ~~9:45~~ - **10:00 A.M.** *****BREAK*****

9) 10:00 - 10:30 ~~11:00~~ A.M. Regulatory Challenges for Advanced Power Reactors (Open) (TSK/MTM)

9.1) Remarks by the Subcommittee Chairman

9.2) Discussion of follow-up matters resulting from the June 4-5, 2001 Workshop of the ACRS Subcommittee on Advanced Reactors and development of a course of action for dealing with these matters in the future.

10) 10:30 ~~11:00~~ - 11:45 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL)

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

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

11) 11:45 - 11:50 ~~12:00~~ Noon Reconciliation of ACRS Comments and Recommendations (Open) (GEA, et al./SD, et al.)

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

11:50 - 1:30 ~~12:00~~ - ~~1:00~~ **P.M.** *****LUNCH*****

12) 1:30 - 1:45 ~~1:00~~ - ~~1:30~~ P.M. Arrangements for the Visit to the Waterford Plant and Region IV Office (Open) (JDS/AS/MWW)

Discussion of arrangements and logistics for the June 27-28, 2001, visit to the Waterford Plant and NRC Region IV Office.

13) 1:45 - 3:05 ~~1:30~~ - ~~2:30~~ P.M. Break and Preparation of Draft ACRS Reports (Open)

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

14) 3:05 - ~~2:30~~ - 7:00 P.M. Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

14.1) Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (WJS/GBW/MTM/RBE)

14.2) Response to Chairman Meserve's May 7, 2001, memorandum, "Differing Professional Opinion On Steam Generator Tube Issues" (DAP/SD)

14.3) Response to the April 12, 2001 EDO Letter on Topics Raised by the ACRS Pertaining to Issues Associated with Industry use of Thermal-Hydraulic Codes (GBW/PAB)

14.4) Risk-Based Performance Indicators (GEA/MTM)

14.5) South Texas Project Exemption Request (JDS/GEA/MWW)

14.6) Potential Margin Reductions Associated with Power Uprates (Tentative) (GBW/PAB/AWC)

FRIDAY, JUNE 8, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

15) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)
Larkinsgram - Proposed Rev 1 to Risk-Informed RG 1.174 & SRP Chapter 19; RG DG-1102

16) 8:35 - 9:00 A.M. Proposed ACRS Reports (Open) (MVB/JTL)
Discussion of proposed ACRS reports.

10:15-11:00 Risk-Based Performance Indicators - Final

17) 9:00 - 10:00 A.M. Meeting with NRC Commissioner Dicus (Open) (MVB/JTL)
The Committee will meet with Commissioner Dicus to discuss items of mutual interest.

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

18) 10:15 - ~~2:30~~ 12:00 P.M. Proposed ACRS Reports (Open)
~~(12:00-1:00 P.M. LUNCH)~~ Continue discussion of proposed ACRS reports listed under Item 14.

~~19) 2:30 - 3:00 P.M.~~ Miscellaneous (Open) (MVB/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.~~

11:10-12:00 STP Request for Exemption (Option 2) discussion

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

483rd ACRS MEETING
JUNE 6-8, 2001

NRC STAFF (June 6, 2001)
M. Mitchell, NRR
S. Bahadur, RES
A. Kuritzky, RES
D. Overland, RES
J. Lazernick, NRR
F. Eltawila, RES
S. Magruder, NRR
A. Ramey-Smith, RES
S. Lee, NRR
D. Harrison, NRR
B. Palla, NRR
P. Shemanski, NRR
B. Gramm, NRR
T. Scarbrough, NRR
A. Park, NRR
J. Fair, NRR
J. Nakoski, NRR
T. Reed, NRR
K. Hack, NRR
H. Garg, NRR
D. Fischer, NRR
S. Dinsmore, NRR
B. Bateman, NRR
C. Carpenter, NRR
G. Bagchi, NRR
J. Calvo, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
B. Osterrieder, Westinghouse
N. Chapman, Bechtel
R. Huston, Licensing Support Services
A. Camp, Sandia
P. Negus, GE
M. Knapik, McGraw-Hill
A. Heymer, NEI
G. Saouto, MIT
J. Newman, NMC
S. Head, STPNOC
R. Granton, STPNOC
G. Schinzel, STPNOC
S. Frantz, ML&B
D. Raleigh, Scientech

NRC STAFF (June 7, 2001)
S. Bahadur, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
R. Wells, Constellation Nuclear

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

June 14, 2001

SCHEDULE AND OUTLINE FOR DISCUSSION
484th ACRS MEETING
July 11-13, 2001

WEDNESDAY, JULY 11, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:00 A.M. Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (Open) (WJS/GBW/MTM)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding proposed risk-informed revisions to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and proposed revisions to the framework for risk-informing the technical requirements of 10 CFR Part 50.

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

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

3) 10:20 - 12:00 Noon SECY-01-0100, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools" (Open/Closed) (TSK/MME)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding SECY-01-0100 and related matters.

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

   [NOTE: A portion of this session may be closed to discuss safeguards information.]

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

4) 1:00 - 2:00 P.M. Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Open) (MVB/SD)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussion with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding the need to revise 10 CFR Part 54.

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

5) 2:15 - 3:45 P.M. Control Rod Drive Mechanism (CRDM) Cracking (Open/Closed) (FPF/JDS/MWW/PAB)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff and NEI regarding the staff and industry proposals for dealing with CRDM cracking.

[NOTE: A portion of this session may be closed to discuss proprietary information.]

6) 3:45 - 4:15 P.M. Break and Preparation of Draft ACRS Reports (Open)
Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

7) 4:15 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:

7.1) Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (WJS/GBW/MTM)

7.2) SECY-01-0100 and Related Matters (TSK/MME)

7.3) South Texas Project Exemption Request (JDS/GEA/MWW)

7.4) Need to Revise 10 CFR Part 54 (MVB/SD)

7.5) Proposals for Dealing with CRDM Cracking (FPF/JDS/MWW/PAB)

THURSDAY, JULY 12, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 - 9:45 A.M. Draft Individual Plant Examination of External Events (IPEEE) Insights Report (Open) (GEA/AS)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding the draft IPEEE Insights Report (NUREG-1742).

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

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

10) 10:00 - 11:00 A.M. Proposed Resolution of Generic Safety Issues (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Pump Performance" (Open) (GML/AS)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff regarding the status of resolution of GSI-191.

11:00 - 11:15 A.M. *****BREAK*****

11) 11:15 - 12:15 P.M. Potential Margin Reductions Associated with Power Uprates (Open) (GBW/MVB/AWC/PAB)

11.1) Remarks by the Subcommittee Chairman

11.2) Discussions with representatives of the NRC staff regarding ongoing or proposed staff activities related to the development of a Standard Review Plan for use in the review of power uprate applications.

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

12) 1:15 - 2:00 P.M. Reactor Oversight Process (Open) (JDS/GEA/MWW)

12.1) Remarks by the Subcommittee Chairman

12.2) Discussion of proposed response to the following items in the April 5, 2000 Staff Requirements Memorandum:

- Review the use of performance indicators (PIs) in the Revised Reactor Oversight Process (RROP) to ensure that the PIs provide meaningful insight into aspects of plant operation that are important to safety.
- Review the initial implementation of the significance determination processes (SDPs) and assess the technical adequacy of the SDP to contribute to the RROP.

13) 2:00 - 2:45 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL/SD)

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

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

14) 2:45 - 3:00 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GEA, et al./SD, et al.)

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

3:00 - 3:45 P.M. *****BREAK*****

15) 3:45 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:
15.1) Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (WJS/GBW/MTM)
15.2) SECY-01-0100 and Related Matters (TSK/MME)
15.3) South Texas Project Exemption Request (JDS/GEA/MWW)
15.4) Need to Revise 10 CFR Part 54 (MVB/SD)
15.5) Draft IPEEE Insights Report (GEA/AS)
15.6) Proposals for Dealing with the CRDM Cracking (FPF/JDS/MWW/PAB)
15.7) Potential Margin Reductions Associated with Power Uprates (tentative) (GBW/MVB/PAB/AWC)

FRIDAY, JULY 13, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

16) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GEA/JTL/SD)

17) 8:35 - 5:30 P.M. Proposed ACRS Reports (Open)
(12:00-1:00 P.M. LUNCH) Continue discussion of proposed ACRS reports listed under Item 15.

18) 5:30 - 6:00 P.M. Miscellaneous (Open) (GEA/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

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

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
483RD ACRS MEETING
JUNE 6-8, 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, 483rd ACRS Meeting dated June 6-8, 2001
2	Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50
	2. Risk-Informing 10 CFR 50.46 presentation by M. Cunningham and A. Kuritzky, RES/DRAA/PRAB (Viewgraphs)
3	Potential Margin Reductions Associated with Power Uprates
	3. Signature Estimates of Margin Reductions for Power Uprates/License Renewal presentation by A. Cronenberg, ACRS Fellow [Viewgraphs]
4	Draft Final Safety Evaluation Report for the South Texas Project Nuclear Operating Company (STPNOC) Request to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations
	4. South Texas Project Requested Exemptions from Special Treatment Requirements Staff Findings presentation by G. Holahan, J. Strosnider, J. Nakoski, NRR [Viewgraphs]
5	Discussion of General Design Criteria
	5. General Design Criteria presentation by J. Sorensen [Viewgraphs and attachments]
8	Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"
	6. Recommendation to the Commission on Whether or not there is a need to Revise 10 CFR Part 54, the License Renewal Rule [provided for internal ACRS use only]
9	Regulatory Challenges for Advanced Power Reactors

10 Report of the Planning and Procedures Subcommittee/Future ACRS Activities
7. Future ACRS Activities [Handout #**11**]

11 Reconciliation of ACRS Comments and Recommendations
8. Reconciliation of ACRS Comments and Recommendations [Handout #**11.1**]

17 Meeting with NRC Commissioner Dicus
9. List of discussion topics

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50
	1. Table of Contents
	2. Proposed Agenda
	3. Status Report, dated June 6, 2000
	2. Email from W. J. Shack to D. A. Powers, "RE: 10 CFR 50.46 Materials, PRA Quality & Meeting Schedule" dated May 28, 2001 (**predecisional for Internal ACRS use only**)
	3. Email from W. J. Shack to D. A. Powers, "RE: 10 CFR 50.46 Materials, PRA Quality & Meeting Schedule" dated May 29, 2001 (**predecisional for Internal ACRS use only**)
	4. Email from W. J. Shack to D. A. Powers, "RE: 10 CFR 50.46 Materials, PRA Quality & Meeting Schedule" dated May 30, 2001 (**predecisional for Internal ACRS use only**)
	5. 10 CFR 50.46 Acceptance criteria for emergency core cooling systems for light-water power reactors
	6. Meeting summary dated March 1, 2001, for the February 23, 2001 internal NRC meeting concerning risk-information 10 CFR 50.46 (**predecisional for internal ACRS use only**)
	7. Letter from A. Heymer to M. Drouin, "Large Break LOCA Redefinition Program, Project Summary," dated January 8, 2001
	8. Letter from R. H. Bryam to T. L. King, "WOG Large Break Loss of Coolant Accident (LBLOCA) Redefinition Discussion of Benefits, dated October 17, 2000
	9. Letter from J. F. Colvin to R. A. Meserve, "SECY-99-264, Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50," dated January 19, 2000
3	Potential Margin Reductions Associated with Power Uprates
	10. ACRS Fellow: A. Cronenberg
4	South Texas Project Exemption Requests from Special Treatments
	11. Table of Contents
	12. Proposed Schedule
	13. Status Report dated June 6, 2000
	14. Memorandum from S. A. Richards to Dr. J. T. Larkins, Subject: Preliminary Safety Evaluation and Draft Notices of Exemptions on STP Nuclear Operating Company (STPNOC) Exemptions from Special Treatment Requirements of 10 CFR Parts 21, 50, and 100

- Attachment 1-Draft Notices of Exemptions
- Attachment 2-Preliminary Safety Evaluation

8 Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"

15. Table of Contents
16. Proposed Agenda
17. Status Report dated June 7, 2001
18. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants
19. Status of Considerations for the revised License Renewal Rule, *Federal Register*, Volume 60, No. 88, dated May 8, 1995
20. Memorandum from A. Vietti-Cook to W. D. Travers, "Staff Requirements-SECY-99-148, Credit for Existing Programs for License Renewal," dated August 28, 1999
21. Letter from D. A. Powers to G. J. Dicus, "SECY-99-148, 'Credit for Existing Programs for License Renewal," dated July 19, 1999
22. Letter from G. E. Apostolakis to R. A. Meserve, "Report on the Safety Aspects of the License Renewal Application for Arkansas Nuclear One, Unit 1," dated May 18, 2001
23. Letter from G. E. Apostolakis to E. D. Travers, "Interim Letter Related to the License Renewal of Edwin I. Hatch Nuclear Station, Units 1 and 2," dated May 18, 2001
24. Letter from G. E. Apostolakis to R. A. Meserve, "Proposed Final License Renewal Guidance Documents," dated April 13, 2001
25. Memorandum from S. T. Hofman, "Summary of License Renewal Meeting Between the U.S. Nuclear Regulatory Commission (NRC) and the NEI License Renewal Working Group; Entergy for Arkansas Nuclear One, Unit 1 (ANO-1); SNC for Hatch Nuclear Plant, Units 1 and 2 (Hatch); and FPL for Turkey Point, Units 3 and 4," dated May 13, 2001

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

483rd FULL COMMITTEE MEETING

JUNE 6- 8, 2001

JUNE 6, 2001
Today's Date

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

483rd FULL COMMITTEE MEETING

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

483rd FULL COMMITTEE MEETING

JUNE 6- 8, 2001

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

483rd FULL COMMITTEE MEETING

JUNE 6- 8, 2001

JUNE 7, 2001
Today's Date

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Shu Bahadur	RES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

483rd FULL COMMITTEE MEETING

JUNE 6- 8, 2001

JUNE 7, 2001
Today's Date

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①

ITEMS OF INTEREST

483rd ACRS MEETING

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ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
483rd MEETING
June 6-8, 2001

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SPEECHES

NRC NEWS

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DECOMMISSIONING THE UNCERTAINTY
Commissioner Nils J. Diaz
Remarks Before the NEI/EPRI Decommissioning,
Planning, and Technology Forum
New Orleans, LA
April 30, 2001

Decommissioning of nuclear power plants, or any facility contaminated or containing radioactive components, is today one of the most demanding and taxing industrial undertakings. This is not only because of the complexity of the operations and the painstaking care that must be exercised for the protection of workers and the public; it is because the regulatory and legal issues are tangled in a myriad of fronts, where multiple interests are at play -- and not necessarily converging. Convergence on acceptable solutions is not occurring because uncertainty still prevails in "how to" decommission radioactive installations, and uncertainty is a Medusa. Uncertainty tries to gobble up what is known and to contaminate what it touches.

Uncertainty is the common enemy of the regulator, the industry and of the people. Uncertainty delays solutions and, consequently, it does not provide finality. It feeds on itself and propagates into areas where it does not belong. And, it certainly can be used to confuse, destabilize and create more uncertainty. Reducing, bounding or minimizing uncertainty, whenever possible, is an important generic need facing the civilian uses of nuclear energy and radiation.

One of the fundamental reasons to have regulation is to decrease uncertainty in the implementation of a nation's interests. Regulation must be a convergent process directed at order, balance, equity, and fairness. But, what is regulation? When I am at a loss for words, I go to Webster's dictionary. I found that "to regulate is to govern or direct according to law, to bring activities under control, to reduce to order in a disciplined manner".... Another thing I do when I can not find words is to create a picture (See Figure 1).

Before I focus on the issue of nuclear decommissioning, I will touch on a couple of fundamental considerations and where regulation fits in our society. A democratic society and a free market provides the most powerful combination for achieving fairness, equity, and the protection of rights, property, health and safety. Moreover, I strongly believe that the free flow of information is crucial in a democracy. I also believe that the free flow of information is crucial for a free market to operate for the benefit of all. If used well, information anchors democracy, even if you don't like what you're hearing or seeing. Democracy needs checks and balances. The free flow of information shines light on the checks and balances. Along with the democratic and free market cornerstones sits a force that feeds on information and that can be used to build or to destroy, to add checks and balances or to skew, to advance democracy and improve quality of life or to arrest the democratic and the free market forces. It is called regulation. ...And what a good thing it can be to enhance democracy and its benefits! ...And what a bad thing it can be if misguided, if uncontrolled, or if it is driven by anything other than the common good.

Regulation is a tool of society to implement what society needs, in an orderly, equitable, and fair manner. It must minimize uncertainty.

I believe that the role of proper regulation is fully compatible with the goals and objectives of our democracy and our free market economy. Therefore, regulation has to provide a meaningful and useful framework for the protection of rights, health, safety, and the environment.

Going back to Webster's for another definition for regulation, it also states that regulation is "to prescribe or control by regulations" (See Figure 2). Perfect. We have done a lot of prescribing and a lot of controlling... and it worked reasonably well but not as well as it should. For about three and a half years, the Nuclear Regulatory Commission has been upgrading the prescribing and controlling factors in regulation, to decrease uncertainty in the exercise of our mandate for protection of public health, safety and the environment, thus increasing predictability and transparency (See Figure 3).

Regulation exists, therefore, to provide an effective framework that allows for the conduct of individual, industrial, commercial, financial, and other activities in a disciplined manner. Although all regulations restrict, good regulations should not deter beneficial activities, but frame them and guide them.

Regulatory actions need to be based on facts, but facts and figures that are placed carefully in the proper context and supported by the best available knowledge and experience. Regulators, therefore, must be mindful of the need to make policy decisions based on unbiased, substantiated and fully-informed state-of-the-art information. As we all know, the best intended efforts can produce misleading results if not placed in the proper context, balanced and checked by the body of knowledge and experience. Let me give you a fictitious example.

A government agency decided, after a favorable poll, to focus its resources on increasing the life span of its citizens. Rulemaking was expected, so two totally independent studies, conducted in isolation, were commissioned with the expectation that some convergence of results would make decision-making achievable within the life span of the agency. To everyone's surprise, the studies arrived at two drastically different conclusions, based on the same mortality data. Here are the results:

(See Figure 4)

Study 1: Everyone that does not receive medical attention eventually dies.

Recommendation: Establish a plan to require that everyone receives mandatory health care at a significantly increased frequency. Monitor improvements for 100 years and report to the Secretary.

Confirmatory Note: A PRA study calculated the risk of death at one.

(See Figure 5)

Study 2: Everyone that receives medical attention eventually dies.

Recommendation: Establish a plan to require that all health care systems be eliminated. Monitor improvements for 100 years and report to the Secretary.

Confirmatory Note: A PRA study calculated the risk of death at one.

It should not go unnoticed that, for the first time in history, two PRA studies got the same result.

These divergent recommendations were based on facts, although it should be noted that the panels did not address the minor issue of quality of life. This would be the objective of a follow-up study. It should also be pointed out that the substantial cost of the two plans were comparable: more health care on Plan 1 and a lot more lawyers on Plan 2 (See Figure 6).

Caught in the ensuing controversy, the Secretary had to announce to the nation that: (See Figure 7) "It is not possible, generically, to rule out the possibility of death." Seriously, if you set out to find the impossible, you will not find it, and what you do find could lead you to the wrong conclusions.

In the nuclear arena. whether done by regulated or regulator, the analysis and results regarding safety need to be comprehensive and, today, they need to be integral. Many licensee activities and regulatory activities today need the additional know-how intrinsic in risk-informed regulation. And, what is risk-informed regulation? Webster's failed me on this one, so I have to present you with my own definition of risk-informed regulation:

(See Figure 8)

Risk-informed regulation is an integral, increasingly quantitative approach to regulatory decision-making that incorporates deterministic, experiential and probabilistic components to focus on issues important to safety, which avoids unnecessary burden to society.

This definition can also be used for risk-informed operations, risk-informed maintenance, risk-informed engineering, risk-informed decommissioning....

One of the added values of risk-informed regulation is the reduction of uncertainty in decision-making. Today, we are learning how to mix and match deterministic. experiential and probabilistic results to improve the predictability of safety considerations and of regulatory requirements. Risk-informed regulation is a systematic upgrade of a functional regulatory system that became outmoded. Risk-informed regulation, slowly and often painfully, replaces the not-so-well-founded prescriptions with what is better known, and preferably quantified. It is not a probabilistic recipe, but a comprehensive methodology that uses state-of-the-art know-how. Risk-informed regulation employs a mix of components to achieve an improved approach to regulating. No one component is perfect nor does it need to be; the whole is much better than any of the parts.

Both the NRC and the industry have made significant progress in reducing uncertainty and improving predictability in the nuclear power reactor area. However, the same progress has not been made in the decommissioning area. I believe it is time, and the Commission is well aware of the need, to ensure predictability and consistency in the decommissioning area, and all the associated activities. I mean, we need predictability and consistency in the existing decommissioning methodologies and upcoming options, as well as in transportation and waste

disposal. Of course, we also need predictability in the regulatory requirements. For example, there may be advantages to entombment for specific sites, and I mean public health and safety advantages. All of the issues I mentioned, and a few others, are an important part of what I referred to earlier as generic needs facing the civilian uses of nuclear energy and radiation.

Last, but not least, I believe you might have an interest in what policy decisions lie ahead for decommissioning, so let me give you a quick update.

The Commission is currently considering a proposed rule for revising its transportation regulations in Part 71 to make them more compatible with the International Atomic Energy Agency (IAEA) standards and to codify other applicable requirements. At least one of the changes under consideration has a direct application to decommissioning of nuclear facilities. As older facilities are decommissioned, the Department of Transportation and the NRC are being asked to approve the shipment of large components, including reactor vessels and steam generators. These components may contain significant quantities of radioactive material, but they are so large that it is not practical to fabricate authorized packagings for them. Section 71.8 provides that NRC may grant an exemption from the requirements of the regulations in Part 71 that it determines is authorized by law and will not endanger life or property nor the common defense and security. For example, the NRC used the exemption approach when it approved the Trojan Reactor Vessel Package for transport to the disposal facility on the Hanford Reservation.

However, NRC's policy is to avoid the use of exemptions for recurring licensing actions. Therefore, as a lesson learned from the Trojan approval, the NRC staff has identified large component package authorizations as an issue for consideration in the proposed rule. Based on previous experience and public comments, the staff is proposing a "special package authorization" for packages for which compliance with the other provisions in the regulations is impracticable. Such an authorization would be issued on a case-by-case basis, and would apply only in limited circumstances, and only to one-time shipments of large components.

The NRC has been considering new regulations that would reduce requirements for emergency planning, onsite and offsite insurance, and safeguards for permanently shutdown plants in a step-wise fashion as the potential for offsite releases decreases with time after plant shutdown. The new regulations would address staffing, training, and backfit applicability to decommissioning. As many of you may know, the NRC staff recently issued a technical report addressing spent fuel pool accident risk at decommissioning plants. This report is intended to support rulemaking in the decommissioning area. There are a lot of details in this report and these details must be taken in the proper context.

The principal technical finding of the report is that the risk from spent fuel pools at decommissioning nuclear power plants is low and well within the Commission's safety goals. However, the spent fuel pool risk study also mentions that the possibility of reaching the zirconium ignition temperature could not be precluded on a generic basis. This brings up once again what I have previously called the "zero factor."

As you know, I believe that the zero factor needs to be eliminated and subsumed into reasonable assurance.

Let me restate the established legal requirements for the NRC's radiological protection mission and its relationship to zero risk. It is clear that the courts, interpreting the law, have ruled "the level of adequate protection, need not, and almost certainly will not, be the level of "zero risk." Furthermore, "the courts have long accepted the Commission's definition of its statutory mandate to 'provide adequate protection of public health and safety' as requiring not a risk-free environment, but a 'reasonable assurance'...."

Radiation is radiation, yet radioactive risks are often treated quite differently depending on the source. The risks from radiation need to be scrutinized and given equal treatment under the law. If different treatment of the same radiation risk were of benefit to this country, I would be its strongest advocate. But it is not beneficial and I disapprove of the arbitrary imposition of a zero factor to narrowly selected radiological risks with no importance to public health and safety. I oppose it not only because it is contrary to the law governing the NRC, but also because it hampers debate and gets in the way of good regulation.

Back to decommissioning and the spent fuel pool risk study. I personally would not impose a zero risk requirement on decommissioning activities. However, there is some discussion about additional work that may need to be done on this risk study to support potential regulatory changes for emergency planning, onsite and offsite insurance, and safeguards for permanently shutdown plants. The staff is to present a paper to the Commission on May 31, 2001, which I expect to be a truly risk-informed analysis. This paper will present policy issues with respect to exemptions and regulations for the decommissioning of nuclear power plants. The Commission has stated that we will also consider the comments and recommendations of stakeholders when we receive the staff's paper. The Commission will have to decide whether or not the spent fuel pool risk study needs to be expanded and whether we have adequate information to proceed with rulemaking.

Rulemaking for partial site release is also being developed. Current rules provide adequate protection of the public and the environment from radioactivity remaining at a reactor site when the reactor license is terminated following decommissioning. However, it is possible for a reactor licensee to sell land that would reduce the size of its site before the license termination criteria would specifically apply to the release of the property. The proposed rulemaking would standardize the process for allowing a licensee to release part of its reactor facility or site for unrestricted use (partial site release) before receiving NRC approval of its license termination plan (LTP). A proposed rule is scheduled to be published in 2002.

There are lessons to be learned from our decommissioning experience. One lesson that we have learned is that you cannot set aside a problematic issue in the hope that it will resolve itself in time. Unresolved issues create uncertainty because the final solution is unclear. It is seldom the case that difficult issues will resolve themselves in time. Challenging issues should be resolved when they arise or they can multiply, causing additional uncertainty. Delays in long-term solutions have also caused additional issues to arise, such as the capabilities of spent fuel pools to hold additional fuel, and the regulatory requirements that should be in place for decommissioned reactors that still have spent fuel on site. We are attempting to address these issues now.

In summary, the NRC has a good process in place -- still in need of improvement -- to conduct the decommissioning of nuclear power plants. I

recommend that you actively participate as stakeholders as we evaluate the needed regulatory changes in this area to help reduce uncertainty. You should be providing solutions or somebody else will. By taking advantage of the experience gained by those undergoing decommissioning you should be able to anticipate the issues that may be raised by the regulator and other stakeholders. The lesson in decommissioning, as in all regulatory actions, is always to be aware of and address the uncertainty -- the Medusa.

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4

NRC NEWS

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Regulating for the Common Good
IG Planning Conference
Keynote Address
by
Chairman Richard Meserve
U.S. Nuclear Regulatory Commission

May 21, 2001

Good morning, and thank you for providing me the opportunity, once again, to address your Annual Planning Conference. I was pleased to learn that the theme for this year's conference is "NRC's New Regulatory Approach and How Effective Communication is Key." The subject is singularly important.

I believe that we cannot accomplish our mission to protect public health and safety and the environment without public confidence and trust in the decisions the Commission and its staff make. The most significant way we can instill public trust and confidence is to engage our stakeholders in effective communication. The two most important aspects of effective communication are clarity and openness.

In our efforts to build public confidence, I believe the existence of an independent office such as yours is a valuable asset. While we may believe that we are communicating clearly and openly, the external perspective may be quite different. Thus, the OIG, through your audit and investigative programs, provides the Commission valuable insights on how well we are communicating and, in the end, whether we are fostering public trust and confidence. I will return to the important role you play in a moment.

However, let me first explain why clear and open communication is essential to increasing public confidence in NRC and in the decisions we make. I will also explain what we are doing to achieve it.

Why Open and Clear Communication?

The regulation of the civilian uses of radioactive materials is a highly technical activity, involving scientific analysis and engineering judgment. It might, therefore, be easy to conclude that, because most members of the general public do not understand such concepts as conditional core damage frequency, special treatment requirements, or emergency core cooling systems, it is, therefore, counterproductive to expend time and energy to involve the public. This conclusion is simply wrong.

First of all, many of our stakeholders, either through formal training or through their involvement with the NRC over a period of time, are quite well-versed in the technical details of the facilities we regulate and are familiar with the NRC's regulations. And for those who are not, our challenge is to explain these details in terms that are understandable. It is not enough to be technically precise in what we say; we must also say it in a way that the public will understand. In other words, we must speak in plain English.

Perhaps one of the greatest challenges we face in communicating with the public is effectively conveying the true risk from nuclear power. Although one recent poll indicates that a majority (65%) of those polled think that nuclear power plants are safer now than they were 10 years ago,[1] there are segments of our population that remain concerned that the risks are too great and that the technology represents a significant threat to the public health, safety and the environment. Others worry about the collective ability to safeguard nuclear materials so that untoward uses of them are avoided. And others are worried about the risks attendant to nuclear waste and the potential hazard that these materials present to future generations. If nuclear power is to remain an option in the mix of energy sources, these concerns must be confronted. And, given the enhanced public and political interest in energy, most recently reflected in the Administration's Energy Policy

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Report,[2] the pressure to confront these matters has recently been magnified.

I mean the words "must be confronted" quite literally. Although our regulatory decisions may have a veneer of technical detail, at core they usually implicate embedded social judgments about the acceptability of risk and the balance of costs and benefits. These social judgments are matters on which the public has a stake and on which the affected public is entitled to have its concerns addressed. There is thus a substantive imperative for the regulator to involve the public in its decision-making. Indeed, the public may on occasion bring to light issues that deserve careful attention that otherwise would not have been examined.

Equally important, there is a procedural imperative to make such licensing decisions through processes accessible to the public. In the absence of such transparency, skeptics who do not have access to the regulatory process cannot be blamed for suspicions that their concerns have not been considered. No matter how careful a job we may do, if our work is performed behind a veil of secrecy, the public will not have confidence that the result is fair, objective, honest, or in the public interest. In the end, it is likely that the public will not accept the decision. There always will be the corrosive suspicion that decisions made outside the sight of the public serve to protect those favored by the decisions, to conceal dangers, or to cloak imprudent, unethical, or illegal acts.

Let me give you a specific example. Although the primary objective of the NRC is to protect public health and safety, we have established certain other performance goals. One of these goals, of course, is the one I've been discussing, increasing public confidence. We also have a goal of reducing unnecessary regulatory burden. Based on four decades of experience with operating nuclear power reactors and on improved techniques of probabilistic risk assessment, we now recognize that some regulations imposed in the past may not serve their intended safety purpose and, therefore, may not be necessary to provide adequate protection of public health and safety. Where that is the case, we should revise or eliminate those regulations, because they are not required to achieve our mission. (Of course, insights about risk can also reveal shortcomings in the current regulatory system and these are also being addressed.) However, some members of the public may view these activities as aimed solely at relaxing our safety regulations so as to make life easier for our licensees. Although this is not the case, it does challenge us to communicate effectively how these changes actually will improve our focus on safety.

At the same time that the NRC is using insights about risk to examine the regulatory program, the U.S., as I am sure all of you are aware, is experiencing a dramatic change in the economic conditions within which the nuclear electric power industry operates. Until recently, the rates that generators received for their service were regulated, state by state. Licensees could readily recover the costs of meeting safety requirements in the state-regulated rate base. Within the last year or two many states have deregulated electricity prices and many more are expected to do so in the future. (California may cause some delay, but the general trend is still clear.) The result is that nuclear-generated electricity now must compete in an open market with other sources of electric power. The costs of our regulatory system now come directly off the economic bottom line, and affect the economic competitiveness of nuclear power.

Although the effort to risk-inform the regulatory system started long before the change in the economic climate, the juxtaposition of the two activities can invite skepticism. How is the public to be assured that the changes in safety regulations that we adopt are not merely intended to promote the economic interests of the industry? Of course, the NRC does not promote nuclear power; that is the responsibility of the Department of Energy. However, this fact does not prevent the question from being asked. And the only way we can satisfy the skeptics is by fully revealing the substance of our efforts to revise our regulatory program so as to show that our actions are reasonable and appropriate. Without a clear and open process, the public cannot be assured that our focus is indeed on health and safety, as it must be, and not on the financial interests of our licensees.

Let me mention one other demand for clear and open communication that arises from the current economic changes. The new climate of economic competitiveness holds the danger of creating an environment in which heightened concerns about nuclear power might fester if not addressed forthrightly. Some may fear, for example, that the new economic environment creates incentives for licensees to cut corners on safety in order to improve the bottom line. It is our responsibility to ensure that such actions are not taking place. And it is equally our responsibility to keep the public informed of our findings so that there can be an accurate factual foundation for the public's perceptions. Fortunately, our review to date has shown the improved economic performance and improved safety performance go hand-in-hand. The changed economic environment, in fact, may be providing increased incentives for safety because a safe plant is also one that is reliable. Ultimately, however, we must ensure that the public is fully and accurately informed of licensee performance so that needless fears are avoided.

Recent Initiatives

I have tried thus far to provide an explanation for the importance of clear and open communication. Let me now turn to some of the ways in the which the NRC is working to achieve this objective.

One fundamental way to achieve clear and open communication is to ensure access to information. We maintain a Public Document Room in which materials are made available to the public. We are also trying to harness information technology so that these materials will be more readily available electronically, offering the prospect for timely and easy access throughout the world. Most of you are familiar with ADAMS and the problems we've had both internally and externally. We are implementing an Action Plan to address these problems and we recently received an independent assessment of ADAMS conducted by the Harvard Computer Group. I am hopeful that we can make significant improvements in our document management system over time. Since we are constrained by software, improvements will not be as rapid as we all would like.

In addition, the Commission is making use of technology to facilitate the hearing process on a potential application for a license to construct a high level waste facility at Yucca Mountain. Through the development and establishment of the Licensing Support Network (LSN), we also to hope increase effective public participation in the hearing process by ensuring timely public access to those documents of potential relevance to the application. Finally, we are working on redesigning the NRC's website. We recognize that the Internet has become an important vehicle

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for making information widely available. The feedback we have received has impressed us with the need to upgrade and redesign our site so that it is more user friendly, is more easily navigated, and provides a richer variety of current mission-related information about the NRC's regulatory activities. We hope to have the revised website in place by September of this year.

Another important aspect of clear and open communication is face-to-face interaction with stakeholders through public meetings. In addition to our public hearing processes for licensing actions, the Commission and the staff routinely conduct public meetings so that the public has opportunities to learn about proposed actions and to express views about the proposals and NRC decisions. Many NRC staff meetings are held in the affected communities, often in the evening, so that all segments of the public can participate. These meetings are extraordinarily popular and usually result in important, mutually informative exchanges.

Finally, we will fail in instilling public confidence if the public fails to understand us. Thus, it is important to avoid using jargon and technical language when speaking to the public. We've have adopted a "plain language" initiative and provide formal training for both our managers and staff on the art of conducting public meetings. The ability to organize and conduct meetings that promote open, effective communications is not a natural one, but it is one that can be learned. Because public meetings often address controversial issues, our staff must be able to provide participants with clear and accurate information. Moreover, the staff must be mindful that half of communication is listening. And thus the staff must be trained to listen carefully and thoughtfully and to react responsively to the views and concerns of others. Our new training courses are aimed at reenforcing a cultural climate of openness and providing our staff with the skills to be responsible shepherds of honest open processes.

Another initiative is to develop explicit communications plans for important activities in our major programs, such as licensing, spent fuel storage, and inspection. The objective is to provide guidance to our staff who routinely work in these areas so their communications with the public are consistently thorough and complete. We want to avoid, for example, inadvertent omissions that could be misinterpreted as attempts to conceal information, thus needlessly creating suspicions. We have already used a communications plan to explain our response to the failure of a steam generator tube at the Indian Point 2 plant. The plan provided a useful framework to guide public discussion of the relevant issues and to facilitate public access to the ongoing decision-making process.

Although I believe we have made great strides forward, I also recognize that we have not fully achieved our goal. In April, the staff conducted a public participation issues workshop. At that workshop, the staff heard concerns ranging from a general sense that the NRC does not listen or understand its stakeholders to problems encountered using ADAMS, and concerns regarding the timing and location of meetings, and the appropriate level of public participation in public meetings. We are working to address these and other matters raised at the workshop. The staff is developing an action plan which I expect to see by the end of July.

Role of OIG

As the NRC strives to be more open, we welcome the scrutiny that OIG provides. As I briefly mentioned in the beginning of my remarks, you are in a unique position to see the agency both as an insider and as an outsider. Your independent audits and evaluations provide valuable insight in areas where we have both achieved our objectives and, more importantly, where we may have failed. I note that OIG listed "clear and balanced communication with external stakeholders" as among the top management challenges that faces the agency.[3] I couldn't agree with you more. Recent OIG audits have also focused on specific issues associated with NRC's efforts to improve its communication with the public, including a report on the NRC's Quality Assurance Process for Official Documents[4] and an evaluation of the status of NRC's website.[5] Both of these reports provided helpful information, and we look forward to more of these audits in the future, particularly the results of your planned audit of ADAMS during FY 2001.[6]

Conclusion

Let me reiterate my view that responsible openness is an essential ingredient in regulatory decision-making. As conscientious public servants, we cannot be successful if we are seen as being secretive. Clarity and openness are particularly important for decisions in highly technical areas because otherwise the public has scant opportunity to understand the issues or to participate in a meaningful way. We want the public to continue to have confidence that the NRC will carry out its mission to protect health and safety in the public interest, and we are investing in our staff and programs to enhance that trust. I look forward to your continued scrutiny as we work towards increasing public confidence in our decisions.

Thank you for the opportunity to join you today.

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

1. Associated Press Poll on Nuclear Power, April 25, 2001.

2. "Reliable, Affordable, and Environmentally Sound Energy for America's Future," Report of the National Energy Policy Development Group, May 17, 2001.

3. Special Evaluation of NRC's Most Serious Management Challenges, OIG-00-A-04 Audit Report January 31, 2001.

4. OIG-01A-02: Review of NRC's Quality Assurance Process for Official Documents, February 23, 2001.

5. Special Evaluation of the Status of NRC's website, OIG/OOE-08, April 20, 2000.

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6. Office of the Inspector General FY 01 Annual Plan, Appendix B: Information Technology and Administrative Team Audits Planned for FY 2001, "Review of Agencywide Documents Access and Management Systems (ADAMS)."

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May 4, 2001

EA-99-234

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: ORDER IMPOSING CIVIL MONETARY PENALTY - $110,000 TENNESSEE VALLEY AUTHORITY

Dear Mr. Scalice:

This refers to your letters dated January 22, 2001, and March 9, 2001, in response to the Notice of Violation and Proposed Imposition of Civil Penalty (Notice) sent to you by our letter dated February 7, 2000. Our letter and Notice described one violation of 10 CFR 50.7, "Employee Protection," which was described in NRC Office of Investigations (OI) Report No. 2-98-013. To emphasize the importance of a safety conscious work environment that is free of discriminatory employment actions, a civil penalty of $110,000 was proposed.

In your response of January 22, 2001, you denied the violation and protested the proposed civil penalty. You contended that the reorganization of Tennessee Valley Authority (TVA) Nuclear in 1996, the elimination of the position of Chemistry and Environmental Protection Program Manager, Operations Support, and the selection of individuals to fill new positions were made solely for legitimate business reasons, and were not in any part taken as retaliation for the Chemistry and Environmental Protection Program Manager's engagement in protected activity.

Your letter of March 9, 2001, provided a supplemental response to our Notice, related to comments submitted to the NRC's Discrimination Task Group by a former NRC Office of Enforcement (OE) staff member. As background, on July 27, 2000, the NRC established a management-level review group to evaluate the NRC's processes used in the handling of discrimination allegations and violations of employee protection standards. The Discrimination Task Group is an ongoing effort whose overall objective is to develop recommendations for revisions to the regulatory requirements, the Enforcement Policy, or other Agency guidelines as appropriate. The former OE staff member's comments involve his perceptions that the NRC has lowered its threshold for taking enforcement action for discrimination, and fails to properly consider a licensee's position that adverse actions taken against their employees were done for legitimate business reasons. TVA considers these comments to be significant because the former OE staff member was involved in the subject escalated action taken against TVA, and because TVA's response of January 22, 2001, also raised these two issues.

After considering your responses, for the reasons given below and in the February 7, 2000, letter and Notice, we have concluded that the violation occurred as stated and that neither an adequate basis for withdrawing the violation, reducing the severity level, or mitigating or rescinding the civil penalty has been provided. In July 1996, TVA eliminated the Chemistry and Environmental Protection Program Manager's position in Operations Support, as part of a reorganization, and took subsequent actions to ensure that he was not selected for one of two new positions within Operations Support. TVA took these actions, at least in part, in retaliation for his involvement in protected activities. These activities included the identification of chemistry related nuclear safety concerns in 1991-1993, and the subsequent filing of a Department of Labor (DOL) complaint in September 1993 based, in part, on these chemistry related nuclear safety concerns. Certain TVA managers were aware of his protected activity when the selection process, designed by these same managers, failed to select him for one of the two new positions.

The selection process for the newly created Chemistry Program Manager positions in Operations Support was not in accordance with TVA's normal process. TVA's rationale for posting the Chemistry Program Manager position and requiring individuals to compete for selection, while filling the Radcon Chemistry Manager position without posting it in 1996, were inconsistent. In both cases, the individuals had previously performed the functions of the new positions they were seeking, yet in the case of the former Chemistry and Environmental Program Manager, he was not permitted to fill the position noncompetitively as had the Radcon Chemistry Manager. Moreover, TVA's explanations with respect to the decision making process for the filling of the Radcon Chemistry Manager position changed over time.

Regarding TVA's supplemental response of March 9, 2001, the NRC welcomes and intends to consider all information provided to the Discrimination Task Group by internal and external stakeholders in accomplishing the overall objective of developing recommendations for revisions to the regulatory requirements, the Enforcement Policy or other agency guidelines as appropriate. However, the NRC has concluded that your response provides no new information related to the specific circumstances of the Notice that would warrant a change in the subject enforcement action.

Accordingly, we hereby serve the enclosed Order on Tennessee Valley Authority imposing a civil monetary penalty in the amount of $110,000. As provided in Section IV of the enclosed Order, payment should be made within 30 days in accordance with NUREG/BR-0254. In addition, at the time payment is made, a statement indicating when and by what method payment was made, is to be mailed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738. We will review the effectiveness of your corrective actions during a subsequent inspection.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice", a copy of this letter and the enclosures will be made available

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electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/signed by/

William F. Kane
Deputy Executive Director
for Regulatory Programs

Docket Nos. 50-390, 50-327, 50-328, 50-269, 50-260, 50-296

License Nos. NPF-90, DPR-77, DPR-79, DPR-33, DPR-52, DPR-68

Enclosure: Order Imposing Civil Monetary Penalty

cc:
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Electronic Mail Distribution

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County Executive
Hamilton County Courthouse
Chattanooga, TN 37402-2801

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	Docket Nos. 50-390, 50-327, 50-328,
	)	50-269, 50-260, 50-296
Tennessee Valley Authority	)	License Nos. NPF-90, DPR-77, DPR-79,
Watts Bar Nuclear Plant, Unit 1	)	DPR-33, DPR-52, DPR-68
Sequoyah Nuclear Plant, Units 1 & 2	)	
Browns Ferry Nuclear Plant, Units 1, 2 & 3	)	EA 99-234

ORDER IMPOSING CIVIL MONETARY PENALTY

I

Tennessee Valley Authority (Licensee) is the holder of Operating License Nos. NPF-90, DPR-77, DPR-79, DPR-33, DPR-52, DPR-68, issued by the Nuclear Regulatory Commission (NRC or Commission) on February 7, 1996, September 17, 1980, September 15, 1981, December 20, 1973, August 2, 1974, and July 2, 1976. The licenses authorize the Licensee to operate Watts Bar Nuclear Plant, Unit 1, Sequoyah Nuclear Plant, Units 1 and 2, and Browns Ferry Nuclear Plant, Units 1, 2, and 3, in accordance with the conditions specified therein.

II

An investigation of the Licensee's activities was completed on August 4, 1999. The results of this investigation indicated that the Licensee had not conducted its activities in full compliance with NRC requirements. A written Notice of Violation and Proposed Imposition of Civil Penalty (Notice) was served upon the Licensee by letter dated February 7, 2000. The Notice states the nature of the violation, the provision of the NRC's requirements that the Licensee had violated, and the amount of the civil penalty proposed for the violation.

The Licensee responded to the Notice in letters dated January 22, 2001, and March 9, 2001. In its response, the Licensee denied the violation and protested the proposed imposition of a civil penalty.

III

After consideration of the Licensee's response and the statements of fact, explanation, and argument for mitigation contained therein, the NRC staff has determined that the violation occurred as stated and that the penalty proposed for the violation designated in the Notice should be imposed.

IV

In view of the foregoing and pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205, IT IS HEREBY ORDERED THAT:

The Licensee pay a civil penalty in the amount of $110,000 within 30 days of the date of this Order, in accordance with NUREG/BR-0254. In addition, at the time of making the payment, the Licensee shall submit a statement indicating when and by what method payment was made, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738.

V

The Licensee may request a hearing within 30 days of the date of this Order. Where good cause is shown, consideration will be given to extending the time to request a hearing. A request for extension of time must be made in writing to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. A request for a hearing should be clearly marked as a "Request for an Enforcement Hearing" and shall be submitted to the Secretary, U.S. Nuclear Regulatory Commission, ATTN: Rulemakings and Adjudications Staff, Washington, DC 20555. Copies also shall be sent to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, to the Assistant General Counsel for Materials Litigation and Enforcement at the same address, and to the Regional Administrator, NRC Region II, 61 Forsyth Street, SW, Suite 23T85, Atlanta, Georgia, 30303-8931.

If a hearing is requested, the Commission will issue an Order designating the time and place of the hearing. If the Licensee fails to request a hearing within 30 days of the date of this Order (or if written approval of an extension of time in which to request a hearing has not been granted), the provisions of this Order shall be effective without further proceedings. If payment has not been made by that time, the matter

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may be referred to the Attorney General for collection.

In the event the Licensee requests a hearing as provided above, the issues to be considered at such hearing shall be:

(a) whether the Licensee was in violation of the Commission's requirements as set forth in the Notice referenced in Section II above, and

(b) whether, on the basis of such violation, this Order should be sustained.

FOR THE NUCLEAR REGULATORY COMMISSION

/signed by/

William F. Kane
Deputy Executive Director
for Regulatory Programs

Dated at Rockville, Maryland
this 4th day May 2001

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May 2, 2001

EA-01-011

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, 2nd Floor
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NOTICE OF VIOLATION
(NRC OFFICE OF INVESTIGATIONS
CASE NO. 1-1999-015, Nine Mile Point 1)

Dear Mr. Mueller:

This letter refers to the results of an investigation initiated by the NRC Office of Investigations (OI) on May 12, 1999, at Nine Mile Point Nuclear Station, Unit 1, to determine whether a former NRC-licensed chief shift operator (CSO) had deliberately provided false, inaccurate, or incomplete information on health history forms. The forms were required by Niagara Mohawk Power Corporation (NMPC), as part of the Fitness-For-Duty (FFD) program and medical certification process required for licensed operators. Based on the evidence developed during this OI investigation, the NRC determined that the former CSO deliberately failed to provide truthful, accurate and complete information on the health history forms for the purpose of misleading your Medical Review Officer (MRO).

In an NRC letter dated March 1, 2001, the NRC provided you a factual summary of the OI investigation, including a basis for the finding, and indicated that an apparent violation of 10 CFR 50.9 occurred. The letter also provided you an opportunity to either (1) respond in writing to the apparent violation within 30 days of the date of that letter, or (2) request a predecisional enforcement conference. In a letter dated March 30, 2001, you provided a written response and indicated that you do not dispute the fact that the former CSO deliberately provided false, inaccurate, or incomplete information on the health history forms.

The NRC has completed its evaluation of the information set forth in the OI report, as well as the information provided in your March 30, 2001, response, and based upon that review, the NRC has determined that a violation of NRC requirements occurred. The CSO's actions caused NMPC to be in violation of 10 CFR 50.9(a) which states that information required by the Commission's regulations to be maintained by a licensee shall be complete and accurate in all material respects. 10 CFR 55.27 requires licensees to document and maintain the results of medical qualification data, test results, and each operator's medical history and to provide the documentation to the NRC upon request.

The information on the health history forms, which were completed by the former CSO in December 1996 and October 1997, was false, inaccurate, and incomplete, in that the CSO denied taking any medications and being under the care of a health care provider, when, in fact, the CSO was taking prescription medications and was under the care of two health care professionals. As a result, the MRO, when reviewing the inaccurate forms, was precluded from making a fully informed decision about the CSO's medical qualifications to perform licensed activities.

On the health history form signed and dated by the CSO on December 11, 1996, the CSO checked "No" to the following two questions: "Taken or are you currently taking any medications (prescription and/or non-prescription)," and "Been treated for any illnesses or injuries." On a subsequent health history form signed and dated by the CSO on October 8, 1997, the CSO checked "No" to the following question: "Presently under a health care provider's care for any condition," and did not list any medications in response to the following question: "List any medications you are currently taking (prescription and/or over the counter)." These answers were considered deliberately inaccurate because the CSO admitted, during a transcribed interview with OI, that at the time the CSO filled out the health history forms, the CSO was taking prescription medications and was being treated by two health care professionals.

As noted in our March 1, 2001 letter, NMPC had an opportunity in 1996 to address this situation prior to NRC involvement. The OI investigation revealed that the CSO had confided to a station shift supervisor (SSS) that the CSO was taking prescription medications at the time. Although the SSS advised the CSO to report this information to the medical department, the SSS never followed up to inform the medical department or to check if the CSO informed the medical department. In your March 31, 2001 response, you indicated that the SSS in question (1) did not realize that it was a requirement to follow up to ensure that the individual had notified FFD personnel, (2) believed that the medication being used was not a detriment to the individual's performance of shift duties, and (3) continued to observe the CSO's performance of licensed duties. You also indicated that the SSS now recognizes that he, as a supervisor, must notify the Site Medical group in accordance with the procedure and then continue to monitor the specific situation.

This case involved a licensed official (the licensed CSO) creating inaccurate information that was required to be maintained by NMPC, and which had the capability of influencing your Medical Review Officer. Therefore, the violation has been classified at Severity Level III in accordance with the Section C.2 of Supplement VII or the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600.

In accordance with the Enforcement Policy in effect at the time this violation occurred and was identified, a base civil penalty in the amount of $55,000 is considered for a Severity Level III violation. Because the Severity Level III violation was deliberate, the NRC considered whether

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credit was warranted for *Identification* and *Corrective Action* in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. In this case, the NRC decided that credit for *Identification* is warranted, even though there was an earlier missed opportunity for detection, since you ultimately did identify, in March 1998, that (a) the CSO was using prescription medications and was under the care of two health care professionals, and (b) the health history forms were inaccurate. Credit for *Corrective Action* is also warranted because your corrective actions were considered prompt and comprehensive. These actions included: (1) removal of the CSO, who is no longer employed by NMPC, from licensed duties in March 1998; (2) training of licensed operators, including the supervisor in question, on the notification requirements pertaining to the taking of prescription medications; and (3) revision of FFD procedures to clarify the obligations of a licensed operator to inform management of the use of prescription medications. Therefore, I have been authorized, after consultation with the Director, Office of Enforcement, not to propose a civil penalty in this case.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in your March 31, 2001 letter. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

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

Sincerely,

/RA/

Hubert J. Miller
Regional Administrator

Enclosure: As Stated

Docket No. 05000220
License No. DPR-63

cc w/encl:
G. Wilson, Esquire
M. Wetterhahn, Winston and Strawn
J. Rettberg, New York State Electric and Gas Corporation
P. Eddy, Electric Division, Department of Public Service, State of New York
C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
J. Vinquist, MATS, Inc.
W. Flynn, President, New York State Energy Research and Development Authority
J. Spath, Program Director, New York State Energy Research and Development Authority
T. Judson, Central NY Citizens Awareness Network

NOTICE OF VIOLATION

Niagara Mohawk Power Corporation
Nine Mile Point, Unit 1

Docket No. 50-220
License No. DPR-63
EA-01-011

During an NRC investigation initiated on May 12, 1999, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions," NUREG-1600, the particular violation is set forth below:

10 CFR 50.9, requires, in part, that information required by statute or by the Commission's regulations, orders, or license conditions be maintained by the licensee and shall be complete and accurate in all material respects.

10 CFR 55.27 requires licensee's to document and maintain the results of medical qualification data, test results, and each operator's medical history for the current license period and to provide the documentation to the NRC upon request.

Contrary to the above, on December 11, 1996, and October 8, 1997, a licensed chief shift operator (CSO) completed and signed health history forms, required for documentation of licensed operator Fitness-For-Duty determination per 10 CFR 55.27, that were not complete and accurate. Specifically,

1. the CSO, on a health history form signed and dated by the CSO on December 11, 1996, checked "No" to the following two questions:

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"Taken or are you currently taking any medications (prescription and/or non-prescription)," and "Been treated for any illnesses or injuries," and;

2. the CSO, on a subsequent health history form signed and dated by the CSO on October 8, 1997, checked "No" to the following question: "Presently under a health care provider's care for any condition," and did not list any medications in response to the following question: "List any medications you are currently taking (prescription and/or over the counter)."

   These answers provided by the CSO on those health history forms were inaccurate and incomplete because, at the time the CSO filled out the health history forms, the CSO was taking prescription medications and was being treated by two health care professionals. These inaccurate answers were material because they misled the Niagara Mohawk Power Corporation's Medical Review Officer and precluded him from making a fully informed decision regarding the CSO's fitness-for-duty.

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

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in the March 31, 2001 letter. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident Inspector at the Nine Mile Point Nuclear Station, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

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

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

Dated this 2nd day of May 2001

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May 14, 2001

EA-01-005

Garry L. Randolph, Senior Vice
President and Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, Missouri 65251

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY - $55,000 (NRC INVESTIGATION REPORT 4-1999-068 and U.S. DEPARTMENT OF LABOR CASE No. 2000-ERA-15)

Dear Mr. Randolph:

This refers to the predecisional enforcement conference conducted on March 7, 2001, in the NRC Region IV office in Arlington, Texas. The conference was held to discuss the NRC's concern that Union Electric's contractor, The Wackenhut Corporation (TWC), had discriminated against a former TWC security officer and a TWC training instructor, in violation of 10 CFR 50.7, at the Callaway Nuclear Plant for identifying a violation of NRC requirements. Our concern was identified to members of your staff during a telephonic exit briefing on January 19, 2001, and was documented in our letter dated February 5, 2001.

After considering the information developed during the NRC investigation and the information provided during the predecisional conference, the NRC has determined that a violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation (Notice). The violation involves a former TWC security officer, who on October 27, 1999, contacted a high school and learned that an individual without a high school diploma or equivalent performance examination had been hired as a temporary watchman at the Callaway Nuclear Plant. The hiring of this individual was contrary to the requirements of 10 CFR Part 73, Appendix B, Section I.A.1.a. The security officer informed a TWC training instructor of this violation, and the training instructor informed TWC officials. On November 19, 1999, TWC unfavorably terminated the security officer, and reprimanded the training instructor for not having brought his concern about the individual's qualifications to the attention of TWC management earlier. Soon afterwards, Union Electric revoked the former security officer's unescorted access authorization based on trustworthiness concerns. On October 28, 1999, TWC unfavorably terminated the temporary watchman who had been improperly hired, and soon afterwards Union Electric terminated (not revoked) his unescorted access authorization. While preparing to respond to the complaint filed by the former security officer with the United States Department of Labor, Union Electric conducted an additional investigation into the educational qualifications of the individual who had been improperly hired, and in August 2000, revoked the temporary watchman's unescorted access authorization based on the falsification of his application for employment and for access authorization.

During the predecisional enforcement conference, Union Electric and TWC representatives asserted that no violation had occurred. Based upon an investigation conducted by the TWC Director of Quality Assurance, Union Electric and TWC managers asserted that the former security officer lacked trustworthiness because she had misrepresented herself to the high school principal, as a licensee screening official performing official business, in order to learn whether the individual in question had a high school diploma and thus to eliminate him as a competitor for a permanent security officer position. Based upon the same investigation, TWC asserted that the training instructor did not meet TWC expectations for a member of the management team because the training instructor had waited until October 1999 to report his concern about the individual's lack of a high school diploma, rather than in August 1999 when the matter first came to the attention of the training instructor. TWC stated that its decisions to terminate the security officer for lack of trustworthiness and to reprimand the training instructor involved no retaliatory intent and were made by TWC corporate managers, not by TWC personnel at the Callaway Nuclear Plant. Union Electric stated that its decision to revoke the former security officer's unescorted access authorization for lack of trustworthiness involved no retaliatory intent.

Based on a review of the circumstances surrounding these events, however, the NRC staff concludes that the former security officer and the training instructor engaged in a protected activity, that TWC managers and Union Electric managers were aware of the protected activity, that TWC managers took adverse actions against the security officer and the training instructor and that Union Electric took adverse action against the security officer, at least in part, because of their protected activity. Our conclusion that retaliation occurred is based, in part, on the following:

(1) TWC concluded that the training instructor should have known in August 1999 to report his concern about the individual's educational qualifications to TWC management. However, the training instructor reasonably believed that any concern about the individual's educational qualification had been properly reported in August 1999. Further, the training instructor was under the same mistaken understanding as his supervisor and the TWC project manager that Union Electric would verify educational qualifications.

(2) The stated intent of the investigation conducted by TWC's Director of Quality Assurance was to determine how TWC had hired the individual when he did not meet the educational requirements of 10 CFR Part 73 Appendix B, in order to take appropriate corrective action. Based upon mere suspicion, however, that investigation quickly became an inquiry into whether the former security officer had learned of the violation by misrepresenting herself to the high school and into her motives for contacting the high school. At the same time, despite the improbability of the individual's claim that he believed he had graduated from high school, the TWC investigation did not make a good faith attempt to determine whether he had deliberately misrepresented his educational qualifications.

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(3) The investigation was conducted with bias against the security officer and the training instructor. Examples of bias include, but are not limited to: (a) The investigative report recommended disciplinary action against the security officer for failing to raise the issue of the individual's lack of educational qualifications through the proper chain of command; (b) The report assumed that the security officer had lied about how she had identified herself to the high school principal and about whether she had reported the matter in August 1999 to the TWC administrative assistant, based upon subjective perceptions of the security officer's "evasiveness" and a change in her handwriting during an interview, while failing to consider the obvious motives of the high school principal and the TWC administrative assistant to not be candid about their interactions with the security officer; and (c) The TWC Director of Quality Assurance relied upon the subjective impressions of and information supplied by a TWC supervisor, without consideration of a warning by the training instructor that information supplied by the supervisor was not reliable.

(4) Union Electric relied upon the biased TWC investigation and report to revoke the former security officer's unescorted access authorization for lack of trustworthiness, while simply terminating the unescorted access authorization of the individual who lacked a high school diploma or equivalent performance examination without conducting an adequate investigation into the individual's trustworthiness. Union Electric did not make a good faith effort to determine whether the individual had deliberately misrepresented his educational qualifications until discovery began in connection with the former security officer's complaint before the United States Department of Labor.

In consideration of the severity of the actions taken against the former security officer and the training instructor, the level of management involved in the adverse action, and the nature of contractor/licensee relationships, this violation has been categorized in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600 at Severity Level III.

In accordance with the Enforcement Policy that existed in 1999, a base civil penalty in the amount of $55,000 was considered for this Severity Level III violation. Because the violation involved willfulness, the NRC Staff considered whether credit was warranted for *Identification* and *Corrective Action* in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. Since the NRC identified the violation, credit for identification is not warranted. Based on our review of Union Electric's corrective actions, corrective action credit is warranted. This results in the assessment of a civil penalty at the base value. During the conference, you identified corrective actions to ensure a safety-conscious work environment (SCWE) by reviewing past allegations for adverse trends, enhancing the Employee Concerns Program (ECP) procedure, developing and posting a SCWE policy, training on the SCWE policy and ECP procedure in General Employee Training for employees and contractors, enhancing the Outage Handbook with guidance on the SCWE policy and ECP procedure, and reviewing with contractor management Union Electric's expectations regarding SCWE. In addition, you assigned responsibility for making future access revocation decisions to a committee of managers.

To emphasize the significance of this violation and the importance of maintaining a safety conscious work environment at the Callaway Nuclear Plant, I have been authorized, after consultation with the Director, Office of Enforcement, to issue the enclosed Notice of Violation and Proposed Imposition of Civil Penalty (Notice) in the base amount of $55,000 for this Severity Level III violation.

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

In addition, the NRC has concluded that Union Electric was also in violation of 10 CFR Part 73, Appendix B Section 1.A.1.a, in that an individual was assigned to the security organization without being in possession of a high school diploma or without having passed an equivalent performance examination. This violation, which you have already corrected, is of minor significance and is not subject to enforcement action in accordance with Section IV of the Enforcement Policy.

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

Sincerely,

/RA/

Ellis W. Merschoff
Regional Administrator

Docket No.: 50-483
License No.: NPF-30

Enclosure: Notice of Violation and Proposed Imposition of Civil Penalty

cc (w/Enclosure):
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19041 Raines Drive

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Derwood, Maryland 20855

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Ronald A. Kucera, Director
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Jefferson City, Missouri 65102

Otto L. Maynard, President and
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Wolf Creek Nuclear Operating Corporation
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Dan I. Bolef, President
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for the Environment
6267 Delmar Boulevard
University City, Missouri 63130

Lee Fritz, Presiding Commissioner
Callaway County Courthouse
10 East Fifth Street
Fulton, Missouri 65251

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Jerry Uhlmann, Director
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Jefferson City, Missouri 65101

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTY

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Union Electric
Callaway Nuclear Plant

Docket No. 50-483
License No. NPF-30
EA-01-005

During an NRC investigation which concluded on November 27, 2000, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the NRC proposes to impose a civil penalty pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The particular violation and the associated civil penalty are set forth below:

> 10 CFR 50.7(a) prohibits discrimination by a Commission licensee against an employee for engaging in certain protected activities. Discrimination includes discharge or other actions relating to the compensation, terms, conditions, and privileges of employment. Under 10 CFR 50.7(a)(1)(i), the activities that are protected include, but are not limited to, the reporting by an employee to his employer information about alleged regulatory violations.
>
> Contrary to the above, The Wackenhut Corporation (TWC), a contractor of Union Electric, a 10 CFR Part 50 licensee, and Union Electric discriminated against a security officer and a training instructor for having engaged in protected activity. Specifically, on October 27, 1999, the security officer and the training instructor identified to TWC a violation of NRC requirements at the Callaway Nuclear Plant, namely that TWC had hired and assigned an individual to the security organization when that individual did not have a high school diploma or equivalent. The hiring of this individual was in violation of 10 CFR Part 73, Appendix B, Section I.A.1.a, which provides that prior to employment or assignment to a security organization, an individual must possess a high school diploma or pass an equivalent performance examination. Based at least in part on this protected activity, TWC unfavorably terminated the security officer's employment for lack of trustworthiness and gave a written reprimand to the training instructor on November 19, 1999, and Union Electric revoked the security officer's unescorted access authorization for lack of trustworthiness.
>
> This is a Severity Level III violation (Supplement VII).
> Civil Penalty - $55,000

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

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

In requesting mitigation of the proposed penalty, the factors addressed in Section VI.C.2 of the Enforcement Policy should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the Licensee is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing a civil penalty.

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

The response noted above (Reply to Notice of Violation, statement as to payment of civil penalty, and Answer to a Notice of Violation) should be addressed to: Mr. Frank Congel, Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the

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NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room). If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 14th day of May 2001

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(2)

RISK-INFORMING 10 CFR 50.46

Presented to
Advisory Committee on Reactor Safeguards

Presented by
Mark Cunningham and Alan Kuritzky
RES/DRAA/PRAB
U.S. Nuclear Regulatory Commission
(301) 415-6189

June 6, 2001

OUTLINE

- Purpose/goal of meeting
- Background - Option 3
- Overview of existing "50.46" requirements
- Options for risk-informed alternative
  - ECCS reliability
  - ECCS acceptance criteria
  - ECCS evaluation model
  - LBLOCA redefinition
- Phase IIA technical work
- Policy issues
- Schedule

PURPOSE/GOAL OF MEETING

- Provide status report on staff's efforts to risk-inform 10 CFR 50.46
- Solicit feedback and comments from ACRS:
  - Options
  - Implementation issues
  - Feasibility
- No letter requested

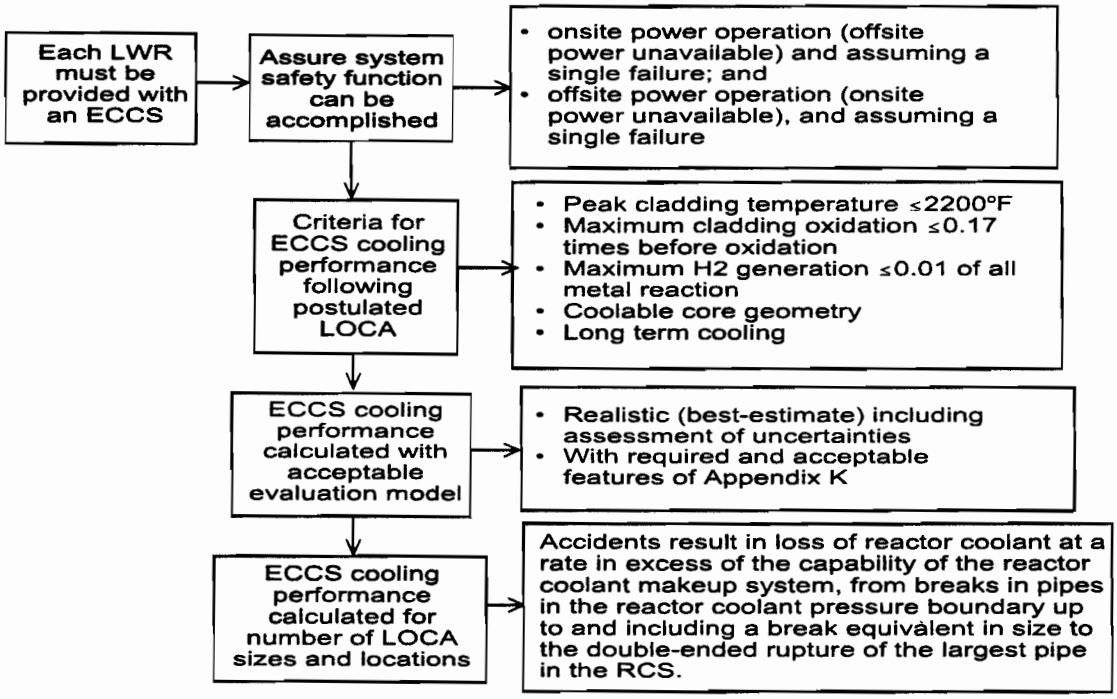
BACKGROUND

SECY-99-264 (Nov 9, 1999) defined plan for Option 3 work

<u>OPTION 3 FRAMEWORK</u>:

- Phase I:
  - Part A: Identify candidate requirement
  - Part B: Prioritize
  - Part C: Evaluate feasibility and provide recommendations to Commission
    - ★ Develop technical content and basis for alternative
    - ★ Identify policy issues
    - ★ Identify required technical work
    - ★ Identify required resources
- Phase II
  - Part A: Perform technical work
  - Part B: Develop and implement rulemaking

OVERVIEW OF 50.46 (including Appendix K and GDC 35)



RISK-INFORMED ALTERNATIVE

Options Being Considered

- Risk-informed requirements such that ECCS reliability is commensurate with the frequency of associated LOCAs
- Improve realism
- Approaches to set ECCS reliability and acceptance criteria:
  - prescriptive criteria
  - performance-based criteria
  - generic versus plant-specific criteria
- Provide option for plant-specific LBLOCA definition, as approved by the Commission (or Director of NRR)

POSSIBLE ECCS FUNCTIONALITY CRITERIA

- Revise requirements to result in ECCS reliability that is commensurate with LOCA frequency
- Two options:
  - Performance-oriented
    - Replace single failure criterion and LOCA/LOOP criterion with ECCS functional reliability
  - Prescriptive-oriented
    - Revise LOCA/LOOP criterion to only apply to the more likely pipe breaks
    - Retain single failure criterion

POSSIBLE ECCS ACCEPTANCE CRITERIA

- Revise the ECCS acceptance criteria such that ECCS performance, for the duration of the accident, is shown to maintain a coolable core geometry
- Two options to ensure a coolable core geometry by ensuring adequate post-quench ductility
  - Performance-oriented
    - Show by tests that cladding integrity is maintained
  - Prescriptive-oriented
    - Limit peak cladding temperature and maximum oxidation

POSSIBLE ECCS EVALUATION MODELS

- Revise the requirements of the evaluation model to be more realistic and address uncertainties in a risk-informed manner
- Three options
  - In Appendix K model, replace current decay heat curve with 1994 ANS standard and NRC-prescribed uncertainty treatment
  - Use realistic model with uncertainty propagation (current best-estimate option)
  - For low frequency large LOCAs (based on 95th percentile of contribution to CDF), apply best-estimate thermal-hydraulic models without formal uncertainty propagation

POSSIBLE LBLOCA REDEFINITION

- Revise the requirement for the number of LOCA sizes and locations by including a provision to allow redefining the LBLOCA if a licensee can demonstrate its technical justification
- Possible rule wording change may include, for example: *"hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary* ***either*** *up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system* ***or an alternate size approved by the Commission (or Director of NRR)****."*

POSSIBLE LBLOCA REDEFINITION (Cont'd)

- Technical justification
  - Estimates of LBLOCA frequency should address
    - Service experience
    - Analysis to predict piping failure frequency
    - Other failures where large break is surrogate
  - Prior analyses addressing piping failure frequency not sufficiently rigorous basis for change to 50.46
    - 1986 change to GDC 4
    - RI-ISI programs
  - Analysis at least as rigorous as PTS

POSSIBLE TECHNICAL WORK TO BE PERFORMED IN PHASE IIA

- Risk
  - Develop method for ensuring reliability requirements are commensurate with challenge frequencies
- Engineering
  - Develop LOCA frequency vs size information sufficient to support development of reliability requirements
  - Interact with industry on technical issues requiring resolution such that LBLOCA DBA can be eliminated
- Thermal-Hydraulics
  - Develop technical basis for evaluation method changes (e.g., decay heat)
  - Develop technical basis for changes to acceptance criteria (e.g., for defining "adequate post-quench ductility")

POLICY ISSUES

Single Failure Criterion

- GDC 35 requires that ECCS safety function be accomplished assuming a single failure
- In 10 CFR Part 50, Appendix A, single failure criterion is applied to more than just ECCS
- As part of risk-informing "50.46" staff considering revising this single failure criterion for the ECCS
- Staff believes that a generic change to the single failure criterion is feasible, and considering recommendation that this be pursued under Option 3

POLICY ISSUES (Cont'd)

Selective Implementation

- In SECY-00-0198, staff recommended that no selective implementation be allowed with respect to the risk-informed alternative to 10 CFR 50.44, so licensees could not preferentially reduce burden without addressing risk-significant concerns
- Commission concurred that selective implementation not be allowed with respect to the risk-informed alternative to 10 CFR 50.44
- Staff considering recommendation that within the context of development of any risk-informed alternative, no selective implementation be allowed

SCHEDULE

- SECY paper due to Commission end of June
- Meet with ACRS, July (letter)

(3)

Signature Estimates of Margin Reductions for Power Uprates/License Renewal

August W. Cronenberg
ACRS Fellow

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Outline

- Margins in the Regulatory Process
- Margin Estimates for Power Uprates
- Margin Estimates for Renewal Plants
- Findings To Date

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Margins in Regulatory Process

- Webster (Margin):
  - Spare amount allowed for contingencies.
  - Bare minimum below which something is no longer desirable.
- General Design Criteria (10CFR50-App. A)

  Criterion 10: Reactor core and associated coolant, control, and protection systems shall be designed with sufficient margin to assure acceptable design limits shall not be exceeded.

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General Design Criteria (continued)

- Criterion-31: The reactor coolant pressure boundary shall be designed with sufficient margin to assure…it behaves in non-brittle manner and probability of rapidly propagating fracture is minimized.
- Criterion-50: The containment, including access openings, penetrations….shall be designed…without exceeding design leakage rate and with sufficient margin…to reflect…metal-water and other chemical reactions…

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Margin Requirements More Explicitly Spelled Out

- Regulatory Guidance/Standard Review Plan (e.g. acceptance criteria for design Press., P-T limits, stress limits, allowable materials, ductility limits, etc.)
- ASME Boiler & Pressure Vessel Code
- Am. Nat. Standards Inst. (ANSI)
- Other

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Impact of Power Uprates on Plant Operating Conditions & Margins

- **Primary System Conditions**
- Core Power/Coolant Enthalpy
- Core Flow Rate/Coolant Temperature
- Fuel Temperature
- BWR-Steam Dome Temp./Press.

- **Secondary System Conditions**
- Steam Generator Flow Rate/Temp.
- Feedwater Flow Rate/Temperature
- Feedwater Pumping Requirements

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Current Power Uprate Applications

Duane Arnold/BWR	15-%	1975
Dresden-2/BWR Dresden-3/BWR	17-% 17-%	1970 1971
Quad Cities-1/BWR Quad Cities-2/BWR	17-% 17-%	1973 1973
Brunswick-1/BWR Brunswick-2/BWR	15-% 15-%	1977 1975
Clinton/BWR	20-%	1987
Arkansas Nuclear One-2/PWR (C-E)	7.5-%	1978

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Margin Estimates for Power Uprates (Hatch Case Study)

- Hatch Plant Characteristics
  - GE-BWR/4 (direct-cycle)
  - Mark-I Containment (inverted light-bulb/ torus suppression pool)
- Power Level
  - 1974 Unit-1/1979 Unit-2 = 2436 MWt
  - 1995 Units 1 & 2 = 2558 MWt (5-% uprate)
  - 1997 Units 1 & 2 = 2763 MWt (8-% uprate)
- License Renewal Application (under review)

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Summary of Hatch Uprate Conditions

Parameter	Hatch Unit-1			Hatch Unit-2		
Thermal Power, MWt (Year at power)	2436 (1975)	2558 (1995)	2763 (1997)	2436 (1979)	2558 (1995)	2763 (1997)
% Power Uprate (from prior value)	–	5-%	8-%	–	5-%	8-%
Core Coolant Flow Rate, 10^6 lb_m/hr (at %-Power)	68.3-82.4 (87-105)	68.3-82.4 (87-105)	68.3-82.4 (87-105)	67-80.9 (87-105)	67-80.9 (87-105)	67-80.9 (87-105)
Vessel Steam Flow, 10^6 lb_m/hr	10.0	10.6	11.5	10.5	11.1	12.0
Steam Dome Pressure, psig	1015	1050	1050	1015	1050	1050
Steam Dome Temp., F	547	551	551	547	551	551
Full-Power Feedwater Flow, 10^6 lb_m/hr	10.1	10.7	11.6	10.5	11.2	12.1
Full-Power Feedwater Temp., F	388	393	398	420	424	425

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Summary Hatch-I Operational Margins

Residual Margin = (Design Limit – Value) / Design Limit

Power Level, MWt	Parameter Value	Residual Margin, %
Main Steam-line Pressure (Design Limit = 1250 psig)		
Original = 2436	1015 psig	18.8
1st Uprate = 2558	1050 psig	16
2nd Uprate = 2763	1050 psig	16
Main Steam-line Temperature (Design Limit = 575 F)		
Original = 2436	546 F	5.04
1st Uprate = 2558	---	---
2nd Uprate = 2763	551 F	4.17

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Summary Hatch-I Operational Margins

Power Level, MWt	Parameter Value	Residual Margin, %
Original = 2436	1130 psig	31.5
1st Uprate = 2558	—	—
2nd Uprate = 2763	1088 psig	34.1
Original = 2436	392 F	30.2
1st Uprate = 2558	—	—
2nd Uprate = 2763	400 F	28.8

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Hatch-I Vessel DBA-LOCA Margins

Power Level, MWt	Predicted Stress, ksi	Residual Margin, %
Original = 2436	8.95 ksi	41.4
1st Uprate = 2558	9.05 ksi	40.8
2nd Uprate = 2763	---	---
Original = 2436	52.7 ksi	24.6
1st Uprate = 2558	53.0 ksi	24.2
2nd Uprate = 2763	---	---
Original = 2436	---	---
1st Uprate = 2558	64.5 ksi	40.1
2nd Uprate = 2763	90.0 ksi	16.4
Original = 2436	31.5 ksi	37.9
1st Uprate = 2558	34.8 ksi	31.4
2nd Uprate = 2763	34.9 ksi	31.2

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Hatch-I Containment DBA-LOCA Margins

Power Level, MWt	Parameter Value	Residual Margin, %
Peak Drywell Pressure (Design Limit = 56 psig/Max=62 psig)		
Original = 2436	47.9 psig	14.5
1st Uprate = 2558	49.6 psig	11.4
2nd Uprate = 2763	50.5 psig	9.8
Peak Drywell Gas Temperature (Design Limit = 281 F)		
Original = 2436	290 F (for short time only)	Exceeds design limit for short time
1st Uprate = 2558	292 F (for short time only)	Exceeds design limit for short time
2nd Uprate = 2763	293 F (for short time only)	Exceeds design limit for short time
Peak Suppression Pool Temperature (Design Limit = 281 F)		
Original = 2436	198 F	29.5
1st Uprate = 2558	202 F	28.1
2nd Uprate = 2763	208 F	26.0

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Hatch-I License Renewal/Margin Summary

$CUF = N_{OBE}/f_1 + N_{Scram}/f_2 + N_{Startup}/f_3 + \ldots\ldots N_{other}/f_n$

Residual Margin (RM) = 1 - [CUF]

Component	Hatch Unit	CUF at 40 yr	RM at 40 yr	CUF at 60 yr	RM at 60 yr
Residual Heat Removal Suction Piping	2	0.57	43-%	0.77	23-%
Reactor Vessel Equalizer Piping	1	0.52	48-%	0.64	36-%
Core Spray Replacement Piping	1	0.16	84-%	0.19	81-%
Feedwater Piping	2	0.61	39-%	0.83	17-%
Standby Liquid Control Piping	1	0.24	76-%	0.25	75-%
Steam Condensate Drainage Piping	2	0.66	34-%	0.89	11-%

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Hatch-I License Renewal/Margin Summary

Relative Residual Margin (RR-Margin) = 1 - $\frac{T_{min}(@\ EFPY) - 157}{157}$

Min. Temperature @ 54 EFPY	291.4 F	14.4
Min. Temperature @ 48 EFPY	283.6 F	19.4
Min. Temperature @ 44 EFPY	277.6 F	23.2
Min. Temperature @ 40 EFPY	271.7 F	26.9
Min. Temperature @ 36 EFPY	265.2 F	31.1
Nil-irradiation damage T-limit at bottom head = 157 F		

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Margin Estimate/Data Sources

- Power Uprate Operational Conditions (e.g. Main-Steamline Press.)
  - Design Limit: FSAR & ULAR
  - Uprate Press: ULAR or ULAR-SAR
- Power Uprate DBA-LOCA Predictions (e.g. bolt stress)
  - Component Design Limit: FSAR & ULAR-SAR
  - Component DBA-LOCA Stress: ULAR-SAR-Appendix (GE sub-contract report; summary tables of LOCA stress for a limited number of components, which can be different than assessed in original FSAR or prior uprate-SAR)
- License Renewal/Component Aging Effects-TLAA
  - LRA/TLAA-Chap. 4
  - CUF (Contractor Refs. To Chap.-4, *Structural Integrity Assoc.)*

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Summary & Observations to Date

- *Safety Margins* used in very broad sense in regulatory process.
- Difficulty in self-consistent data for margin impact.
- Some success for Hatch. Generally reduced margins indicted for various components for both power uprates and plant life extension.
- License Amendment-SARs and NRC-SERs do not appear to be of sufficient detail or consistency for in-depth/quantitative assessment of margin impact for multiple licensing actions (glean bits & pieces).

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Observations/Suggestions for Power Uprates

- NRC Uprate Review: Centers on assessment that current regulatory requirements are satisfied (minimal or no requirements for risk impact, margin reductions, impact of multiple licensing actions/synergies)
- Endorse Prior Recommendations for Uprate Standard Review Plan
  - 1996 Main Yankee Lesson Learned Report (Cota, Cubbage, Dorman---NRR)
  - 1996 Power Uprate Review (Scientech)
  - 1999 Review of Operational Events for Power Uprates (ACRS Fellow Report)
- Uprate-SRP Should Include Following:
  - Standardized listing of all systems, structures, & components (SSC) subject to uprate review
  - Assessment of impact of uprate on SSC margins for both operational and DBA conditions (clear definition of methods/criteria)

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Power Uprate/Suggestions (cont'd)

- Recommend Legacy Tables (Original FSAR→ Latest Uprate)
  - Standardized Table of plant conditions impacted by power uprates, and other license amendments (LA) impacting same plant conditions
  - Standardized Table of DBA predicted loads on SSC impacted by power uprate, and other LA impacting same loads
  - Standardized Table of margins impacted by power uprates, and other LA impacting same margins
- Risk assessment requirements for significant power uprates
- Bottom Line: A clear understanding of safety & margin implications of <u>significant power uprates</u> to an <u>aging fleet</u> of plants

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United States
Nuclear Regulatory Commission

SOUTH TEXAS PROJECT REQUESTED EXEMPTIONS FROM SPECIAL TREATMENT REQUIREMENTS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

STAFF FINDINGS

Gary M. Holahan
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Office of Nuclear Reactor Regulation (NRR)

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Senior Project Manager, STP Units 1 and 2
Division of Licensing Project Management, NRR

June 6, 2001

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SOUTH TEXAS PROJECT RISK INFORMED EXEMPTION REQUEST TIMELINE

- *7/13/99* *Exemption Request Submitted*
- *1/18/00* *Request for Additional Information Issued*
- *8/31/00* *Revised STP Exemption Request Submitted*
- *11/15/00* *Draft Safety Evaluation Issued*
- *12/7/00* *ACRS Briefing on Draft Safety Evaluation*
- *1/24/01* *Response to Draft SE Open Items Submitted*
- *2/21/01* *ACRS Subcommittee Meeting on Categorization*
- *4/6/01* *ACRS Committee Meeting on Treatment*
- *4/24/01* *RILP Meeting with STPNOC on FSAR Content*
- *5/18/01* *Open Items from Draft SE resolved*
- *6/4/01* *Preliminary Final Safety Evaluation Provided to EDO*
- **6/6/01** **ACRS Committee Meeting on Safety Evaluation**
- **6/11/01** **Commission Paper Due to the Commission**
- **7/20/01** **Commission Briefing**
- **8/3/01** **Issue Final SE and Exemptions**

STAFF CONCLUSIONS

CATEGORIZATION:

The categorization process is acceptable to categorize the risk significance of both functions and SSCs for use in reducing the scope of SSCs subject to special treatment and for defining those SSCs for which exemptions from the special treatment requirements can be granted.

TREATMENT:

The alternative treatment program includes the necessary elements that, <u>if effectively implemented by the licensee</u>, can result in the safety-related LSS and NRS SSCs remaining capable of performing their safety functions under design basis conditions.

STAFF FINDINGS

- ★ Relaxing the Special Treatment Requirements on LSS and NRS Safety-Related SSCs Poses No Undue Risk to Public Health and Safety.
- ★ The Categorization Process Is a Material Circumstance Not Considered When the Special Treatment Regulations Were Adopted.
- ★ It is in the Public Interest to Grant the Exemptions.
- ★ Proposed FSAR Section is a Sufficient Basis to Grant Exemptions.

EXEMPTIONS TO BE GRANTED

★10 CFR 21.3 - Definition of Basic Component
★10 CFR 50.34(b)(10) and (11) - Related to 10 CFR Part 100, App. A
★10 CFR 50.49(b) - Scope of Electrical Equipment Important to Safety
★10 CFR 50.55a(f) - ASME Inservice Testing
★10 CFR 50.55a(g) - ASME Repair/Replacement & Inspection
★10 CFR 50.55a(h) - IEEE 279 Quality & Qualification Requirements
★10 CFR 50.59 - Changes, Tests, & Experiments
★10 CFR 50.65(b) - Scope of Maintenance Rule
★10 CFR Part 50, App. B - Quality Assurance Criteria
★10 CFR Part 50, App. J - Type C Containment Leak Testing
★10 CFR Part 100, App. A, VI, (a)(1) & (2) - SSE and OBE

EXEMPTIONS NOT GRANTED

Licensee Proposed Approach Meets Regulations

- ♦ GDC 1 - Quality Standards and Records
- ♦ GDC 2 - Protection Against Natural Phenomena
- ♦ GDC 4 - Environmental and Dynamic Effects
- ♦ GDC 18 - Inspect/Test Electrical Power Systems

Update to QA Program Required to Reflect Changes to Commitments

- 10 CFR 50.34(b)(6)(ii) - App. B Information Included in FSAR
- 10 CFR 50.54(a)(3) - Changes to QA Program

General Design Criteria

Presentation to the
Advisory Committee on Reactor Safeguards

June 6, 2001

J. N. Sorensen

Viewgraphs - Rev. 2, 5/7/01

Review of General Design Criteria

- Are they risk informed as written?
- Are they an impediment to risk-informing Part 50?
- How they can be made risk informed?
- Do they apply to new reactor types?

Structure

55 individual criteria in six groups:

- Overall Requirements (1-5)
- Protection by Multiple Fission Product Barriers (10-19)
- Protection and Reactivity Control Systems (20-29)
- Fluid Systems (30-46)
- Reactor Containment (50-57)
- Fuel and Radioactivity Control (60-64)

Scope, content, and specificity vary widely from one criterion to the next

History & Content

GDC were published as Appendix A to Part 50 in February 1971

They reflect state of the art in reactor design at that time

They reflect phenomenology important for LWR safety.

They contain four or five different kinds of information

Contents

- Conservatism
- Completeness
- Capability
- Reliability
- Defense in depth

Introduction to Appendix A

- Application for CP must include principle design criteria
- Criteria establish requirements for structures, systems & components (SSC) "important to safety"
- "Important to safety" means SSCs that provide reasonable assurance of no undue risk
- GDC establish minimum requirements for water cooled nuclear plants
- Considered to be "generally applicable" to other types, can provide guidance in establishing criteria

Additional Remarks

Development of GDC is not yet complete.

Need to consider, for example:

- single failures of passive components
- redundancy and diversity requirements for fluid systems
- type, size and orientation of primary system breaks
- systematic, non-random, concurrent failures of redundant elements

Exceptions

GDC may not be sufficient for some plants

Some GDC may not be necessary for some plants.

Over half of currently operating plants received construction permits before GDC were promulgated.

Definitions

Loss of coolant accidents

Single failure

Anticipated Operational Occurrences.

Risk informing the GDC may have limited benefit.

- Past studies have judged them to be important to safety and not a significant burden
- There are 160 Division 1 Regulatory Guides, 129 of them supporting one or more GDCs
- Part 50 incorporates the ASME BPVC (III & XI) and several IEEE standards by reference
- Risk informing GDC may only provide benefits in conjunction with other changes to Part 50.

Options for risk informing the GDC

- Modify scope
- Modify individual requirements
- Replace GDC with safety goals or risk acceptance criteria

Alternate ways of classifying the GDC

In terms of scope

13 have a scope defined as "important to safety"

In terms of single failure requirements

9 require safety function success assuming a single failure

In terms of revision to be "risk informed":

13 could be changed from "important to safety" to "important to risk"

30 require no change (4 of these are not intended to address risk)

19 could be recast in risk terms.

In terms of applicability to non-LWRs:

36 are probably applicable to all reactor types

19 are probably not applicable to some reactor types

Appendix B
Quality Assurance Criteria

18 criteria

Published in 1970

Implied standard is "no undue risk."

Scope is activities affecting quality ... consistent with importance to safety

Appendix A to Part 50 -- General Design Criteria for Nuclear Power Plants

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Introduction

Pursuant to the provisions of §50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and

components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26, and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

Definitions and Explanations

Nuclear power unit. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.(1)

Single failure. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.(2)

Anticipated operational occurrences. Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

Criteria

I. Overall Requirements

Criterion 1 -- Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Criterion 2 -- Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have

been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Criterion 3 -- Fire protection. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Criterion 4 -- Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Criterion 5 -- Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

II. Protection by Multiple Fission Product Barriers

Criterion 10 -- Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11 -- Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 12 -- Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Criterion 13 -- Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Criterion 14 -- Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 15 -- Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 16 -- Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 17 -- Electric power systems. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Criterion 18 -- Inspection and testing of electric power systems. Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Criterion 19 -- Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

III. Protection and Reactivity Control Systems

Criterion 20 -- Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21 -- Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22 -- Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 23 -- Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Criterion 24 -- Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Criterion 25 -- Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Criterion 26 -- Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 27 -- Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28 -- Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support

structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29 -- Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30 -- Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31 -- Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Criterion 32 -- Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33 -- Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34 -- Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 35 -- Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 36 -- Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37 -- Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38 -- Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 39 -- Inspection of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40 -- Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41 -- Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Criterion 42 -- Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43 -- Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44 -- Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation

(assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 45 -- Inspection of cooling water system. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Criterion 46 -- Testing of cooling water system. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. Reactor Containment

Criterion 50 -- Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by §50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Criterion 51 -- Fracture prevention of containment pressure boundary. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

Criterion 52 -- Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Criterion 53 -- Provisions for containment testing and inspection. The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment

design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Criterion 54 -- Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55 -- Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Criterion 56 -- Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Criterion 57 -- Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. Fuel and Radioactivity Control

Criterion 60 -- Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 61 -- Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 62 -- Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Criterion 63 -- Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Criterion 64 -- Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

[36 FR 3256, Feb. 20, 1971, as amended at 36 FR 12733, July 7, 1971; 41 FR 6258, Feb. 12, 1976; 43 FR 50163, Oct. 27, 1978; 51 FR 12505, Apr. 11, 1986; 52 FR 41294, Oct. 27, 1987]

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1 Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

2 Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

Appendix B to Part 50 -- Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

Introduction. Every applicant for a construction permit is required by the provisions of §50.34 to include in its preliminary safety analysis report a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Every applicant for an operating license is required to include, in its final safety analysis report, information pertaining to the managerial and administrative controls to be used to assure safe operation. Nuclear power plants and fuel reprocessing plants include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

As used in this appendix, "quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

I. Organization

The applicant(1) shall be responsible for the establishment and execution of the quality assurance program. The applicant may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, or any part thereof, but shall retain responsibility therefor. The authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components shall be clearly established and delineated in writing. These activities include both the performing functions of attaining quality objectives and the quality assurance functions. The quality assurance functions are those of (a) assuring that an appropriate quality assurance program is established and effectively executed and (b) verifying, such as by checking, auditing, and inspection, that activities affecting the safety-related functions have been correctly performed. The persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided. Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the quality assurance program may take various forms provided that the persons and organizations assigned the quality assurance functions have this required authority and

organizational freedom. Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the quality assurance program at any location where activities subject to this appendix are being performed shall have direct access to such levels of management as may be necessary to perform this function.

II. Quality Assurance Program

The applicant shall establish at the earliest practicable time, consistent with the schedule for accomplishing the activities, a quality assurance program which complies with the requirements of this appendix. This program shall be documented by written policies, procedures, or instructions and shall be carried out throughout plant life in accordance with those policies, procedures, or instructions. The applicant shall identify the structures, systems, and components to be covered by the quality assurance program and the major organizations participating in the program, together with the designated functions of these organizations. The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety. Activities affecting quality shall be accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanness; and assurance that all prerequisites for the given activity have been satisfied. The program shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality by inspection and test. The program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained. The applicant shall regularly review the status and adequacy of the quality assurance program. Management of other organizations participating in the quality assurance program shall regularly review the status and adequacy of that part of the quality assurance program which they are executing.

III. Design Control

Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. The verifying or checking process

shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions. Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization.

IV. Procurement Document Control

Measures shall be established to assure that applicable regulatory requirements, design bases, and other requirements which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, and services, whether purchased by the applicant or by its contractors or subcontractors. To the extent necessary, procurement documents shall require contractors or subcontractors to provide a quality assurance program consistent with the pertinent provisions of this appendix.

V. Instructions, Procedures, and Drawings

Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

VI. Document Control

Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization.

VII. Control of Purchased Material, Equipment, and Services

Measures shall be established to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures shall include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products

upon delivery. Documentary evidence that material and equipment conform to the procurement requirements shall be available at the nuclear powerplant or fuel reprocessing plant site prior to installation or use of such material and equipment. This documentary evidence shall be retained at the nuclear powerplant or fuel reprocessing plant site and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment. The effectiveness of the control of quality by contractors and subcontractors shall be assessed by the applicant or designee at intervals consistent with the importance, complexity, and quantity of the product or services.

VIII. Identification and Control of Materials, Parts, and Components

Measures shall be established for the identification and control of materials, parts, and components, including partially fabricated assemblies. These measures shall assure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, erection, installation, and use of the item. These identification and control measures shall be designed to prevent the use of incorrect or defective material, parts, and components.

IX. Control of Special Processes

Measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

X. Inspection

A program for inspection of activities affecting quality shall be established and executed by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. Such inspection shall be performed by individuals other than those who performed the activity being inspected. Examinations, measurements, or tests of material or products processed shall be performed for each work operation where necessary to assure quality. If inspection of processed material or products is impossible or disadvantageous, indirect control by monitoring processing methods, equipment, and personnel shall be provided. Both inspection and process monitoring shall be provided when control is inadequate without both. If mandatory inspection hold points, which require witnessing or inspecting by the applicant's designated representative and beyond which work shall not proceed without the consent of its designated representative are required, the specific hold points shall be indicated in appropriate documents.

XI. Test Control

A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during

nuclear power plant or fuel reprocessing plant operation, of structures, systems, and components. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results shall be documented and evaluated to assure that test requirements have been satisfied.

XII. Control of Measuring and Test Equipment

Measures shall be established to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits.

XIII. Handling, Storage and Shipping

Measures shall be established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, specific moisture content levels, and temperature levels, shall be specified and provided.

XIV. Inspection, Test, and Operating Status

Measures shall be established to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the nuclear power plant or fuel reprocessing plant. These measures shall provide for the identification of items which have satisfactorily passed required inspections and tests, where necessary to preclude inadvertent bypassing of such inspections and tests. Measures shall also be established for indicating the operating status of structures, systems, and components of the nuclear power plant or fuel reprocessing plant, such as by tagging valves and switches, to prevent inadvertent operation.

XV. Nonconforming Materials, Parts, or Components

Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. These measures shall include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items shall be reviewed and accepted, rejected, repaired or reworked in accordance with documented procedures.

XVI. Corrective Action

Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause

of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

XVII. Quality Assurance Records

Sufficient records shall be maintained to furnish evidence of activities affecting quality. The records shall include at least the following: Operating logs and the results of reviews, inspections, tests, audits, monitoring of work performance, and materials analyses. The records shall also include closely-related data such as qualifications of personnel, procedures, and equipment. Inspection and test records shall, as a minimum, identify the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any deficiencies noted. Records shall be identifiable and retrievable. Consistent with applicable regulatory requirements, the applicant shall establish requirements concerning record retention, such as duration, location, and assigned responsibility.

XVIII. Audits

A comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits shall be performed in accordance with the written procedures or check lists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audit results shall be documented and reviewed by management having responsibility in the area audited. Followup action, including reaudit of deficient areas, shall be taken where indicated.

[35 FR 10499, June 27, 1970, as amended at 36 FR 18301, Sept. 11, 1971; 40 FR 3210D, Jan. 20, 1975]

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1 While the term "applicant" is used in these criteria, the requirements are, of course, applicable after such a person has received a license to construct and operate a nuclear powerplant or a fuel reprocessing plant. These criteria will also be used for guidance in evaluating the adequacy of quality assurance programs in use by holders of construction permits and operating licenses.

Preliminary Categorization of General Design Criteria
DRAFT (a work in progress)

		Attribute	Requires Function	Requires Margin	Req. Test	Safety Function	"Safety Goal"	Corner Stone	Mod. Scope	Recast in Risk Terms?	Apply to non-LWRs?
	Notes	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)
	Appendix A: General Design Criteria										
I	Overall Requirements										
1	Quality Standards and Records	Reliability				all	all	all	1	Scope	yes
2	Design Bases for Protection Against Natural Phenomena	Capability				all	all	all	1	Scope	yes
3	Fire Protection	Capability	yes			all	all	all	1	Scope	yes
4	Environmental and Dynamic Effects Design Bases	Capability				all	all	all	1,2	Scope	yes
5	Sharing of Structures, Systems and Components	Reliability				all	all	all	1	Recast	yes
II	Protection by Multiple Fission Product Barriers										
10	Reactor Design	Conservatism		margin		CPR	CD	IE		No change	yes
11	Reactor Inherent Protection	Conservatism		stability		CPR	CD	IE		No change	yes
12	Suppression of Reactor Power Oscillations	Capability		stability		CPR	CD	IE		No change	yes
13	Instrumentation and Control	Capability	yes			all	all	IE	1	Scope	yes, but
14	Reactor Coolant Pressure Boundary	Conservatism		margin	yes	HR	CD	IE, BI		No change	yes
15	Reactor Coolant System Design	Capability		margin		HR	CD	IE, BI		No change	yes
16	Containment Design	Capability, DiD	yes			FPC	LR	MS, BI	1	Recast	maybe
17	Electric Power Systems	Reliability	yes		yes	all	all	IE, MS	1,3	Recast (SFC)	yes
18	Inspection and Testing of Electric Power Systems	Reliability			yes	all	all	IE, MS	1	Recast	yes
19	Control Room	Capability	yes			all	LR	MS, BI	2	Recast(?)	yes, but
III	Protection and Reactivity Control Systems										
20	Protection System Functions	Capability	yes			CPR	CD	IE	1	No change	yes
21	Protection System Reliability and Testability	Reliability			yes	CPR	CD	IE		Recast (SFC)	yes
22	Protection System Independence	Reliability		margin		CPR	CD	IE		No change	yes
23	Protection System Failure Modes	Reliability		margin		CPR	CD	IE		No change	yes
24	Separation of Protection and Control Systems	Reliability				CPR	CD	IE		Recast (SFC)	yes
25	Protection System Requirements for Reactivity Control Malfunctions	Conservatism	yes			CPR	CD	IE		Recast (SFC)	yes, but
26	Reactivity Control System Redundancy and Capability	Capability				CPR	CD	IE		Recast	no
27	Combined Reactivity Control Systems Capability	Capability		margin		CPR	CD	IE		No change	no
28	Reactivity Limits	Conservatism		margin		CPR	CD	IE		No change	yes, but
29	Protection Against Anticipated Operational Occurrences	Conservatism		margin		CPR	CD	IE		No change	yes
IV	Fluid Systems										
30	Quality of Reactor Coolant Pressure Boundary	Reliability			yes	RCI	CD	BI	4	No change	yes, but
31	Fracture Prevention of Reactor Coolant Pressure Boundary	Capability		margin		RCI	CD	BI		No change	yes, but
32	Inspection of Reactor Coolant Pressure Boundary	Reliability			yes	RCI	CD	BI		No change	yes, but
33	Reactor Coolant Makeup	Capability	yes			RCI	CD	IE		Recast(?)	yes, but
34	Residual Heat Removal	Capability	yes			HR	CD	IE		Invokes SFC	yes, but
35	Emergency Core Cooling	Capability, DiD	yes			HR	CD	MS	2	Invokes SFC	no
36	Inspection of Emergency Core Cooling System	Reliability			yes	HR	CD	MS		No change	no
37	Testing of Emergency Core Cooling System	Reliability			yes	HR	CD	MS		No change	no
38	Containment Heat Removal	Capability, DiD	yes			FPC	LR	MS, BI	2	Recast (SFC)	no
39	Inspection of Containment Heat Removal System	Reliability			yes	HR	LR	MS, BI		No change	no
40	Testing of Containment Heat Removal System	Reliability			yes	HR	LR	MS, BI		No change	no

Preliminary Categoriz[illegible] of General Design Criteria
DRAFT (a work in progress)

		Attribute	Requires Function	Requires Margin	Req. Test	Safety Function	"Safety Goal"	Corner Stone	Mod. Scope	Recast in Risk Terms?	Apply to non-LWRs?
	Notes	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)
41	Containment Atmosphere Cleanup	Capability, DiD	yes			FPC	LR	MS		Recast (SFC)	no
42	Inspection of Containment Atmosphere Cleanup Systems	Reliability			yes	FPC	LR	MS		No change	no
43	Testing of Containment Atmosphere Cleanup Systems	Reliability			yes	FPC	LR	MS		No change	no
44	Cooling Water	Capability	yes			HR	CD	MS	1	Invokes SFC	yes
45	Inspection of Cooling Water System	Reliability			yes	HR	CD	MS		No change	yes
46	Testing of Cooling Water System	Reliability			yes	HR	CD	MS	2	No change	yes
V	Reactor Containment										
50	Containment Design Basis	Capability, DiD	yes			FPC	LR	BI	2	No change	no
51	Fracture Prevention of Containment Pressure Boundary	Reliability		margin		FPC	LR	BI		No change	no
52	Capability for Containment Leakage Rate Testing	Reliability			yes	FPC	LR	BI		Recast(?)	no
53	Provisions for Containment Testing and Inspection	Reliability			yes	FPC	LR	BI		No change	no
54	Systems Penetrating Containment	Reliability	yes		yes	FPC	LR	BI	1	Recast	no
55	Reactor Coolant Pressure Boundary Penetrating Containment	Reliability	yes			FPC	LR	BI		Recast	no
56	Primary Containment Isolation	Reliability	yes			FPC	LR	BI		Recast	no
57	Closed Systems Isolation Valves	Reliability	yes			FPC	LR	BI		Recast	no
VI	Fuel and Radioactivity Control										
60	Control of Releases of Radioactive Materials to the Environment	Capability	yes			FPC	SR	BI		No change	yes
61	Fuel Storage and Handling and Radioactivity Control	Capability	yes		yes	FPC	SR	BI	1	No change	yes
62	Prevention of Criticality in Fuel Storage and Handling	Capability	yes			FPC	SR	BI		No change	yes
63	Monitoring Fuel and Waste Storage	Capability	yes			FPC	SR	BI		No change	yes
64	Monitoring Radioactivity Releases	Capability	yes			FPC	LR, SR	BI, EP		No change	yes

Notes

(1) This column identifies the design attribute(s) each criterion addresses. DiD denotes defense in depth.
(2) Requires provision of a system or function
(3) Requires design conservatism (stability or margin)
(4) Requires inspection, testing, or provision of capability to do so
(5) This column identifies the primary safety function addressed by each criterion: heat removal (HR), control of power or reactivity (CPR), fission product containment (FPC), control of reactor coolant inventory (RCI) or all of the above
(6) This column identifies the primary focus of each criterion: core damage (CD), large releases (LR), small releases (SR), or all of the above.
(7) Associates each criterion with the reactor safety cornerstones: initiating events (IE), barrier integrity (BI), mitigating systems (MS), emergency planning (EP) or all of the above
(8) 1 - scope is "important to safety" or addresses "adequate safety"
2 - references LOCA, defined as including double ended break of largest pipe
3 - requires electric power to be available "within a few seconds" following LOCA
4 - could be modified to require testing commensurate with importance to risk
(9) This column identifies whether or not a criterion can be usefully cast in risk terms
"Scope" means that the scope could be changed from "important to safety" to "important to risk"
"Invokes SFC" means that the criterion requires safety function success assuming a single failure
"Recast (SFC)" means that the criterion requires safety function success assuming a single failure, and should also be recast for other reasons
(10) This column identifies whether or not a criterion applies (unambiguously) to non-LWRs.

(7)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
483	10	10.1

Title **PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES**

Authors
JOHN T. LARKINS/SAM DURAISWAMY

List of Documents Attached

10

Instructions to Preparer	From Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	JOHN T. LARKINS/ SAM DURAISWAMY

G:PlanPro:ppmins.483
June 7, 2001

MINUTES OF THE
PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
WEDNESDAY, JUNE 6, 2001

The ACRS Subcommittee on Planning and Procedures held a meeting Wednesday, June 6, 2001, in Room 2-B1, Two White Flint North Building, Rockville, Maryland. The purpose of this meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 12:20 p.m. and adjourned at 1:20 p.m.

ATTENDEES

G. Apostolakis, Chairman
M. Bonaca
T. Kress

ACRS STAFF

J. T. Larkins
J. Lyons
S. Duraiswamy
R. P. Savio
C. Harris
J. Gallo

NRC STAFF

S. Bahadur

DISCUSSION

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

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

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the June 2001 ACRS meeting be as shown in the attached list.

2) Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through September 2001 is attached (pp. 6-11). The objectives are to:

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

During this session, the Subcommittee discussed and developed recommendations on the items that require Committee decision, which are included in Section II of the Future Activities list (pp. 12-14).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload, especially on the reassignments of the items for the September ACRS meeting. Changes will be made, as appropriate. The Committee needs to consider the Subcommittee's recommendations on items noted in Section II of the Future Activities list.

3) Quadripartite Meeting Update

During the April meeting, the Committee was informed that recently Mr. Lothar Hahn, Chairman of the RSK, told us that preparations are continuing for Germany to host the next Quadripartite meeting, possibly later this year. The French GPR have confirmed their participation and the RSK is currently working to confirm the participation of the Japanese NSC.

Subsequent to the April meeting, Mrs. Waldorf from GRS in Germany said that the RSK planned to meet on May 3, 2001 to discuss among other topics the next Quadripartite meeting. Mr. Lathar Hahn would be attending this meeting. According to Mrs. Waldorf, consideration is being given to hold the next Quadripartite meeting in November 2001.

During the May 2001 meeting, the Committee proposed November 13-15, 2001 as possible dates for this meeting and suggested that the ACRS Executive Director get feedback from RSK on these dates. Also, the Committee suggested that the members propose topics for this meeting.

The ACRS Executive Director spoke with Mr. Kersting, Vice-Chairman of the RSK, about a possible meeting in November and a format for the meeting. During that discussion, Mr. Kersting noted that the RSK will probably propose that the meeting be held in May of 2002 because there was not enough time to plan and orchestrate a meeting in November 2001.

Dr. Larkins volunteered to help the RSK develop the format for the next Quadripartite meeting. An issue which needs to be resolved is the participation of the Japanese Advisory Committee, the NSC, from whom we have not had any response as to whether they are willing or available to participate in the next Quadripartite meeting.

During the May meeting, the Committee suggested that the members propose topics for this meeting. Topics proposed by ACRS Members Dana Powers, Graham Wallis, George Apostolakis, Tom Kress, and Peter Ford are as follows (Attachment, pp. 15-17).

- Risk-informed regulation
- Thermal hydraulic analysis and code issues
- Revised reactor oversight process
- High burnup fuel
- Fire protection and insights from the IPEEE process
- Risk analysis of spent fuel storage
- Quality of data and the predictive capabilities for environmentally assisted cracking
- Defense-in-depth needs and the safety criteria for future reactors
- Human performance and safety culture
- Level of inherent safety needed to eliminate the need for emergency response and evacuation

RECOMMENDATION

The Subcommittee recommends the following four topics for the Quadripartite meeting:

- Risk-informed regulation
- Thermal-hydraulic analysis and code issues
- High burnup fuel
- Risk analysis of spent fuel storage

The Subcommittee agrees with the proposal by the ACRS Executive Director that a few breakout sessions be planned to discuss some other topics proposed by the members. The Committee should decide which topics should be discussed at the breakout sessions.

In addition, the Subcommittee recommends that the ACRS Executive Director, in consultation with RSK, establish dates and month for this meeting in 2002, and also obtain feedback on the proposed list of topics.

4) Annual Report to the Commission on the NRC Safety Research Program

Advance copy of the ACRS report for 2001 has been provided to the Commissioners and is expected to be published as NUREG-1635, Vol. 4 in the near future. The 2001 ACRS report is currently available on the ACRS web site.

The 2002 report is due to the Commission in March 2002. During previous meetings, the Committee discussed briefly whether to write a comprehensive report every year,

but did not make a decision. In view of the fact that the NRC research programs may not change significantly within a year, a comprehensive report may not be needed each year.

RECOMMENDATION

The Subcommittee recommends the following:

- Dr. Bonaca should take the lead in preparing the 2002 report
- The Committee should decide whether to prepare a comprehensive report each year. If the answer is no, the ACRS Chairman should discuss this matter with the Commissioners and obtain their feedback.
- The Committee should decide on the scope, format, and content of the 2002 report.

5) RSK Workshop on Risk Informed Decisionmaking in Nuclear Safety

During the May 2001 ACRS meeting, the Committee was informed about the RSK Workshop on Risk-Informed Decisionmaking in Nuclear Safety that was scheduled to be held on June 11-12, 2001, in Germany. The Committee asked Dr. Larkins to inform Dr. Lothar Hahn, RSK, that even though the ACRS has an interest in the topics proposed for discussion at this Workshop, it would be difficult for the ACRS members to attend this Workshop because of the lateness of the invitation.

Subsequently, in a letter to Dr. Larkins dated May 28, 2001 (Attachment pp. 18-21), Dr. Lothar Hahn states that since several anticipated participants have difficulty attending the Workshop on June 11-12, he has postponed the Workshop. Now, he plans to hold the Workshop for two days (to be selected) between October 15 and November 2, 2001 and seeking feedback from anticipated participants.

Dr. Apostolakis proposes that the Workshop be held any two consecutive days between October 29 and November 2.

RECOMMENDATION

The Subcommittee recommends that Drs. Apostolakis and Bonaca attend this workshop. Other members who are interested in attending this Workshop should inform the Committee. In consultation with the ACRS Executive Director on the availability of resources, the Committee should decide how many members can attend this Workshop.

6) Travel Request

Dr. Apostolakis has been invited to participate as a speaker at the Thirteenth Annual Procedure Symposium organized by Scientech (pp. 22-24), which will be held between September 11 and 14, 2001, at the Crown Reef Resort in Myrtle Beach, South Carolina. He requests Committee approval to attend this symposium.

RECOMMENDATION

The Subcommittee recommends that the Committee approve the travel request by Dr. Apostolakis.

7) Items of Interest

- We issued the ACRS/ACNW self-assessment paper (SECY-01-0092) on May 21, 2001, to the Commission satisfying a Commission milestone. We are currently in the process of revising the Operating Plan, which will be issued by the end of June 2001.

- The ACRS Action Plan has been placed on our web site and is expected to be published as a brochure within the next week or two.

- In response to an inquiry from former ACRS Member Hal Lewis regarding an ACRS reunion for the Committee's 50th Anniversary, the ACRS Executive Director forwarded an e-mail to Chairman Meserve that the Committee has had some preliminary discussions, but has not planned anything specific at this time (pp. 25-26).

ANTICIPATED WORKLOAD
June 6-8, 2001

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	Bonaca	Larkins	Meeting with Commissioner Dicus	--	P&P 6/4	AR 6/4-5
		Markley	Risk-Based Performance Indicators **[PRESENTATION COMPLETED AT THE MAY MEETING]**	Report		
		Duraiswamy/Sorensen	Discussion General Design Criteria **[PRESENTATION BY SORENSEN]**	--		
Bonaca	--	Duraiswamy/ Elliott	Need for Revising 10 CFR Part 54-The License Renewal Rule **[Committee Discussion]**	--	--	P&P 6/4 AR 6/4-5
Kress	Apostolakis	Markley/El-Zeftawy	Regulatory Challenges for future Nuclear Plants **[Committee Discussion]**	--	AR 6/4-5	P&P 6/4
Powers	--	Duraiswamy	Response to Chairman's May 7, 2001 memo on the Steam Generator DPO	Report	--	AR 6/4-5
Shack	Wallis	Markley/Elliott	Proposed Risk-Informed Revisions to 10 CFR 50.46 and Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 **[Status Report]**	--	--	AR 6/4-5
Sieber	Apostolakis	Weston	South Texas Project Exemption Request	Report to be completed in July	--	AR 6/4-5
		Singh/Weston	Prearrangement Briefing to ACRS for Waterford & and Region IV site visit.	--	--	--

ANTICIPATED WORKLOAD
June 6-8, 2001
(CONTINUED)

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Wallis	Bonaca	Boehnert/ Cronenberg	Potential Margin Reductions Associated with Power Uprates **[PRESENTATION BY Cronenberg]**	--	--	AR 6/4-5

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ANTICIPATED WORKLOAD
July 11-13, 2001

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Singh	Draft IPEEE Insights Report	Report	RPRA 6/22 (a.m.) P&P 7/9 (p.m.)	PLR 6/22 (p.m.) PO/FP 6/27-28 M&M/THP/RPRA 7/9 PO 7/10
Bonaca	Ford	Duraiswamy/Dudley	Need for Revising 10 CFR Part 54-The License Renewal Rule	Report	PLR 6/22 (p.m.)	RPRA 6/22 (a.m.) M&M/THP/RPRA 7/9 P&P 7/9 (p.m.) PO 7/10
Kress	--	El-Zeftawy	Spent Fuel Accident Risk at Decommissioning Plants	Report	--	THP 6/12 RPRA 6/22(a.m.) PLR 6/22 (p.m.) M&M/THP/RPRA 7/9 PO 7/10 P&P 7/9 (p.m.)
Leitch	Bonaca	Singh	Proposed Resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance **[STATUS REPORT]**	--		PLR 6/22 PO/FP 6/27-28 PO 7/10

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ANTICIPATED WORKLOAD
July 11-13, 2001 (CONTINUED)

Sieber	Apostolakis	Weston	South Texas Project Exemption Request Reactor Oversight Process **[Committee Discussion]**	Report Draft Report	PO/FP 6/27-28 PO 7/10	--
Shack	-- Sieber	Markley Boehnert	Proposed Risk-Informed Revisions to 10 CFR 50.46 and Revisions to the Framework for risk-informing the Technical Requirements of 10 CFR Part 50 Staff and Industry proposals for dealing with Control Rod Drive Mechanism Cracking	Report Report	M&M/THP/RPRA 7/9	THP 6/12 RPRA 6/22 PLR 6/22 PO/FP 6/27-28
Wallis	Bonaca	Boehnert/Cronenberg	Potential Margin Reductions Associated with Power Uprates **[TENTATIVE]**	--	THP 6/12	M&M/THP/RPRA 7/9 PO 7/10

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ANTICIPATED WORKLOAD
SEPTEMBER 6-8, 2001

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Larkins	Meeting with Commissioner Merrifield		P&P 9/5	M&M 9/4
		Markley	Risk-Informed Regulation Implementation Plan	Report (Tent.)		
		Markley	Proposed Revision to Reg. Guide 1.174 to provide acceptance criteria guidelines for PRA quality	Report		
Bonaca	--	Duraiswamy	Proposed Update to 10 CFR Part 52	--	--	M&M 9/4 P&P 9/5
		Singh	Proposed Resolution of GSI-191, Assessment of Debris Accumulation on PWR Sump Performance	Report		THP 7/17-18 THP 8/21-23
Kress	--	Dudley	Proposed Regulatory Guide on Control Room Habitability	Report	THP 8/21-23	THP 7/17-18 M&M 9/4 P&P 9/5
		El-Zeftawy	EPRI Report on Waterhammer Issues	Report	--	

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ANTICIPATED WORKLOAD
SEPTEMBER 6-8, 2001 (CONTINUED)

Powers	--	Dudley Boehnert	Proposed Revision 1 to Reg. Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" Duane Arnold Core Power Uprate	Report Report	--	THP 7/17-18 THP 8/21-23
Sieber	Apostolakis	Weston	Reactor Oversight Process	Report	--	--
Wallis		Boehnert Boehnert/Duraiswamy	TRACG Best-Estimate Thermal-Hydraulic Code RETRAN-3D Transient Analysis Code	Report Report	THP 7/17-18	M&M 9/4

II. ITEMS REQUIRING COMMITTEE ACTION

1. Oconee Nuclear Station Unit 3 Reactor Pressure Vessel Head Leakage (Open) (WJS/JDS/MWW) ESTIMATED TIME: 1 ½ hours

Purpose: Decide on a Course of Action

Review requested by the Staff. On February, 18, 2001, Oconee Nuclear Station, Unit 3, found evidence of small accumulations of boric acid deposits at the base of several control rod drive mechanisms (CRDMs) on the top surface of the Reactor Pressure Vessel (RPV) head. The boric acid deposits were identified around nine of the sixty-nine CRDM nozzles. The amount of boric acid around each nozzle signified that reactor coolant system pressure boundary leakage had occurred. The apparent root cause of the nine CRDM nozzle leaks is primary water stress corrosion cracking (PWSCC). The leaking CRDMs have been repaired.

The staff sent a letter requesting justification for continued operation. During the May 2001 Committee meeting, the Committee decided not to hear a presentation at this time because of the ongoing effort with this event, but to consider hearing a briefing on this matter after the staff has received the justification for continued operation.

Recently, Dr. Sheron, NRR, informed Dr. Larkins that the staff is in the process of developing options for dealing with this issue and also reviewing the industry proposals. There appears to be some differing views between the industry and the staff on measures to be taken to preclude recurrence of this problem and the urgency of requiring plants to perform inspections. Dr. Sheron suggested that the ACRS review the staff's and industry's proposals and provide its views at the July ACRS meeting. It would be helpful if this issue could be discussed at a Subcommittee meeting prior to the full Committee discussion. One possibility is to discuss this on July 9, 2001.

The Planning and Procedures Subcommittee recommends that this item be scheduled for the July ACRS meeting.

2. Revision 3 to Regulatory Guide 1.52 (DG-1102) "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (Open) (TSK/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Decide on a Course of Action

Review requested by the NRC staff [J. Segala, NRR]. The staff has provided the Committee with a copy of the draft final version Regulatory Guide for review and comment. This Guide relates to GDC 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A to 10 CFR Part 50 regarding engineered-

safety feature (ESF) air filtration systems for ensuring occupancy of plant control rooms under accident conditions.

In 1999, the staff developed a proposed Generic Letter (GL-99-02) to address concerns with the laboratory testing of nuclear-grade activated charcoal for use in ESF air filtration systems. Specifically, it was found that testing of nuclear-grade activated charcoal to standards other than a 1989 ASTM Standard provided non-conservative results and did not provide assurance of compliance with the dose limits of GDC 19, "Control Room" and 10 CFR Part 100. The Committee was twice offered an opportunity to review GL-99-02 (proposed and final versions) and decided not to review it.

NRR updated RG 1.52 to make it consistent with GL99-02 and provided the draft version of the guide to the ACRS for its review. The Committee, via a July 18, 2000 memorandum from the ACRS Executive Director, stated that it had no objection to issuance of the Guide for public comment.

NRR has now revised RG 1.52 in response to the public comments received and has provided copies of the proposed final version of this Guide to the Committee for review.

The staff has also provided the Committee with a copy of a revised companion regulatory guide [Regulatory Guide 1.140 (DG-1103)] which addresses similar criteria for non-ESF filtration systems.

Dr. Kress has reviewed the proposal final RG 1.52, along with a staff Paper summarizing the significant changes made to the Guide in response to public comments as well as a summary of the comments themselves.

The Planning and Procedures Subcommittee agrees with the recommendation by Dr. Kress that the Committee not review this Guide and that a Larkinsgram be issued.

3. Proposed Revision to Regulatory Guide 1.174 to Address PRA Quality in Risk-Informed Activities (Open)(GEA/MTM) ESTIMATED TIME: 1 ½ hours

Purpose: Decide on a Course of Action

Review requested by the NRC staff [M. Drouin, RES]. During the September 2000 ACRS meeting, the Committee discussed the staff's approach for addressing the issue of quality of PRAs described in SECY-00-0162. The Committee provided a report to the Commission dated September 7, 2000. In an SRM dated October 27, 2000, the Commission directed the staff to: (1) proceed with its review of NEI 00-02, Peer Review Process Guidelines, (2) encouraged the staff to work with ASME and other stakeholders to resolve differences in the development of a consensus standard, (3) expand Attachment 2 to SECY-00-0162 to include examples of how PRA quality influences risk-informed decision-

making, and (4) increase public confidence by maximizing the information available to the public on PRAs and decision tools. These Commission directives were consistent with the recommendations provided in the Committee's report dated September 7, 2000.

The staff requests to meet with the Committee during the September 6-8 ACRS meeting, to discuss a proposed revision to Regulatory Guide 1.174 to provide acceptance criteria guidelines related to PRA quality following reconciliation of public comments.

The Planning and Procedures Subcommittee recommends that the Committee review this Guide after reconciliation of public comments.