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

September 29, 2000

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

SUBJECT: SUMMARY REPORT — 475th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AUGUST 29–SEPTEMBER 1, 2000 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

Dear Chairman Meserve:

During its 475th meeting, August 29–September 1, 2000, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the reports and letters listed below and authorized Dr. Larkins, Executive Director of the ACRS, to transmit the memoranda noted below:

REPORTS

- Assessment of the Quality of Probabilistic Risk Assessments (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated September 7, 2000)

- Causes and Significance of Design Basis Issues at U.S. Nuclear Power Plants (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated September 8, 2000)

- Proposed Risk-Informed Revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors" (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated September 13, 2000)

- Pre-Application Review of the AP1000 Standard Plant Design—Phase 1 (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated September 14, 2000)

LETTERS

- Proposed High-Level Guidelines for Performance-Based Activities (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, ACRS, dated September 8, 2000)

- Proposed Final Regulatory Guide DG-1093, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases" (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, ACRS, dated September 12, 2000)

MEMORANDA

- Final Regulatory Guide 1.18x on 10 CFR 50.59, "Changes, Tests, and Experiments" (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated September 5, 2000)

- Draft Regulatory Guide DG-1075, "Emergency Planning and Preparedness for Nuclear Power Reactors" (Proposed Revision 4 to Regulatory Guide 1.101) (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated September 7, 2000)

HIGHLIGHTS OF KEY ISSUES CONSIDERED BY THE COMMITTEE

1. Proposed Risk-Informed Revisions to 10 CFR Part 50

The Committee heard presentations by and held discussions with representatives of the NRC staff, the Nuclear Energy Institute (NEI), and Performance Technology, Inc., concerning proposed risk-informed revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," and related matters. The Committee discussed the staff's recommendations for revising 10 CFR 50.44, including the staff's proposed approach for resolving the petition for rulemaking submitted by Performance Technology, Inc. The Committee discussed the staff's draft framework document (Option 3) for risk-informed revisions to the technical requirements of 10 CFR Part 50. The Committee also discussed the staff's proposed resolution of public comments on the Advance Notice of Proposed Rulemaking for 10 CFR 50.69 and Appendix T (Option 2). These documents pertain to special treatment requirements for structures, systems, and components (SSCs).

The Committee considered NEI's views on the staff's proposed approaches to Option 3 policy and implementation issues (definition of defense in depth, use of safety goals, selective implementation, and backfit considerations) and Option 2 regulatory treatment of RISC-3 category SSCs (safety-related, low risk significant). The Committee also considered NEI's views on the proposed American Society of Mechanical Engineers document, "Standard for PRA for Nuclear Power Plant Applications," and the industry certification process described in the document NEI 00-02, "Industry PRA Peer Review Process Guidelines."

Conclusion

The Committee provided a report dated September 13, 2000, to Chairman Meserve on this matter. The Committee also decided to schedule a briefing during the October 5–7, 2000, ACRS meeting, to review NEI 00-02.

2. Causes and Significance of Design Basis Issues at U.S. Nuclear Power Plants

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning the causes and significance of Design Basis Issues (DBIs). The presentations summarized a systematic and comprehensive study of design basis issue trends and patterns. The study provides insights into reported DBIs— their causes, significant patterns both in the power reactor industry and in particular power reactor systems, frequency trends, safety consequences, and risk significance. The insights from this study are intended to assist NRC and the industry in their ongoing efforts to make NRC's regulatory framework and oversight process more risk informed and performance-based and to reduce unnecessary regulatory burden.

The study was based on information gathered from 1985 through 1997. It showed that the most common causes of DBIs were original design error, procedure deficiency, and human error and that three safety-related systems accounted for a most of the potentially risk-significant DBIs. It also showed that older plants generally reported more DBIs than newer plants and that, from 1990 to 1997, the percent of LERs on DBIs with accident sequence precursor events steadily decreased while the number of DBIs increased.

Conclusion

Although this was an information briefing, the Committee decided to send a letter to Chairman Meserve expressing the Committee's satisfaction with this ongoing analysis of experiential data.

3. Proposed Final Regulatory Guide (DG-1093) Endorsing NEI 97-04 Document on Design Basis

The Committee heard presentations by and held discussions with representatives of the NRC staff and NEI concerning a regulatory guide endorsing NEI's design basis guidelines. The term "design basis" is used in several regulations in 10 CFR Part 50. It is also useful for evaluating degraded and nonconforming conditions.

The objective of NEI 97-04 was to clarify the definition of design basis information as defined in 10 CFR 50.2.

The NEI guidance was developed as a result of system-specific engineering inspections. The inspections showed that some licensees were not maintaining design basis information as required by NRC regulations and that the staff and the industry disagreed on what the 10 CFR 50.2 definition meant. In response to the problems, licensees initiated design basis reconstitution programs. These programs sought to identify and selectively regenerate missing documentation. During the documentation effort, it became clear that definitions of what constituted design basis information differed from licensee to licensee. The lessons learned from events at Millstone and Maine Yankee showed that the definition of design basis should be clarified. NEI began developing guidance in response to this finding. After several years and many meetings between NRC staff and NEI representatives, a regulatory guide to endorse the NEI guidance was developed.

The proposed final regulatory guide endorses the NEI guidance without exception because NRC staff and NEI representatives were able to resolve differences that had previously existed.

Conclusion

The Committee sent a report dated September 12, 2000, to the Executive Director for Operations on this matter.

4. AP1000 Standard Plant Design

The Committee heard a presentation by and held discussions with the staff regarding the results of the staff's pre-application (Phase 1) review of Westinghouse Electric Company's proposed AP1000 Standard Plant Design. Westinghouse plans to seek certification of a 1000 MWe nuclear plant similar to the certified AP600 design, and seeks NRC feedback on the scope and cost of reviewing and certifying the AP1000 design. Westinghouse proposed five "assumptions" for Phase II review:

- The AP1000 design certification application (DCA) will reference sections of the AP600 Design Control Document that do not change for the AP1000.
- The AP1000 DCA will not require the applicant to do additional tests.
- The AP1000 DCA can use the AP600 analysis codes with limited modifications.
- The AP1000 DCA can use the AP600 PRA, supplemented with a sensitivity study, to meet the requirements for a plant-specific PRA.
- The AP1000 DCA can defer selected design activities to the combined license applicant.

The Committee discussed the staff's position on Westinghouse's assumptions.

Conclusion

The Committee sent a report, dated September 14, 2000, to Chairman Meserve on this matter.

5. Performance-Based Regulatory Initiatives

The Committee heard presentations by and held discussions with the NRC staff regarding a proposed Commission paper on proposed high-level guidelines for performance-based activities. The staff presented two case studies demonstrating that the guidelines are useful in evaluating the viability of a performance-based approach within the regulatory framework. The Committee and the staff discussed setting capability and performance parameters at the highest possible level of the event tree and providing explicit guidance for selecting the appropriate number of redundant or overlapping parameters.

Conclusion

The Committee sent a letter, dated September 8, 2000, to the Executive Director for Operations on this matter.

6. License Renewal Guidance Documents

With respect to license renewal, the Committee heard presentations by and held discussions with the NRC staff on the content of the proposed Standard Review Plan, the Generic Aging Lessons Learned (GALL) report, the regulatory guide, and the industry implementation. The staff summarized the contents of these documents. The Committee and the staff discussed the differences between the various drafts of these documents, the status of disposition of the concerns identified in the Union of Concerned Scientists' reports, the details in the guidance documents on the scoping and screening processes, and the disposition of license renewal generic issues.

Conclusion

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

7. Operating Events at Indian Point Nuclear Power Plant Unit 2

- Reactor Trip with Complications
- Steam Generator Tube Failure

The Committee heard presentations by and held discussions with representatives of the NRC staff on two operating events at Indian Point Unit 2 (IP2). The first event was an August 31, 1999, reactor trip with complications. The second event, a steam generator tube failure, occurred on February 15, 2000. The purpose of the presentations was to hear the findings and conclusions of the Augmented Inspection Team (AIT) about the events at IP2.

Reactor Trip with Complications

On August 31, 1999, the IP2 reactor automatically tripped from 99% power because of a spurious trip signal. The offsite power breakers also tripped unexpectedly and the diesels (EDGs) started. A short time later, the EDG output breaker tripped, leaving a vital bus deenergized. This resulted in a loss of power to vital equipment including a battery charger. The battery subsequently discharged, causing a loss of power, which eventually required the declaration of an Unusual Event. Although there was no immediate threat to public health and safety, the event was risk significant. There was no radiological release from the event.

The AIT determined that the event was preventable and was caused primarily by problems in plant configuration control. Contributing to these problems were weaknesses in the corrective action and technical support areas. In addition, weaknesses in management oversight during the event contributed to the delay in restoring normal electrical power supplies.

Some of the discussion concerned the load tap changer being outside of design basis and the licensee's looking at the secondary side of the amp current instead of the primary side. There was also discussion of the revised oversight process and whether it would have identified these problems. It was concluded that some of the problems at IP2 were corrective action problems. The latter part of the discussion focused on the risk significance of the event. The NRC estimate of the conditional core damage probability for this event was about 2E-4. The licensee's estimate was about 1.88E-04. The IP2 baseline core damage frequency is 3.3E-05 for internal events.

Steam Generator Tube Failure

On February 15, 2000, IP2 experienced a steam generator tube failure, leading to a manual reactor trip and the declaration of an Alert. The steam generator (SG) that was the source of the leak was identified and isolated. The high-pressure steam dump valves were opened, causing an excessive plant cooldown rate and a rapid reduction in the pressurizer level, which then required the initiation of safety injection (SI). The SI was reset, reactor coolant system pressure was reduced, and plant cooldown was resumed. The residual heat removal system was placed in service and plant pressure was reduced below the SG pressure to stop the SG tube leakage. The plant entered cold shutdown and the Alert was exited. The event had moderate risk significance. It resulted in a minor radiological release that was well within regulatory limits. No radioactivity above normal background levels was measured offsite, and the event did not impact upon public health and safety.

Problems were identified in several areas, including procedure quality, equipment performance, technical support, and emergency response. These problems challenged the operators, complicated the event response, and delayed the plant cooldown.

A short film was shown of the crack and much of the ensuing discussion focused on the event and the location of the failure.

Conclusion

This was an information briefing and no action was required.

8. Siemens S-RELAP-5 Appendix K Small-Break LOCA Code

The Committee received a report on the results of the August 8–9, 2000, meeting of the Thermal-Hydraulic Phenomena Subcommittee. The Subcommittee met to begin a review of the Siemens Power Corporation (SPC) S-RELAP5 code for application to modeling of Appendix K (evaluation model) small-break LOCAs. SPC had recently submitted this version of its code to the NRC staff for review. Discussions during the Subcommittee meeting centered on the details of the code's models and correlations with the small-break LOCA evaluation model version. The Subcommittee identified concerns relative to code documentation and the limitations of specific code models. The staff has a copy of the code and will investigate issues identified by the staff and the Subcommittee. Committee review of this matter will follow issuance of the staff safety evaluation, scheduled for late this year.

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

The Committee continued its discussion of the NRC Safety Research Program and the format and content of the ACRS 2001 report. The Committee indicated that the focus of its report will be on the long-term research needed to facilitate the execution of the NRC's mission in the future. In addition, the report should help the Commission determine when a research effort has yielded enough information to support regulatory decisionmaking. The Office of Nuclear Regulatory Research will help the Committee with aspects of the report.

Conclusion

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

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

The Committee held an unplanned, unscheduled discussion with a representative of the Union of Concerned Scientists (UCS) on the UCS August 2000 report entitled "Nuclear Plant Risk Studies: Failing the Grade." The Committee discussed the UCS concern over the industry's use or misuse of risk information for burden reduction. The Committee also discussed the UCS concern over the number of risk-informed license amendment requests being processed by the NRC staff without the benefit of a detailed licensee risk analysis. UCS contends that the staff has limited ability to detect poor risk analysis because licensees normally only submit their conclusions, omitting the applicable portions of the PRA or supplemental analyses.

Conclusion

The Committee decided to continue its review of this UCS report during the October 5–7, 2000 ACRS meeting.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee discussed the response from the Executive Director for Operations (EDO) dated July 25, 2000, to ACRS comments and recommendations included in its letter dated June 20, 2000, concerning the proposed final Regulatory Guide and Standard Review Plan Section associated with the Alternative Source Term Rule.

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

- The Committee discussed the response from the EDO dated July 27, 2000, to the ACRS comments and recommendations included in the ACRS report dated June 22, 2000, concerning the staff's draft report, "Regulatory Effectiveness of the Station Blackout Rule."

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

- The Committee discussed the response from the EDO dated August 30, 2000, to the ACRS comments and recommendations included in the ACRS report dated July 20, 2000, concerning the Nuclear Energy Institute letter dated January 19, 2000, addressing NRC plans for risk-informing the technical requirements in 10 CFR Part 50.

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

- The Committee discussed the response from the EDO, dated July 14, 2000, to ACRS comments and recommendations included in the ACRS/ACNW joint report dated May 25, 2000, concerning use of defense in depth for risk-informing the activities of the Office of Nuclear Materials Safety and Safeguards.

  The Committee decided that it was satisfied with the EDO's response but recommended that the ACRS/ACNW Joint Subcommittee follow-up during future meetings on selected issues such as defense in depth versus safety margins, risk acceptance criteria and safety goals, and options to achieve balance between compensatory measures and reduction in risk concerning the high-level waste repository.

- The Committee discussed the response from the EDO, dated July 20, 2000, to the ACRS comments and recommendations included in the ACRS report dated June 20, 2000, concerning the proposed resolution of Generic Safety Issue-173A, "Spent Fuel Storage Pool for Operating Facilities."

  The Committee decided it was satisfied with the EDO's response, but it will continue to follow-up on this issue as work progresses.

- The Committee discussed the response from the EDO, dated July 17, 2000, to the ACRS comments and recommendations included in the ACRS/ACNW report (NUREG-1635, Vol. 3) dated March 2000, concerning the review and evaluation of the Nuclear Regulatory Commission safety research program.

  The Committee decided it was satisfied with the EDO's response, but it will continue to follow-up and discuss this matter with the NRC staff as work progresses.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from July 12 through August 28, 2000, the following Subcommittee meetings were held:

- Thermal-Hydraulic Phenomena - August 8-9, 2000

  The Subcommittee discussed the Siemens Power Corporation's S-RELAP5 thermal-hydraulic systems code. Most of the meeting was closed to public attendance to discuss proprietary information per 5 U.S.C. 552b(c)(4) pertinent to Siemens Power Corporation.

- Planning and Procedures - August 28, 2000

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

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Date Issued: 11/17/2000
Date Certified: 11/27/2000

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- Cases and Significance of Design Basis Issues at U.S. Nuclear Power Plants (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated September 8, 2000)

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I. Federal Register Notice
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475th ACRS Meeting
August 29-September 1, 2000

MINUTES OF THE 475TH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AUGUST 29–SEPTEMBER 1, 2000 ROCKVILLE, MARYLAND

The 475th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on August 29–September 1, 2000. Notice of this meeting was published in the *Federal Register* on August 18, 2000 (65 FR50576) (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.

A transcript of selected portions of the meeting was kept and is available in the NRC Public Document Room at the One White Flint North Building, Mail Stop 1F-15, Rockville, MD, 20852-2738. [Copies of the transcript are available for purchase from Ann Riley & Associates, Ltd., 1025 Connecticut Avenue, NW, Suite 1014, Washington, DC 20036, and on the ACRS/ACNW Web page at (www.NRC.gov/ACRS/ACNW).]

ATTENDEES

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

I. Chairman's Report (Open)

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

Dr. Dana A. Powers, Committee Chairman, convened the meeting at 8:30 a.m. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee. The Chairman introduced Mr. Graham M. Leitch, a new ACRS member.

II. Proposed Risk-Informed Revisions to 10 CFR Part 50 (Open)

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

Dr. William Shack, the cognizant ACRS member for this issue, introduced the topic. He stated that the purpose of the meeting was to discuss proposed risk-informed revisions to 10 CFR 50.44, “Standards for combustible gas control in light-water cooled power reactors,” and related matters. He noted that the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment met on June 29 and July 11, 2000.

NRC Staff Presentations

Mr. Timothy Reed, NRR, gave a brief presentation on proposed risk-informed revisions to the special treatment requirements of 10 CFR Part 50 (Option 2). Ms. Cynthia Carpenter and Mr. Joe Williams, NRR, provided supporting discussion. The staff summarized the proposed reconciliation of public comments on the Advance Notice of Proposed Rulemaking (ANPR) for 10 CFR 50.69 and Appendix T. They also discussed selective implementation, the need for prior NRC review, PRA quality, and the need for changes to other regulations (e.g., 10 CFR Part 21 for reporting of defects and noncompliance). Significant points made during the presentation include the following:

- Public comments were in general agreement with the approach proposed in the ANPR, particularly with respect to the staff's plans for a phased approach. Some public comments suggested that the approach be optional and not mandatory, allow for performance-based methods to meet the requirements, and allow for selective implementation. Some public comments also suggested that the Backfit Rule be applied if any new requirements are proposed.

- Other public comments were that Appendix T is too detailed, prescriptive, and burdensome that the NRC should not endorse consensus standards as the “only” method for meeting PRA quality expectations, and that the NEI peer review certification process described in NEI 00-02 should also be considered as a means of meeting NRC criteria for risk-informed decision making.

- In general, the industry and staff are in close agreement on the categorization of structures, systems, and components (SSCs). However, the staff and industry differ in their views on the regulatory treatment of SSCs, particularly with regard to RSIC-2 (safety-related, not risk significant).

- The staff's review of the South Texas Project exemption request is continuing. The staff expects to complete its draft safety evaluation report in early November 2000.

Mr. Thomas King and Ms. Mary Drouin, RES, briefed the Committee on proposed risk-informed revisions to the technical requirements of 10 CFR 50.44 and related matters (Option 3). Significant points made during the presentation include the following:

- Fuel damage associated with a core melt accident can potentially produce combustible gases (i.e., hydrogen and carbon monoxide) from reactions of the fuel cladding and core with concrete.

- Hydrogen is not a significant challenge to containment within the first 24 hours of core damage. Unmitigated, long-term hydrogen buildup can reach into explosive concentrations. Core damage, combined with a breach of containment, could result in an offsite release and have an adverse impact on public safety and the environment.

- Based on its technical evaluation of the hazards and in response to the petition for rulemaking submitted by Performance Technology, Inc. the staff proposes to modify the following regulatory provisions in 10 CFR 50.44 as follows:

  - enhance the analytical requirements associated with the hydrogen source term,
  - eliminate the requirement to measure hydrogen concentration,
  - the requirement to ensure containment atmosphere mixing,
  - eliminate the requirement for post-accident hydrogen recombiners,
  - enhance the requirements for hydrogen igniters in BWR Mark III and PWR ice condenser containments, and
  - allow for risk-informed and performance-based methods.

- The staff will retain the following requirements to:

  - for high-point reactor vessel vents, and
  - inerting BWR Mark I and II containments.

- The staff should take action on the proposed rulemaking 10 CFR 50.44 independently of the Option 3 initiative. The NEI Task Zero initiative demonstrated that removal of combustible gas control systems is a risk-positive change.

- New regulatory requirements and safety enhancements should be required to pass the Backfit Rule.

With respect to Option 2 of the ANPR, Dr. Apostolakis asked what the staff expected to review in terms of categorization and special treatment. The staff stated that it would be desirable for the revised rule and associated guidance to enable licensees to make certain changes without NRC review of both the categorization and the special treatment. Dr. Apostolakis asked whether the staff would review the PRA and/or risk analysis supporting the proposed change. He also suggested that it would be worthwhile to know how the expert panel made decisions. The staff stated that verification of allowed changes would likely be considered in the post-implementation phase but acknowledged that there may be some difficulty with the PRA.

Dr. Seale asked what success criterion would be used for risk-informed changes under Option 2. Dr. Apostolakis stated that the impact on core damage frequency (CDF) and large early release frequency (LERF) might not be known because the system might be insensitive to the change. Dr. Powers suggested that CDF and LERF may not be the right measures. Dr. Wallis suggested that the criterion might be that sufficient safety margins are maintained. The staff stated that it hoped to develop a better understanding of how the expert panels treat risk information and safety margin during the pilot applications. Dr. Apostolakis stated that the approach relies heavily on importance measures and noted that the Committee previously expressed concern over the need for training expert panels on the proper use of importance measures.

Dr. Apostolakis stated that there is some merit in the industry's suggestion that the proposed Appendix T might be more effective as a regulatory guide. He noted that a regulatory guide may provide more flexibility in the use of alternative risk analysis techniques, e.g., the Top Event Prevention (TEP) methodology used by Consumers Power Company. The staff agreed that there might be some merit to using a regulatory guide, which might endorse some form of industry guidance, and noted that a decision had not yet been made on the proposed use of Appendix T.

Dr. Powers asked about the types of accidents being considered in the 24-hour cutoff for 10 CFR 50.44. In particular, he asked about events in which the containment atmosphere would become stratified. The staff said that a station blackout event represents a substantial hazard for certain containment designs because of the loss of containment mixing and the need for emergency power to igniters.

Conclusion

The Committee sent a report dated September 12, 2000, to Chairman Meserve on this matter. The Committee also decided to schedule a briefing during the October 5-7, 2000, ACRS meeting, to review NEI 00-02.

III. Causes and Significance of Design Basis Issues at U.S. Nuclear Power Plants (Open)

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

Dr. Robert L. Seale, the cognizant member, introduced this topic. He mentioned the Committee's previous concern regarding the loss of independence as a result of a reorganization in which AEOD became a part of RES and NRR. Additionally, he emphasized the importance of the information being presented because of the movement toward risk-informed regulation and the opportunity for comparison.

NRC Staff Presentations

Mr. Ronald Lloyd, RES, gave a presentation on the causes and significance of design basis issues (DBIs). He stated that the study report documents results of a systematic and comprehensive study of design basis issue trends and patterns. The study provides insights from reported design basis issues with respect to (1) their causes, significant patterns within both the power reactor industry and power reactor systems, frequency trends, safety consequences, and risk significance; (2) the lessons that may be useful in assessing regulatory effectiveness of NRC's evolving inspection and plant performance assessment processes and the definition of plant design basis; and, (3) regulatory burden implications related to NRC licensee event reporting requirements for design basis issues. The insights from this study are intended to assist NRC and the industry with ongoing efforts to make NRC's regulatory framework and oversight process more risk informed and performance based and to reduce unnecessary regulatory burden.

The information for the report was compiled from data gathered from 1985 through 1997. The report showed that (1) there were more than 3100 licensing event reports (LERs) with DBIs during the reporting period and more than 500 in 1997 which was the focus year, (2) the number of reported events increased during NRC initiatives, (3) only a small percentage of DBIs were classified as accident sequence precursor events. The most common causes of DBIs were original design error, procedure deficiency, and human error. Three safety-related systems accounted for a majority of the potentially risk significant DBIs, older plants generally reported more DBIs than newer plants, and from 1990 to 1997, the percent of LERs with DBIs with accident sequence precursor events steadily decreased while the number of DBIs increased. The Committee discussed the risk assessment tools for fire and again concluded that we do not have a good risk model.

Conclusion

A letter dated September 8, 2000, was sent to Chairman Meserve expressing satisfaction with the agency efforts to continue analyses of experiential data.

IV. Proposed Final Regulatory Guide (DG-1093) Endorsing NEI 97-04 Document on Design Basis

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

Dr. Robert L. Seale, the cognizant member, introduced this topic. It was noted earlier that this and the previous topics were related in that they both dealt with design bases. The term "design basis" is used in several regulations in 10 CFR Part 50. It is also useful for evaluating degraded and nonconforming conditions.

NRC Staff Presentations

The presentation on the regulatory guide endorsing NEI 97-04, Appendix B, "Design Basis Program Guidelines," was made by Mr. Steward Magruder, NRR, and Mr. Russ Bell, NEI. Mr. Magruder stated that the purpose of this part of the meeting was to present the proposed final regulatory guide and obtain Committee approval for issuance. The regulatory guide endorses NEI 97-04 and the objective was to develop guidance that provides a clearer understanding of what constitutes design basis information as defined in 10 CFR 50.2.

The NEI guidance was developed as a result of system-specific engineering inspections that showed that some licensees were not maintaining design basis information as required by NRC regulations. In response to the problems identified during these inspections and other problems identified by the licensees, most nuclear power plant licensees initiated design basis reconstitution programs. These programs sought to identify and selectively regenerate missing documentation. During the documentation effort, it became clear that the definitions of what constituted design basis information differed from licensee to licensee. The lessons learned from events at Millstone and Maine Yankee showed that the definition of design basis should be clarified. A Senior Requirements Memorandum (SRM) dated August 7, 1998, requested that the guidance be developed.

The proposed final regulatory guide endorses the NEI guidance without exception because the NRC staff and NEI representatives were able to resolve differences that had previously existed.

The general guidance defines design basis functions as those performed by systems, structures, and components that are (1) required, or otherwise necessary, to comply with regulations, license conditions, orders, or technical specifications, or (2) credited in licensee safety analyses to meet NRC requirements.

The guidance defines design bases values as values or ranges of values of controlling parameters established as reference bounds for design to meet design basis functional requirements. These values may be (1) established by NRC requirement, (2) derived from or confirmed by safety analyses, or (3) chosen by the licensee from an applicable code, standard, or guidance document.

Conclusion

The Committee voted to support staff endorsement of the NEI guidance. Dana A. Powers sent a letter dated September 12, 2000, to the Executive Director of Operations (EDO) recommending issuance of DG-1093 and endorsing NEI 97-04, Appendix B.

V. AP1000 Standard Plant Design (Open)

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

The Committee heard a presentation by and held discussions with the staff regarding the results of the staff's preapplication (Phase 1) review of the Westinghouse Electric Company's proposed AP1000 standard plant design. Westinghouse plans to seek certification of a 1000 Mwe nuclear plant design similar to the certified AP600 design and seeks NRC feedback on the scope and cost to reviewing and certifying the AP1000 design.

The NRC and Westinghouse have agreed to a three-phase review approach noted below:

<u>Phase I</u>

- Identify the review assumptions and issues that need to be evaluated in Phase II.
- Identify the information that the NRC will need to evaluate these assumptions and issues.
- Estimate the schedule and resources needed to perform the Phase II review.

<u>Phase II</u>

- Determine the scope of the AP1000 design certification review.
- Estimate the schedule and resources needed to perform the Phase III review.
- Request Commission approval of Phase II evaluation.

<u>Phase III</u>

- Perform design certification review.

<u>Preapplication Review Items Proposed by Westinghouse and the NRC Staff's Response to the Westinghouse Proposal</u>

In a letter dated May 31, 2000, Westinghouse identified five fundamental assumptions, noted below, for evaluation by the staff during the Phase II preapplication review of the AP1000 design. In a letter dated July 27, 2000, the NRC staff provided the results of its assessment of the Westinghouse proposal. Staff responses to the Westinghouse proposal are also included under each item.

— The AP1000 Design Certification Application will reference sections of the AP600 Design Control Document (DCD) that do not change for AP1000.

  Westinghouse will submit a table of contents of the DCD for the AP1000 design for review by the NRC. At the conclusion of the Phase II review, Westinghouse expects to reach an agreement with the NRC on the table of contents for the DCD, including a determination of the sections that can be retained from the AP600 DCD that will not be subject to re-review.

  The staff states that in order to determine which sections of AP600 DCD will not require re-review for AP1000, Westinghouse should provide a description of its proposed design changes, with a level of detail comparable to that provided in Section 1.2 of the AP600 DCD and a rationale for why changes are not needed in certain sections of the AP600 DCD.

— The AP1000 design certification will not require additional tests to be performed by the applicant.

  Westinghouse will submit an AP1000 analysis plan and scaling assessment of the AP600 test program. The NRC should determine whether the AP600 test program meets the requirements of 10 CFR Part 52 for the AP1000 design.

— The AP1000 design certification can utilize the AP600 analysis codes with limited modifications. Westinghouse will submit the AP1000 analysis plan and the scaling assessment of AP600 test program and the AP1000 passive core cooling system design margins assessment. Westinghouse will provide an assessment of the applicability of each code and will identify code changes to address the most significant comments documented in NUREG-1512, "Final Safety Evaluation Report Related to Certification of AP600 Standard Design." The NRC should determine whether the AP600 analysis codes, including the proposed changes are adequate for analyzing the AP1000 design.

For items 2 and 3, the staff states that in order to determine whether the AP600 test program (including test matrices) and code validation are sufficient for AP1000, Westinghouse must develop a phenomena identification and ranking table (PIRT) for AP1000, identify key thermal-hydraulic phenomena and parameter ranges, and identify any new phenomena or differences from the AP600 PIRTs for large- and small-break LOCAs and non-LOCA transients. In addition, the staff requests Westinghouse to provide necessary information on various thermal-hydraulic tests and codes for use by the staff to determine whether additional tests and code changes are needed for AP1000. For example:

- Westinghouse must demonstrate that the existing separate effects tests on the passive residual heat removal system heat exchanger, automatic depressurization system, and core makeup tank sufficiently cover the range of key thermal-hydraulic phenomena and parameters or acquire additional test data.

- Westinghouse must submit a scaling report for the integral system tests, such as OSU/APEX and SPES-2 (high pressure, full vertical scale) for AP1000 and demonstrate that the test matrices of OSU/APEX and SPES-2 provided adequate coverage of the break sizes and locations to address important system-related phenomena identified in the AP1000 design. It is possible that additional integral system tests may be required, especially for validation of the NOTRUMP code for small-break LOCA analysis and the WCOBRA/TRAC code for long-term cooling analysis.

- Westinghouse will have to (a) provide justification on the acceptability of the WRB-2 CHF correlation to the new fuel design by demonstrating that sufficient test data exist to cover the geometrical and thermal-hydraulic conditions of the new fuel design, (b) acquire additional critical heat flux data to cover the new fuel design and thermal-hydraulic conditions and demonstrate that the WRB-2 correlation adequately predicts new data, or (c) develop a new CHF correlation (including WRB-2 modification).

- Westinghouse needs to explain how the LOFTRAN code has been or will be changed to model AP1000 and why these changes are appropriate.

  - The limitations and restrictions, identified in NUREG-1512, on using the WGOTHIC code model for the AP600 evaluation need to be justified or modified accordingly for AP1000.

— The AP1000 design certification application can utilize the AP600 PRA supplemented with a sensitivity study to meet the requirements for a plant-specific PRA.

Westinghouse will submit the table of contents for the AP1000 PRA sensitivity study and AP1000 Level 1 PRA LOCA success sequences analysis report. The NRC should determine whether the AP600 PRA supplemented with a suitable sensitivity study meets the requirements for the AP1000 plant-specific PRA.

The staff states that Westinghouse should provide the following Level 1 PRA information.

- A detailed description of the approach that will be followed to confirm the validity of the success criteria for both systems and operator actions. In the AP600 PRA, the success criteria were determined by a risk-based margin approach that used conservative assumptions for key thermal-hydraulic parameters, such as decay heat. This process resulted in success criteria that are sequence dependent and take into account thermal-hydraulic uncertainties. Westinghouse should discuss how the proposed design changes will affect the implementation of the margins approach for AP1000. If it is proposed that some portion of the AP600 margins approach implementation be retained, Westinghouse should provide documentation showing that this action will not compromise the robustness of the success criteria (for both systems and human actions) used in the AP1000 PRA models.

- A list of changes is in the AP600 design with an explanation of why such changes would not introduce additional hardware failure mechanisms or increased hardware failure rates. Both power operation and shutdown operation need to be addressed.

— The AP1000 design certification application can defer selected design activities to the combined license (COL) applicant.

Westinghouse proposes to include less design detail in the AP1000 design certification application than was included in the AP600 application. The general arrangement, structural configuration, equipment and piping layout are substantially the same. However, qualification analyses will be deferred to the COL applicantion. Westinghouse requests that the NRC provide feedback on the level of design detail to be included in the AP1000 application.

The NRC staff states that Westinghouse should provide information necessary for the staff to determine whether Westinghouse can use design acceptance criteria (DAC) instead of detailed design information for the AP1000 seismic analysis, structural design, and piping design. Also, Westinghouse should demonstrate several things:

- the dynamic stability of the nuclear island (sliding and overturning)
- the adequacy of the 6-foot thick foundation mat (in the balance of plant area) under the increased design loads (dead loads and seismic loads).
- the design adequacy of the subcompartment walls to withstand higher pressures resulting from the increased size of nuclear steam supply system (NSSS) components
- that AP1000 steel containment will continue to meet the containment performance requirement for severe accidents (withstand the internal pressure at 24 hours after the start of an accident at ASME Service Level C limits)

The members provided the following comments:

- Supplementing the AP600 PRA with a sensitivity analysis may not be sufficient. The AP1000 PRA should include uncertainty distributions on core damage frequency, conditional containment failure probability, and large early release frequency.
- The seismic analysis should not be left solely to the COL applicant and should be included in the AP1000 PRA using a representative site.
- The staff obtained copies of the NOTRUMP, WCOBRA/TRAC, LOFTRAN, and WGOTHIC codes and performed an independent evaluation of these codes to determine their applicability to assess the adequacy of the AP1000 design.

- An uncertainty analysis should be performed to assess the uncertainties associated with the results of the NOTRUMP, WCOBRA/TRAC, LOFTRAN, and WGOTHIC codes.

VI. Performance-Based Regulatory Initiatives (Open)

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

Mr. John Sieber, Acting Chairman of the Reliability and Probabilistic Risk Assessment Subcommittee, provided background information regarding the development of the proposed Commission paper concerning high-level guidelines for performance-based activities. He summarized the Committee's previous review activities related to the proposed paper.

Mr. Prasad Kadambi, RES, provided an overview of the development of the proposed guidelines. Mr. Robert Youngblood, ISL, Inc., presented a case study that applied the proposed guidelines to the present requirements for combustible gas control in certain types of containment. He concluded that some aspects of capability and performance parameters are not amenable to performance-based treatment and that the guidelines are useful in evaluating the viability of a performance-based approach within the regulatory framework.

The members and representatives of the staff discussed how the uncertainties associated with the selected parameters are addressed. They also discussed setting capability and performance parameters at the highest possible level of the event tree, and providing explicit guidance for selecting the appropriate number of redundant or overlapping parameters.

Mr. Christopher Smith, ISL, Inc., presented a case study that applied the proposed guidelines to a recently revised rule associated with respiratory protection requirements. He concluded that the results of applying the performance-based guidelines were consistent with the changes made to rule.

Mr. Kadambi explained the interrelationships among regulatory initiatives and the staff's plans for applying the guidelines to future regulatory activities. He concluded that the staff had demonstrated the usefulness of the guidelines and that it expected to improve the guidelines as experience dictated.

The members and the staff discussed the differences between the probability of risk related to radiation and chemicals, why the viability guidelines were tested, and the need for other kinds of acceptance criteria besides core damage frequency (CDF) and large early release frequency (LERF).

Conclusion

The Committee sent a letter dated September 8, 2000 to the EDO on this matter.

VII. License Renewal Guidance Documents (Open)

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

Dr. Bonaca, Chairman of the Plant License Renewal Subcommittee, noted that the staff and industry were developing a set of license renewal guidance documents, which would be released for public comment. He explained that the Subcommittee planned to review these documents during the October 19-20, 2000, ACRS Subcommittee meeting. Dr. Bonaca noted that the purpose of the staff presentation was to explain the status of the documents.

Mr. Christopher Grimes, NRR, informed the Committee that the documents would be distributed to the public over the next several days. Mr. Samson Lee, NRR, provided background related to the development of the guidance documents and an overview of how the documents are intended to work together. He also presented the schedule for review and approval of the documents. Mr. Lee summarized the contents of the standard review plan section, the Generic Aging Lessons Learned Report, the Regulatory Guide, and Revision 2 to NEI 95-10.

The members and the staff discussed differences between the various drafts of the documents, the disposition of the concerns identified in the Union of Concerned Scientists' report, the extent of guidance regarding the scoping and screening processes, and the disposition of license renewal generic issues.

Conclusion

This briefing was for information only. No Committee action is required.

VIII. Operating Events at Indian Power Nuclear Power Plant Unit 2 (Open)

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

Dr. Robert L. Seale, cognizant member, introduced this topic. He said that there would be presentations on two events at Indian Point Unit 2 (IP2). The first event was a reactor trip with complications that occurred on August 31, 1999. The second event, a steam generator tube failure, occurred on February 15, 2000.

NRC Staff Presentations

The presentations on operating events at IP2 focused on two events. After introductory remarks by Mr. Ledyard Marsh, NRR, presentation of the reactor trip with complications was given by Mr. Jimi Yerokum, Region I. The steam generator tube failure presentation was given by Mr. Raymond Lorson, Region I. Mr. James Trapp, Region I, participated in both presentations discussing the risk significance of the events. Mr. Brian Holian, Region I, provided comments throughout the presentations and did a summary at the conclusion of both presentations. The purpose of the presentations was to hear findings and conclusions of the augmented inspection team (AIT) that reviewed the two events at IP2.

Reactor Trip with Complications

Mr. Yerokum discussed the event and its causes. On August 31, 1999, the IP2 reactor automatically tripped from 99% power due to a spurious reactor protection system (RPS) overtemperature delta-temperature (OTΔT) trip signal. The normal offsite power breakers to all four 480 volt (V) vital buses also tripped unexpectedly, and all three emergency diesel generators (EDGs) started and began to assume loads on their respective 480 V buses. A short time later, the 23 EDG output breaker tripped, leaving the 6A vital bus deenergized. This resulted in a loss of power to one of the two motor-driven auxiliary feedwater pumps, battery charger 24, some emergency core cooling components, and other equipment. Battery 24 subsequently discharged in about 7 hours, causing a loss of power to the direct current (dc) loads on dc panel 24 and the loads on 118 volt alternating current (ac) instrument bus 24. The deenergization of the instrument bus caused a loss of most of the control room annunciators for various safety-related systems, which required the declaration of an Unusual Event. On September 1, 1999, vital bus 6A was reenergized and normal offsite power restored.

Although there was no immediate threat to public health and safety, the event was risk significant. There was no radiological release from the event.

The AIT determined that the event was preventable and was caused primarily by problems in plant configuration control. Contributing to these problems were some notable weaknesses in the corrective action and technical support areas. In addition, weaknesses in management oversight during the event contributed to the delay in restoring normal electrical power supplies.

A configuration control problems was that the station auxiliary transformer load tap changer was left in a position contrary to the licensing basis. This led to a loss of offsite power to the vital buses following the plant trip. Poor control of emergency diesel generator output breaker short time overcurrent trip settings, compounded by a deficiency in the timing of the sequencing relays for some safety-related loads, caused the loss of emergency power to one of the vital buses.

Management did not promptly recognize the significance of the degrading conditions associated with the event. Managers appeared to focus primarily on developing shutdown work plans and schedules instead of establishing and prioritizing activities to restore plant equipment and to limit further risk. As a result of these weaknesses, station personnel provided poorly coordinated and untimely support to plant operators in restoring normal electrical power. Likewise, the post-trip response organization did not provide support to operations in the review of plant conditions related to the emergency plan. As a result, station personnel did not recognize that they should have declared an Unusual Event when offsite power was lost to all 480 volt vital buses.

Some of the discussion centered around the circumstances of the event, the load tap changer was outside of design basis, and personnel looked at the secondary side of the amp current instead of the priority side. There was also discussion regarding the revised oversight process and whether or not some of these problems would have been identified with the process. The Committee concluded that some of the problems at IP2 were corrective action problems. The latter part of the discussion focused on the risk significance of the event. The NRC estimate of the conditional core damage probability for this event was estimated to be about 2E-4. The licensee's estimate was about 1.88E-04. The IP2 baseline core damage frequency is 3.3E-05 for internal events.

Steam Generator Tube Failure

Mr. Lorson discussed this event as follows. On February 15, 2000, the IP2 nuclear plant experienced a steam generator tube failure (SGTF) that required the declaration of an Alert and a manual reactor trip. The #24 steam generator

(SG) was determined to be the source of the leak and was isolated. The high-pressure steam dump valves were opened causing an excessive primary plant cooldown rate which caused a rapid reduction in the pressurizer level, which required the initiation of safety injection (SI). The SI was reset, reactor coolant system (RCS) pressure was reduced, and plant cooldown was recommenced. The residual heat removal (RHR) system was placed in service and primary plant pressure was reduced below the #24 SG pressure to terminate the SG tube leakage. The plant entered cold shutdown and the Alert was exited.

The event had moderate risk significance. It resulted in a minor radiological release well within regulatory limits. No radioactivity was measured offsite above normal background levels, and the event did not impact the public health and safety.

Problems were identified in several areas, including operator performance, procedure quality, equipment performance, technical support, and emergency response. These problems challenged the operators, complicated the event response, and delayed the plant cooldown.

A short film was shown of the crack and much of the ensuing discussion focused on the event and the location of the failure.

Conclusion

This was an information briefing and no action was taken.

IX. Siemens SRELAP-5 Appendix K Small-Break LOCA Code (Open)

Dr. G. Wallis, Chairman, Thermal-Hydraulic Phenomena (T/H-P) Subcommittee, reported on the results of the T/H-P Subcommittee meeting of August 8-9, 2000 which was held to begin review of the Siemens Power Corporation (SPC) S-RELAP5 code. Specifically, SPC has submitted for NRC staff review and approval an Appendix K small-break (SB) LOCA version of the code. The subcommittee discussions centered on two topics: the details of the code models and correlations, and the specifics of the Appendix K SB LOCA code version.

Dr. Wallis made the following points:

— Perusal of the models and correlations documentation showed numerous instances of missing or incomplete/poor documentation. A number of typos were also found. In some instances the modeling methods used were not explained. Dr. Wallis has a list of these concerns; he will send it to the NRC staff.

— The detailed presentation by Mr. J. Kelly, SPC, on the models and correlations provided substantial information on the code not found in the documentation. The Subcommittee members agreed that Mr. Kelly's presentation made the code appear more robust.

— The code exhibits problems with regard to modeling momentum. However, for the case at hand (the SBLOCA evaluation model), the impact of momentum is small. The Subcommittee believes that SPC should provide a quantitative argument to this effect.

— NRR needs to consider what acceptance criteria it will apply to the uncertainty in the code outputs. Mr. Caruso, NRR, said that this issue is addressed in a regulatory guide addressing use of "best estimate" ECCS codes (Regulatory Guide 1.157).

— For the SBLOCA code, the SPC assessment process appeared weak. Dr. Powers said that SPC agreed that a more logical and disciplined approach is needed here.

— Problems were seen in the modeling of void distribution in the core and the liquid level model for the loop seal clearing. For the latter, SPC biased the model to ensure consistent results, as the code cannot model two-phase instability.

— Mr. Landry, NRR, said that the SPC SBLOCA code will only be applicable to three-and four-loop Westinghouse PWR plants. He also said that the staff will impose conditions on the use of this code version. In a related matter, NRR said that the draft regulatory guide and the SRP section pertaining to submittal and review of codes are scheduled to be issued for public comment in September 2000.

— In closing, Dr. Wallis said that the Subcommittee does not plan further review of this matter until the staff has issued its safety evaluation, scheduled for the December or January.

X. <u>Annual Report to the Commission on the NRC Safety Research Program</u> (Open)

The Committee continued its discussion of the NRC Safety Research Program and the format and content of the ACRS 2001 report. The Committee indicated that the focus of its report will be on the long-term research needed to facilitate the execution of the NRC's mission in the future. In addition, the report should be helpful to the Commission in determining when a research effort has yielded enough information for regulatory decision making. The Office of Nuclear Regulatory Research will cooperate with ACRS on this report.

Conclusion

The Committee will continue its discussion and preparation of the ACRS 2001 report to the Commission on the NRC research programs at future ACRS meetings and at a Subcommittee meeting scheduled for November 1, 2000.

XI. Union of Concerned Scientists Report, "Nuclear Plant Risk Studies: Failing the Grade" (Open) (Unscheduled Agenda Item)

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

The Committee held an unplanned, unscheduled discussion with Mr. David Lochbaum of the Union of Concerned Scientists (UCS) concerning the UCS report August 2000 report entitled "Nuclear Plant Risk Studies: Failing the Grade." The Committee discussed the UCS concern regarding the industry's use or misuse of risk information for burden reduction. The Committee also discussed the UCS concern over the number of risk-informed license amendment requests being processed by the NRC staff without the benefit of licensee's detailed risk analysis. UCS contends that the staff has limited ability to detect poor risk analysis because licensees normally only submit their conclusions, omitting the applicable portions of the PRA or supplemental analysis.

Dr. Apostolakis asked about the apparent omission of PRA contributions to the development of regulations such as the Station Blackout (SBO) Rule, the Anticipated Transient Without Scram (ATWS) Rule, and the requirement for automatic actuation of auxiliary feedwater. Mr. Lochbaum stated that these regulations were promulgated in response to operating plant events and not PRA.

Dr. Apostolakis asked about the UCS recommendation that no risk decisions be made until industrial standards (e.g., ASME, ANS, NFPA, etc.) are approved or endorsed by the NRC. Mr. Lochbaum stated that he had a discussion with representatives of the NRC Office of the Inspector General about this recommendation. He stated that the UCS believes that no risk-informed decision should be made by the NRC without reviewing the licensee's risk analysis.

Conclusion

The Committee decided to continue its review of the UCS report during the October 5-7, 2000 ACRS meeting.

XII. 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 Executive Director for Operations (EDO) dated July 25, 2000, to ACRS comments and recommendations included in its letter dated June 20, 2000, concerning the proposed final Regulatory Guide and Standard Review Plan Section associated with the Alternative Source Term Rule.

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

- The Committee discussed the response from the EDO dated July 27, 2000, to the ACRS comments and recommendations included in the ACRS report dated June 22, 2000, concerning the staff's draft report, "Regulatory Effectiveness of the Station Blackout Rule."

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

- The Committee discussed the response from the EDO dated August 30, 2000, to the ACRS comments and recommendations included in the ACRS report dated July 20, 2000, concerning the Nuclear Energy Institute letter dated January 19, 2000, addressing NRC plans for risk-informing the technical requirements in 10 CFR Part 50.

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

Member assignments and priorities for ACRS reports and letters for the September ACRS 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 November 2000 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 a Committee decision.

Differing Professional Opinion (DPO) Issues Associated with Steam Generator Tube Integrity

In a memorandum dated July 20, 2000, to the ACRS Executive Director from the EDO, it was requested that the ACRS assist in the process of reviewing a DPO on steam generator tube integrity issues. Specifically, the EDO requested that the ACRS function as the equivalent of an ad hoc panel, under the NRC Management Directive 10.159 to review the DPO.

Subsequent to the EDO memorandum, the DPO author requested a meeting with the ACRS Executive Director. On July 24, 2000, Dr. Larkins and Mr. Duraiswamy met with the DPO author to discuss the EDO's request to the ACRS, previous ACRS comments on Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes," and other related matters. During that meeting, the DPO author stated that he did not have any objection to the ACRS reviewing the DPO issues as requested by the EDO and has some concerns that warrant the attention of the EDO. In a memorandum dated July 28, 2000, the DPO author provided his concerns to the EDO. The EDO responded to the DPO author on August 4, 2000 stating that: "In selecting the ACRS as the ad hoc panel, I considered its previous involvement in and knowledge of the technical issues." Dr. Larkins also sent a memorandum to the DPO author on August 14, 2000 documenting the items discussed with the DPO author on July 24, 2000. The EDO plans to provide consultants (Dr. Catton, Thermal-Hydraulic Issues; Dr. Richer, NIST, IGSCC; and Mr. Higgins, BNL, Human Performance) to the ACRS to provide technical support in reviewing the DPO issues.

ASLB Decision on Shearon Harris

The ACRS reports on spent fuel pool fires at decommissioning plants and the report on generic safety issue for spent fuel pools for operating plants have been referenced in the ASLB petition on Shearon Harris' amendment to its operating license to modify its spent fuel pool (pp. 12-32). As a result of interveners referencing the ACRS reports in their case to support the need for NRC staff to prepare an environmental impact statement, the ACRS members, staff, or consultants could be subject to discovery in these proceedings, which may require ACRS members, staff, or consultants to provide testimony or written material for these hearings.

The Board of Commissioners of Orange County (BCOC), North Carolina, is seeking admission of four late-filed environmental contentions (ECs) in the matter of Carolina Power & Light Co. (Shearon Harris Nuclear Power Plant). The Atomic Safety and Licensing Board (ASLB) on August 7, 2000, ordered that one contention (EC-6) be admitted for litigation; and rejected three contentions (EC-7, EC-8, EC-9) as inadmissible for litigation.

The ASLB in its ruling ordered the parties to conduct discovery beginning on August 21, 2000, and ending on October 20, 2000. The ASLB also notes that any attempt to obtain discovery materials from the ACRS is subject to the exceptional circumstances of 10 CFR 2.720 (h).

Power Uprate Issues

Mr. Boehnert summarized the list of issues associated with power uprates along with an anticipated schedule for ACRS review of power uprate applications.

Also, Dr. Cronenberg, ACRS Senior Fellow, developed a list of central issues associated with power uprates. This list was distributed to the members during the July 2000 ACRS meeting for review and comment.

Technical Exchange Meeting with RSK

During the July 2000 ACRS meeting, the Committee selected November 6-10, 2000, for a technical exchange meeting with RSK. The RSK has agreed to these dates for this meeting. ACRS members Apostolakis, Bonaca, Kress, Sieber, and Wallis plan to attend this meeting. Current plans would include travel to Germany and travel to Erlangen for a visit and discussion with Siemens and GRS consultants on digital I&C systems. Subsequently, we would travel to Munich, Garching, for a meeting with members of the RSK and GRS and BMU to discuss I&C issues, use of PRA in the regulatory process, future research needs for reactor safety, and other generic safety issues of interest to either Committee.

American Nuclear Society 2000 Utility Working Conference

Mr. Noel Dudley, ACRS staff, attended the ANS 2000 Utility Working Conference held at the Amelia Island Plantation, Florida, on August 6-10, 2000. The primary focus of the conference was on managing the business of nuclear power.

New ACRS/ACNW Compensation Report Form

The ACRS/ACNW Member Compensation Report has been revised to capture data on how much time members spend on the review of technical topics (e.g., license renewal, AP 1000, etc.).

License Renewal White Paper

The Subcommittee discussed a paper prepared by Dr. Bonaca on Potential Synergistic Effects of Industry Initiatives to Extend Plant Life, Increase Production, and Reduce Regulatory Burden.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 476th ACRS Meeting, October 5-7, 2000.

The 475th ACRS meeting was adjourned at 12:05 p.m. on Friday, September 1, 2000.

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

November 27, 2000

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

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

SUBJECT: CERTIFIED MINUTES OF THE 475th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), AUGUST-SEPTEMBER, 2000

I certify that based on my review of the minutes from the 475th ACRS full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

Dana A. Powers

Dana A. Powers, Chairman

November 27, 2000

Date

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

November 17, 2000

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 475th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - AUGUST-SEPTEMBER, 2000

Enclosed are the proposed minutes of the 475th 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

Signed at Washington, D.C. this 10th day of August, 2000.

Carl J. Poleskey,

hief, Branch of Construction Wage eterminations.

[FR Doc. 00–20771 Filed 8–17–00; 8:45 am]

BILLING CODE 4510-27-M

NUCLEAR REGULATORY COMMISSION

Revised Meeting Notice; Reactor Safeguard Advisory Committee

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 August 29–September 1, 2000, in Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the Federal Register on Thursday, October 14, 1999 (64 FR 55787).

Tuesday, August 29, 2000

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

8:35 A.M.–10:00 A.M.: Proposed Risk-formed Revisions to 10 CFR Part 50 pen)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding proposed NRC framework document for risk-informing the technical requirements of 10 CFR Part 50, proposed revisions to 10 CFR 50.44 concerning combustible gas control systems, and advance notice of proposed rulemaking (10 CFR 50.69 and Appendix T).

10:15 A.M.–11:15 A.M.: Causes and Significance of Design Basis Issues (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding a study of design basis issues and trends.

11:15 A.M.–12:00 Noon: Proposed Final Regulatory Guide (DG–1093) Endorsing NEI 97–04 Document on Design Bases (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed final version of the Regulatory Guide.

1:00 P.M.—1:45 P.M.: AP1000 Standard Plant Design (Open)—The Committee will hear presentations by d hold discussions with resentatives of the NRC staff and the estinghouse Electric Company regarding issues identified during AP1000 pre-application review (Phase 1).

1:45 P.M.—3:15 P.M.: Break and Preparation of Draft ACRS Reports (Open)—Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

3:15 P.M.—7:00 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting. In addition, the Committee will discuss a proposed ACRS report on Assessment of the Quality of PRAs.

Wednesday, August 30, 2000

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

8:35 A.M.—9:30 A.M.: Performance-Based Regulatory Initiatives (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding a Commission paper associated with performance-based regulatory initiatives.

9:30 A.M.—10:15 A.M.: License Renewal Guidance Documents (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the contents of the proposed Standard Review Plan, Generic Aging Lessons Learned Report, and a Regulatory Guide and associated NEI guidance documents.

10:30 A.M.—12:00 Noon: Operating Events at Indian Point Nuclear Power Plant Unit 2 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the licensee regarding the events, noted below, that occurred at the Indian Point Unit 2 Nuclear Power Plant and the associated staff findings, conclusions, and recommendations resulting from the evaluations of these events: (1) February 15, 2000 steam generator tube rupture event and (2) August 31, 1999 event involving reactor trip and loss of all off-site power.

1:00 P.M.—1:30 P.M.: Siemens SRELAP–5 Best-Estimate Small-Break LOCA Code (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Siemens Corporation regarding the Siemens SRELAP–5 best-estimate code for application to analysis of transients and small-break loss of coolant accident (LOCA). [NOTE: A portion of this session may be closed to discuss Siemens Corporation's proprietary information pursuant to 5 U.S.C. 552b(c)(4)].

1:30 P.M.–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.

2:30 P.M.–7:00 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Thursday, August 31, 2000

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

8:35 A.M.–8:45 A.M.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

8:45 A.M.–9:45 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.

9:45 A.M.–10:45 A.M.: Annual Report to the Commission on the NRC Safety Research Program (Open)—The Committee will discuss the format and content of the annual ACRS report to the Commission on the NRC Safety Research Program.

11:00 A.M.–12:00 Noon: 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.

1:00 P.M.–4:00 P.M.: Meeting with the NRC Commissioners on October 6, 2000 (Open)—The Committee will discuss and prepare topics for meeting with the Commissioners scheduled for October 6, 2000.

4:00 P.M.–6:00 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

Friday, September 1, 2000

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

8:35 A.M.–1:00 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 28, 1999 (64 FR 52353). In accordance with these procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify Mr. Howard J. Larson, 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. Howard J. Larson 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. Howard J. Larson 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. Howard J. Larson (telephone 301/415–6805), 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 :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: August 14, 2000.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 00–21061 Filed 8–17–00; 8:45 am]

BILLING CODE 7590–01–P

RAILROAD RETIREMENT BOARD

Agency Forms Submitted for OMB Review

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (44 U.S.C. Chapter 35), the Railroad Retirement Board (RRB) has submitted the following proposal(s) for the collection of information to the Office of Management and Budget for review and approval.

Summary of Proposal(s)

(1) *Collection title:* Evidence for Application of Overall Minimum.

(2) *Form(s) submitted:* G–319, G–320.

(3) *OMB Number:* 3220–083.

(4) *Expiration date of current OMB clearance:* 10/31/2000.

(5) *Type of request:* Extension of a currently approved collection.

(6) *Respondents:* Individuals or households.

(7) *Estimated annual number of respondents:* 290.

(8) *Total annual responses:* 121.

(9) *Total annual reporting hours:* 121.

(10) *Collection description:* Under section 3(f)(3) of the Railroad Retirement Act, the total monthly benefit payments payable to a railroad employee and his family are guaranteed to be no less than the amount which would be payable if the employee's railroad service had been covered by the Social Security Act.

ADDITIONAL INFORMATION OR COMMENTS: Copies of the forms and supporting documents can be obtained from Chuck Mierzwa, the agency clearance officer (312–751–3363). Comments regarding the information collection should be addressed to Ronald J. Hodapp, Railroad Retirement Board, 844 North Rush Street, Chicago, Illinois, 60611–2092 and the OMB reviewer, Joe Lackey (202–395–7316), Office of Management and Budget, Room 10230, New Executive Office Building, Washington, D.C. 20503.

Chuck Mierzwa,

Clearance Officer.

[FR Doc. 00–21068 Filed 8–17–00; 8:45 am]

BILLING CODE 7905–01–M

RAILROAD RETIREMENT BOARD

Agency Forms Submitted for OMB Review

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (44 U.S.C. Chapter 35), the Railroad Retirement Board (RRB) has submitted the following proposal(s) for the collection of information to the Office of Management and Budget for review and approval.

Summary of Proposal(s)

(1) *Collection title:* Student Beneficiary Monitoring.

(2) *Form(s) submitted:* G–315, G–315a, G–315a.1.

(3) *OMB Number:* 3220–0123.

(4) *Expiration date of current OMB clearance:* 10/31/2000.

(5) *Type of request:* Extension of a currently approved collection.

(6) *Respondents:* Individuals or households.

(7) *Estimated annual number of respondents:* 1,230.

(8) *Total annual responses:* 1,230.

(9) *Total annual reporting hours:* 121.

(10) *Collection description:* Under the Railroad Retirement Act (RRA), a student benefit is not payable if the student ceases full-time school attendance, marries, works in the railroad industry, has excessive earnings or attains the upper age limit under the RRA. The report obtains information to be used in determining if benefits should cease or be reduced.

ADDITIONAL INFORMATION OR COMMENTS: Copies of the forms and supporting documents can be obtained from Chuck Mierzwa, the agency clearance officer (312–751–3363). Comments regarding the information collection should be addressed to Ronald J. Hodapp, Railroad Retirement Board, 844 North Rush Street, Chicago, Illinois, 60611–2092 and the OMB reviewer, Joe Lackey (202–395–7316), Office of Management and Budget, Room 10230, New Executive Office Building, Washington, D.C. 20503.

Chuck Mierzwa,

Clearance Officer.

[FR Doc. 00–21069 Filed 8–17–00; 8:45 am]

BILLING CODE 7905–01–M

REVISED

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

August 9, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
475TH ACRS MEETING
AUGUST 29 - SEPTEMBER 1, 2000

TUESDAY, AUGUST 29, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

10:55
2) 8:35 - ~~10:00~~ A.M. Proposed Risk-Informed Revisions to 10 CFR Part 50 (Open) (WJS/GA/MTM)
- 2.1) Opening remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding proposed NRC framework document for risk-informing the technical requirements of 10 CFR Part 50, proposed revisions to 10 CFR 50.44 concerning combustible gas control systems, and advance notice of proposed rulemaking (10 CFR 50.69 and Appendix T).

10:55 - 11:10
~~10:00~~ - 10:15 A.M. ***BREAK***

11:10 - 12:25
3) ~~10:15~~ - ~~11:15~~ A.M. Causes and Significance of Design Basis Issues (Open) (RLS/MWW)
- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the NRC staff regarding a study of design basis issues and trends.

12:25
4) ~~11:15~~ - NOON Proposed Final Regulatory Guide (DG-1093) Endorsing NEI 97-04 Document on Design Basis (Open) (RLS/MWW)
- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff and NEI regarding the proposed final version of the Regulatory Guide.

12:45 - 2:00
~~12:00~~ - ~~1:00~~ P.M. ***LUNCH***

5)	2:00 - 2:40 ~~1:00~~ - ~~1:45~~ P.M.	AP1000 Standard Plant Design (Open) (TSK/SD) 5.1) Remarks by the Subcommittee Chairman 5.2) Briefing by and discussions with representatives of the NRC staff regarding issues identified during AP1000 pre-application review (Phase 1). Representatives of the nuclear industry will provide their views, as appropriate.
6)	2:40 ~~1:45~~ - 3:15 P.M.	Break and Preparation of Draft ACRS Reports (Open) Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.
	3:15-3:45	Break
7)	~~3:15~~ - ~~7:00~~ P.M. 3:45 - 6:00	Discussion of Proposed ACRS Reports (Open) Discussion of proposed ACRS reports on: 7.1) Proposed NRC Framework Document for Risk-Informing 10 CFR Part 50 and Associated Revisions to 10 CFR 50.44 Concerning Combustible Gas Control Systems, and Advance Notice of Proposed Rulemaking (10 CFR 50.69 and Appendix T) (WJS/GA/MTM) 7.2) Proposed Final Regulatory Guide (DG-1093) Endorsing NEI 97-04 Document on Design Basis (RLS/MWW) 7.3) AP1000 Pre-Application Review (TSK/SD) 7.4) Assessment of the Quality of PRAs (GA/MTM)

WEDNESDAY, AUGUST 30, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

8)	8:30 - 8:35 A.M.	Opening Remarks by the ACRS Chairman (Open) (DAP/JTL)
9)	8:35 - ~~9:30~~ 9:45 A.M.	Performance-Based Regulatory Initiatives (Open) (JDS/NFD) 9.1) Remarks by the Subcommittee Chairman 9.2) Briefing by and discussions with representatives of the NRC staff regarding a Commission paper associated with performance-based regulatory initiatives. Representatives of the nuclear industry will provide their views, as appropriate.
10)	9:45-10:25 ~~9:30~~ - ~~10:15~~ A.M.	License Renewal Guidance Documents (Open) (MVB/NFD) 10.1) Remarks by the Subcommittee Chairman 10.2) Briefing by and discussions with representatives of the NRC staff regarding the contents of the proposed Standard Review Plan, Generic Aging Lessons Learned Report, a Regulatory Guide and associated NEI guidance documents. Representatives of the nuclear industry will provide their views, as appropriate.
	10:25-10:45 ~~10:15~~ - ~~10:30~~ A.M.	***BREAK***

10:45 - 12:28

11) ~~10:30~~ - ~~12:00~~ Noon — Operating Events at Indian Point Nuclear Power Plant Unit 2 (Open) (RLS/MWW)

- 11.1) Remarks by the Subcommittee Chairman
- 11.2) Briefing by and discussions with representatives of the NRC staff and the licensee regarding the events, noted below, that occurred at the Indian Point Unit 2 Nuclear Power Plant and the associated staff findings, conclusions, and recommendations resulting from the evaluation of these events.
  - February 15, 2000 steam generator tube rupture event.
  - August 31, 1999 event involving reactor trip and loss of off-site power.

12:30 - 1:30

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

1:30 - 2:10

12) ~~1:00~~ - ~~1:30~~ P.M. — Siemens SRELAP-5 Appendix K Small-Break LOCA Code (Open/Closed) (GBW/PAB)

Report by the Chairman of the Thermal-Hydraulic Phenomena Subcommittee regarding the Siemens SRELAP-5 Appendix K code version for application to analysis of small-break loss of coolant accident (LOCA).

2:15-

Discuss the members revised compensation form

13) ~~1:30~~ - 2:30 P.M. — ~~Break and Preparation of Draft ACRS Reports~~

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

2:50

14) ~~2:30~~ - 7:00 P.M. — Discussion of Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

- 3:00-3:10 14.1) Performance-Based Regulatory Initiatives (JDS/NFD)
- 3:42-5:10 14.2) Proposed NRC Framework Document for Risk-Informing 10 CFR Part 50 and Associated Revisions to 10 CFR 50.44 Concerning Combustible Gas Control Systems, and Advance Notice of Proposed Rulemaking (10 CFR 50.69 and Appendix T) (WJS/GA/MTM)
- 3:35-3:40 14.3) Proposed Final Regulatory Guide (DG-1093) Endorsing NEI 97-04 Document on Design Basis (RLS/MWW)
- 5:25-5:50 14.4) AP1000 Pre-Application Review (TSK/SD)
- 14.5) Assessment of the Quality of PRAs (GA/MTM)

UCS letter
6:10-6:50

THURSDAY, AUGUST 31, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

15) 8:30 - 8:35 A.M. — Opening Remarks by the ACRS Chairman (Open) (DAP/JTL)

8:35 - 9:15 — Discuss UCS letter w/ D. Lochbaum, UCS

16) ~~8:35~~ - ~~8:45~~ A.M. — Reconciliation of ACRS Comments and Recommendations (Open) (DAP, et al./HJL, et al.)

9:15 - 9:30

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

9:30 - 10:20

17) ~~8:45~~ - ~~9:45~~ A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (DAP/JTL/HJL)

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

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

10:20 - 10:30 Break

18) ~~9:45~~ - ~~10:45~~ A.M. 1:30 - 2:30 Annual Report to the Commission on the NRC Safety Research Program (Open) (DAP/MME)

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

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

19) 11:00 - 12:00 Noon Miscellaneous (Open) (DAP/JTL)

Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

11:50 - 12:50

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

12:50 - 1:30 Discuss Ad Hoc S/C w/Members for topics & reviewer (see chart)

20) ~~1:00 - 4:00 P.M.~~ ~~Meeting with the NRC Commissioners on October 6, 2000 (Open) (DAP, et al./JTL, et al.)~~

~~Discussion of topics and preparation for meeting with the Commissioners scheduled for October 6, 2000.~~

7:00

21) ~~4:00~~ - ~~6:00~~ P.M. Discussion of Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

11:05-11:40 + 4:55-5:47 21.1) Performance-Based Regulatory Initiatives (JDS/NFD) Final

21.2) Proposed NRC Framework Document for Risk-Informing 10 CFR Part 50 and associated Revisions to 10 CFR 50.44 concerning Combustible Gas Control Systems, and Advance Notice of Proposed Rulemaking (10 CFR 50.69 and Appendix T) (WJS/GA/MTM) Final

3:40-4:00 21.3) Proposed Final Regulatory Guide (DG-1093) Endorsing NEI 97-04 Document on Design Basis (RLS/MWW) Final

4:10-4:55 21.4) AP1000 Pre-Application Review (TSK/SD) Final

11:40-11:50 21.5) Assessment of the Quality of PRAs (GA/MTM) Final

FRIDAY, SEPTEMBER 1, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

22) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (DAP/JTL)

23) 8:35 - ~~1:00~~ 12:05 P.M. Discussion of Proposed ACRS Reports (Open) - The Committee will continue its discussion of proposed ACRS reports as noted in item 22.

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

475TH ACRS MEETING
AUGUST 29-SEPTEMBER 1, 2000

<u>NRC STAFF</u> (August 29, 2000)
A. Levin, OCM/RAM
A. H. Hsia, OCM/NJD
J. Beall, OCM/EM
J. Munday, OCM
J. Calvo, NRR
T. Bergman, NRR
J. Williams, NRR
T. Reed, NRR
A. Markley, NRR
J. Fair, NRR
E. McKenna, NRR
S. Magruder, NRR
G. Imbro, NRR
G. Parry, NRR
M. Shuarbi, NRR
M. Cheok, NRR
J. Golla, NRR
S. West, NRR
C. Carpenter, NRR
E. Rodrick, NRR
G. Bagchi, NRR
K. Heck, NRR
D. Fischer, NRR
C. Ader, NRR
M. Rubin, NRR
G. Hsi, NRR
D. Mathews, NRR
D. Allison, NRR
C. Berlinger, NRR
J. Wilson, NRR
C. Grimes, NRR
T. Johnson, NMSS
R. Wescott, NMSS
H. VanderMolen, RES
F. Eltawila, RES
J. Mitchell, RES

<u>ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC</u> (August 29, 2000)
B. Christie, Performance Technology
C. W. Fleming, Winston & Strawn
A. Heymer, NEI
P. Negus, GE
N. Chapman, SERCH/Bechtel
R. Huston, Licensing Support Services
R. Bell, NEI
M. Knapik, McGraw-Hill
J. Weil, McGraw-Hill
C. Brinkman, Westinghouse
M. Corletti, Westinghouse

<u>NRC STAFF</u> (August 30, 2000)
A. Levin, OCM/RAM
A. Hsia, OCM/NJD
F. Eltawila, RES
N. Kadambi, RES
J. Muscara, RES
J. Mitchell, RES
G. Lanik, RES
D. Marksberry, RES
T. Bloomer, NRR
D. Matthews, NRR
R. Franovich, NRR
P. Kuo, NRR
J. Dozier, NRR
G. Bagchi, NRR
L. B. March, NRR
P. King, NRR
W. Liu, NRR
C. Ader, NRR
S. Lee, NRR
J. Strisha, NRR
S. Mithia, NRR
K. Ross, NRR
O. Tabatabai, NRR
C. Gratton, NRR
E. Benner, NRR
F. Gallardo, NRR
I. Jung, NRR

S. Long, NRR
R. Benedict, NRR
R. Schaaf, NRR
C. Grimes, NRR
S. Newberry, NRR
R. Landry, NRR
R. Caruso, NRR
S. Mitina, NRR
R. Lorson, NRR
J. Yerokim, Region I
B. Holian, Region I
J. Trapp, Region I
J. Talieri, Region I

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
V. Youngblood, Self
R. Janaty, PA Dept.
N. Chapman, SERCH/Bechtel
R. Huston, Licensing Support Services
P. Negus, GE
K. Sutton, Winston & Strawn
B. Youngblood, ISL, Inc.
C. Smith, ISL, Inc.
M. Wetterhahn, Winston & Strawn
J. Groth, ConEd, NY
J. McCann, ConEd, NY
J. Weil, McGraw-Hill

NRC STAFF (August 31, 2000)
G. Millman, EDO
R. Barrett, NRR
J. Mitchell, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
D. Lochbaum, UCS
J. Weil, McGraw-Hill

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

September 13, 2000

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

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

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

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

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

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

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

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

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

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

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

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

6) 3:30 - 4:30 P.M. Break and Preparation of Draft ACRS Reports (Open)
Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

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

- 7.1) Union of Concerned Scientists Report on Nuclear Plant Risk Studies (GA/MTM)
- 7.2) NEI 00-02, "Industry PRA Peer Review Process Guidelines" (GA/MTM)
- 7.3) Pressurized Thermal Shock Technical Bases Reevaluation Project (WJS/NFD)

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

- 8.1) Risk Informing 10 CFR 50 (WJS/MTM)
  - NEI Letter of January 19, 2000
  - Proposed Revision to 10 CFR 50.44 Concerning Combustible Gas Control System and Advance Notice of Proposed Rulemaking (10 CFR 50.69 and Appendix T)
- 8.2) Quality of PRAs (GA/MWW)
  - Assessment of the Quality of PRAs
  - ASME Standard on PRAs
- 8.3) Spent Fuel Pool Fire Safety Study (TSK/MME)
- 8.4) More Realistic (Best Estimate) Thermal-Hydraulic Codes (GW/PAB)
- 8.5) Status of ACRS Activities on License Renewals (MVB/NFD)

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

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

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

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

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

12:00 - 1:30 P.M. — ***LUNCH***

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

3:00 - 3:15 P.M. — ***BREAK***

13) 3:15 - 4:45 P.M. — GSI-168, Equipment Qualification (Open) (REU/AS)
14.1) Remarks by the Subcommittee Chairman
14.2) Briefing by and discussions with representatives of the NRC staff regarding the GSI-168, Equipment Qualification.

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

15) 5:30 - 5:50 P.M. — Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (DAP/JTL/HJL)
15.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
15.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

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

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

18) 6:30 - 7:30 P.M. — Discussion of Proposed ACRS Reports (Open)
Preparation of proposed ACRS reports on:
19.1) GSI-168, Equipment Qualification (REU/AS)
19.2) Union of Concerned Scientists Report on Nuclear Plant Risk Studies (GA/MTM)
19.3) NEI 00-02, "Industry PRA Peer Review Process Guidelines" (GA/MTM)
19.4) Pressurized Thermal Shock Technical Bases Reevaluation Project (WJS/NFD)

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

19)	8:30 - 8:35 A.M.	Opening Remarks by the ACRS Chairman (Open) (DAP/JTL)
20)	8:35 - 12:30 P.M.	Discussion of Proposed ACRS Reports (Open) - The Committee will continue its discussion of proposed ACRS reports as noted in Item 19.
21)	12:30 - 1:00 P.M.	Annual Report to the Commission on the NRC Safety Research Program (Open) (DAP/MME) Discussion of the current status of the review by the members of the topical areas previously assigned.
22)	1:00 - 1:30 P.M.	Miscellaneous (Open) (DAP/JTL) Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

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

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
475th ACRS MEETING
AUGUST 29-SEPTEMBER 1, 2000

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

MEETING HANDOUTS

AGENDA ITEM NO.	DOCUMENTS
1	Opening Remarks by the ACRS Chairman
	1. Items of Interest, dated August 29-September 1, 2000
2	Proposed Risk-Informed Revisions to 10 CFR Part 50
	2. Risk-Informed Part 50 Option 2 presentation by T. Reed, NRR [Viewgraphs]
	3. Risk-Informed 10 CFR 50.44 "Standard for Combustible Gas Control System in Light-Water-Cooled Power Reactors" presentation by RES [Viewgraphs]
	4. Risk-Informing NRC Regulations presentation by A. Heymar, NEI [Viewgraphs]
	5. ACRS Combustible Gas Control presentation by B. Christie, Performance Technology [Viewgraphs]
	6. Proceedings of PSAM-3 Meeting, Crete Greece (1997) "An Assessment of the Risk-Impact of Reactor Power Upgrade for a BWR-6 MARK-III Plant [Handout]
3	Causes and Significance of Design Basis Issues
	7. Draft Report: "Causes and Significance of Design-Basis Issues at U.S. Nuclear Power Plants" presentation by R. Lloyd, RES [Viewgraphs]
4	Proposed Final Regulatory Guide (DG-1093) Endorsing NEI 97-04 Document on Design Basis
	8. Clarifying the Definition of Design Bases presentation by S. Magruder, NRR, and R. Bell, NEI [Viewgraphs]
5	AP1000 Standard Plant Design
	9. Briefing on AP1000 Standard Plant Design presentation by J. Wilson, NRR [Viewgraphs]
	10. Letter from Westinghouse dated 8/28/00 Subject: AP1000 Phase 2 Review
9	Performance-Based Regulatory Initiatives
	11. High-Level Guidelines for Performance-Based Activities presentation by RES and ISL, Inc. [Viewgraphs]

12. High-Level Guidelines for Performance-Based Activities (Predecisional) Draft received by the ACRS on 8/25/00 from RES to the EDO Subject: High-Level Guidelines for Performance-Based Activities [Handout 9.1]

10 License Renewal Guidance Documents

13. Memorandum to ACRS Members from N. Dudley, dated 8/29/2000, Subject: ACRS Review Plans for License Renewal Guidance Documents [Handout]
14. License Renewal Generic Activities presentation by S. Lee, NRR [Viewgraphs]

11 Operating Events at Indian Point Nuclear Power Plant Unit 2

15. Unit 2 Electrical Distribution System [diagram]
16. Event 1: Reactor Trip and Partial Loss of Vital Power, 8/31/99; Event 2: Steam Generator Tube Failure, 2/15/00 [Viewgraphs]

16 Reconciliation of ACRS Comments and Recommendations

17. Reconciliation of ACRS Comments and Recommendations [Handout #16.1]

17 Report of the Planning and Procedures Subcommittee

18. Future Activities, October 5-7, 2000 [Handout 17.2
19. Final Draft Minutes of Planning and Procedures Subcommittee Meeting - August 28, 2000 [Handout #17.1]

18 Discussion of SGTR DPO S/C and F/C, Members with Topic & Reviewer

20. DPO Plan [Viewgraph]

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

21. Ground Rules presentation by D. Powers, Chairman, ACRS [Viewgraphs]

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS

2 Activities Associated with Risk-Informing 10 CFR Part 50

1. Table of Contents
2. Proposed Schedule
3. Status Report, dated August 29, 2000
4. Draft Commission paper on risk-informing 10 CFR Part 50 (Option 3)
5. SRM dated January 31, 2000
6. SRM dated February 3, 2000
7. SRM dated April 5, 2000
8. Letter dated April 18, 2000 from Steven D. Floyd, NEI, to Thomas L. King, RES, Subject: Industry Comments on SECY-00-0086 and draft NRC report on risk-informing 10 CFR 50.44
9. Report dated October 12, 1999, from Dana A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, NRC, Subject: Proposed Plans for Risk-Informing 10 CFR Part 50

3 Causes and Significance of Design-Basis Issues at U.S. Nuclear Power Plants

11. Table of Contents
12. Proposed Schedule
13. Status Report
14. Draft Report, "Causes and Significance of Design-Basis Issues at U.S. Nuclear Power Plants," dated May 2000

4 Proposed Final Regulatory Guide (DG-1093) Endorsing NEI 97-04 Document on Design Basis

15. Table of Contents
16. Proposed Schedule
17. Status Report
18. Memorandum to J. Larkins, ACRS, from D. Mathews, NRR, Subject: Final Regulatory Guide 1.xxx (DG-1093), "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases," dated August 7, 2000
19. Memorandum to C. Carpenter, NRR, from S. Magruder, NRR, Subject: Summary of July 27, 2000 Meeting with NEI on Revision to NEI 97-04 on the Definition of 10 CFR 50.2 Design Bases," dated July 31, 2000

5 Pre-Application (Phase 1) Review of the AP1000 Design

20. Proposed Schedule
21. Status Report
22. Memorandum dated May 31, 2000, from M. Corietti, Westinghouse, to

Document Control Desk (Attention J. Wilson), NRR, Subject: AP1000 Pre-Application Review Items

23. Memorandum dated July 27, 2000, from S. Collins, NRR, to W. Cummins, Westinghouse, Subject: AP1000 Pre-Application Review - Phase One
24. Memorandum dated June 21, 2000, from J. Larkins, ACRS, to W. Travers, EDO, Subject: AP1000 Pre-Application Review

9 Performance-Based Regulatory Initiatives

25. Table of Contents
26. Proposed Schedule
27. Status Report
28. Draft Letter dated June 9, 2000, from D. Powers, ACRS, to W. Travers, EDO, Subject: Proposed High-Level Guidelines for Performance-Based Activities [Predecisional]
29. Memorandum dated June 8, 2000, from G. Apostolakis, ACRS, to J. Sieber, ACRS, Subject: Performance-Based Activities
30. US Nuclear Regulatory Commission, Draft Commission Paper, "High Level Guidelines for Performance-Based Activities" received 8/9/2000

10 License Renewal Guidance Documents

31. Table of Contents
32. Proposed Schedule
33. Status Report
34. US Nuclear Regulatory Commission, Draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, 4/21/2000, Table of Contents and Introduction
35. US Nuclear Regulatory Commission, Draft Generic Aging Lessons Learned (GALL) Report, 12/6/99, Table of Contents and Introduction
36. NEI 95-10 [Revision 2], "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," August 2000
37. Guidance Documents on License Renewal www.nrc.gov/NRC/REACTOR/LR/guidance.html, 5/10/2000

11 Operating Events at Indian Point 2

38. Table of Contents
39. Proposed Schedule
40. Status Report
41. List of Related Documents
42. Attachments (selected pages included in file)
    Reactor Trip with Complications, pages 10-56
    — LER 247/99-15, dated 9/30/99, regarding Reactor Trip, ESF Actuation, Entry into TS 3.0.1, and Notification of an Unusual Event
    — Letter, NRC to ConEd. Dated 10/19/99, transmitting AIT Inspection

Report (IR) 247/99-08, Reactor Trip with Complications. Enclosures include the AIT Charter and the Briefing Slides for the 9/27/99 Exit Meeting

— Letter, NRC to ConEd, dated 12/21/99, transmitting IR 247/99013 - Results from the Follow Up Inspection to the AIT
— Letter, NRC to ConEd, dated 1/5/00, transmitting IR 05000247/99014- Results of the Enforcement Followup Inspection to the AIT

<u>Steam Generator Tube Failure, Pages 57-127</u>

— LER 247/001-01, dated 3/17/00, regarding Manual Reactor Trip Following Steam Generator Tube Rupture
— Letter, NRC to ConEd, dated 4/28/00, transmitting IR 247/2000-002, NRC Augmented Inspection Team - Steam Generator Tube Failure. Enclosures include AIT Charter and Briefing Slides for the 3/29/00
— Letter, NRC to ConEd, dated 5/23/00, IP2 Agency Focus Plant Status
— NRC Information Notice 2000-09, Steam Generator Tube Failure at IP2, dated 6/28/00
— Letter, NRC to ConEd, dated 7/10/00, transmitting IR 247/2000-07 - NRC Augmented Inspection Team Follow-up, Steam Generator Tube Failure
— Letter, NRC to ConEd, dated 7/27/00, transmitting Preliminary Results of NRC Special Inspection 247/200010 - Steam Generator Tube Failure

12 <u>Report on Thermal Hydraulics Phenomena Subcommittee Meeting, Siemens S-RELAP5 Appendix K Small-Break LOCA Code</u>

43. Table of Contents
44. Project Status Report
45. Working Copy of Minutes of August 8-9, 2000 Thermal-Hydraulic Phenomena Subcommittee Meeting dated 8/23/00
46. Excerpt from Minutes of March 15, 2000 Thermal Hydraulic Phenomena Subcommittee Meeting on Siemens S-RELAP5 Thermal-Hydraulic Code, dated 4/17/00.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

475th FULL COMMITTEE MEETING

August 29, 2000

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Russ Bell	NEI
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475 FULL COMMITTEE MEETING

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August 30, 2000

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475 FULL COMMITTEE MEETING

August 31, 2000

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ITEMS OF INTEREST

475th ACRS MEETING

AUGUST 29 - SEPTEMBER 1, 2000

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
475th MEETING
AUGUST 29 - SEPTEMBER 1, 2000

Remarks of
Jeffrey S. Merrifield
Commissioner
United States Nuclear Regulatory Commission

at the

American Nuclear Society's
2000 Utility Working Conference

August 7, 2000
Amelia Island, Florida

Good Morning. Thank you very much for the opportunity to speak to you today. Given the dynamic state of the electric industry in the U.S., I can't think of a more appropriate focus for this conference than "Managing the Business of Nuclear Power". As I have said on many occasions, today, the outlook for nuclear power is arguably the brightest its been since the Three Mile Island accident. Competitive market forces have led to a resurgence of nuclear power by forcing dramatic improvements in the manner in which nuclear plants are managed and operated. Licensees have improved operator training, made significant process improvements, developed sound maintenance and corrective action programs, shortened refueling outages, and as a result, significantly increased both the safety and generation of power in the nuclear fleet. Plants are operating better than ever before, with forced outage rates at an all time low and capacity factors at an all time high. By almost any measure, most of our licensees are doing an excellent job of managing the business of nuclear power in a safe manner.

Managing the business of nuclear regulation is my business and that of the Commission. The dynamic state of the electric industry is also creating many challenges for the NRC. The consolidation of nuclear utilities through mergers, plant sales and the formation of multi-plant operating companies has resulted in an influx of license transfers. Industry interest in license renewal has never been greater and projections indicate that the NRC will face a daunting number of license renewal applications in the coming years. Also, the competition inherent with electric industry deregulation is increasing the nuclear industry's focus on reducing the cost of regulation. This has challenged the NRC on two fronts. First, given that essentially 100% of the NRC's budget is recovered from our licensees, we are being challenged to reduce our costs and make significant strides in the areas of financial responsibility and accountability. Second, we are being challenged to reduce unnecessary regulatory burden on licensees and to risk-inform our regulations. We are being asked to meet these challenges at the same time we are challenging ourselves to become more responsive to our stakeholders and to enhance public confidence. I believe we are up to these challenges. I feel very good about the NRC's ongoing reform efforts and believe that most of our stakeholders recognize that the NRC is doing a much better job managing the business of nuclear regulation.

The nuclear industry, the NRC staff, and our many stakeholders deserve credit for the significant improvements that have been made in the way in which licensees manage the business of nuclear power and the way in which the NRC manages the business of nuclear regulation. However, this is certainly no time for any of us to celebrate. Dynamic environments demand dynamic performance expectations. If we are going to be top-performing organizations in the dynamic environment we undoubtably will face, we must ensure that the accomplishments we celebrate today only serve to raise our expectations for tomorrow. If the history of the nuclear industry has taught us anything, it is that those content with the status quo quickly become faint images in the rear view mirrors of those that recognize that success must be redefined every time we think we have achieved it.

In light of the fact that the NRC recently celebrated its 25th anniversary, I'd like to discuss managing the businesses of nuclear power and nuclear regulation in a historical context. I recently read "A Short History of Nuclear Regulation, 1946-1999" by Sam Walker, the NRC's historian. It is an informative account of the evolution of the commercial nuclear power industry and the regulation of that industry. As a history buff, I found that Mr. Walker's account reinforced the notion that history has a tendency of repeating itself. I encourage you to read this account as I think you will be amazed that many of the challenges and opportunities facing the nuclear industry and the NRC today, are the same challenges and opportunities that faced industry pioneers in the 50s and 60s. I'll draw from Mr. Walker's historical account to make my point.

Licensing Bottlenecks

I will refer to the first such challenge as licensing bottlenecks. During the late 1960s, the nation's utilities rapidly increased their orders for nuclear power stations, participating in what Philip Sporn, past president of American Electric Power Service Corporation, described in 1967 as the "great bandwagon market." The sudden arrival of commercially competitive nuclear power placed unprecedented demands on the Atomic Energy Commission's (AEC) regulatory staff. The flood of applications inevitably caused licensing delays because the staff simply lacked the resources to get the job done. The growing backlog drew bitter complaints from utilities applying to build plants. Many in the industry openly criticized the AEC's licensing process and believed that if the delays continued, the bright future once predicted for nuclear power would be lost. One utility executive quoted in Mr. Walker's historical account called the licensing process "a modern day Spanish Inquisition" carried out by "AEC engineers, scientists, and consultants who have no serious economic discipline". The AEC attempted to streamline its licensing procedures but found it impossible to reduce review time or to satisfy the licensing demands of the industry.

The NRC faces a similarly ominous licensing challenge in 2000. About 10% of the existing U.S. nuclear plant licenses will expire by the end of 2010, and more than 40% will expire by 2015. While the economics associated with new plant construction remain uncertain, nuclear power's favorable environmental and economic position relative to fossil plants, and a much more stable and disciplined regulatory environment, have fueled remarkable interest in license renewal. Earlier this year, the NRC renewed the Calvert Cliffs and Oconee licenses for another 20 years. We currently have the renewal applications for Southern Company's Hatch plant, and Entergy's Arkansas Nuclear One plant under review. We expect to receive more than 20 applications for license renewal over the next 5 years. Based on my discussions with industry executives, I am hard-pressed to identify more than a handful of currently operating plants that may not seek to renew their licenses.

The NRC can be very proud of the fact that we met or beat every milestone we set for the Calvert Cliffs and Oconee license renewals. However, as I stated earlier, we must ensure that the accomplishments we celebrate today only serve to raise our expectations for tomorrow. For the agency to successfully meet the unprecedented demands represented by the new "great bandwagon market" associated with license renewal, our review process must become more efficient and more timely. I believe there are 2 ways to get there. First, we must apply the lessons learned from the first two applications. Second, it is imperative that we promptly build a regulatory infrastructure - and what I mean by infrastructure are things like the Generic Aging Lessons Learned (GALL) report and Standard Review Plan - to support thorough, consistent, disciplined, and timely reviews in the future. Sacrificing our regulatory infrastructure for the sake of saving resources or shaving a few weeks off of our ongoing reviews would be shortsighted. For me, the bottom line is quite simple. We must carefully plan and budget our resources so that we don't fall victim to our own success in the area of license renewal. We must dedicate the resources necessary to build a robust and predictable regulatory infrastructure while at the same time providing the resources necessary to perform ongoing reviews in a thorough and even more timely manner. It would simply be irresponsible for the NRC to allow itself to repeat the problems that plagued our licensing process during the 60s and 70s.

Economies of Scale

The second such challenge facing the NRC and our licensees involves the aggressive pursuit of economies of scale. During the 1960s, there were several important considerations that convinced a growing number of utilities to buy nuclear plants. One was the spread of power pooling arrangements among utilities, which encouraged the construction of larger generating stations by easing fears of excess capacity and over-expansion. A utility with extra or reserve power could sell it to other companies through interconnections. Utility executives believed that large nuclear plants would produce economies of scale that would cut capital costs per unit of power and improve efficiency. This helped to overcome a major disadvantage of nuclear power relative to fossil fuel - the heavy capital requirements for building nuclear plants. This quest for economies of scale resulted in the output of plants leap-frogging from the 100 to 500 to 800 to the 1000 electrical megawatt range.

Today, the economic realities of a deregulated electric industry are driving industry leaders to once again place a high priority on economies of scale. However, today's economies of scale look quite different than those of the 60s. While licensees continue to achieve economies through power uprates, the primary focus of the industry has clearly changed from larger plants to larger nuclear fleets achieved through license transfers. The PECO/Unicom merger, the acquisitions by Amergen and Entergy, and the Nuclear Management Company formed by several midwest licensees, all reflect the financial importance being placed on large nuclear fleets by our licensees. It is my hope that these transfers will provide a tremendous opportunity to further improve the operational performance of the plants.

License transfers represent a significant licensing challenge for the NRC. From my perspective, the NRC's primary responsibility in this area is to ensure that the economies of scale never come at the expense of public health and safety. However, I strongly believe that we owe it to the American people to carry out this responsibility in a manner that does not unnecessarily impede market forces. We simply must provide the resources and the management oversight necessary to ensure that our staff reviews license transfers in a thorough, timely, and disciplined manner.

To our licensees I say, in your quest to get more value from your generating assets, don't jeopardize their future. Proceed responsibly - ensure that your technical and financial analyses are sound, your staff remains focused on operational performance and safety, and your business decisions are not undermined by false economics.

As consolidation in the ownership of nuclear plants continues, the few large companies operating these plants must not become insular. They must continue to recognize the value of looking outside of their organization for solutions, and of sharing information outside of their organization for the common good of the industry. As I said at the Regulatory Information Conference in March, for those who are so bold as to believe that all of the nuclear industry's solutions, all of its best practices, all of its operating experience, lie within your organization, I ask you this: "Are you bold enough to stake your assets on it? I hope the answer is no.

Unnecessary Regulatory Burden

Eliminating unnecessary regulatory burden is another challenge and another opportunity faced by both the early pioneers of the nuclear industry as well as today's industry leaders.

The AEC's fundamental objective in drafting regulations was to ensure that public health and safety were protected without imposing overly burdensome requirements that would impede industrial growth. Commissioner Willard Libby articulated an opinion common among AEC officials when he remarked in 1955, "Our great hazard is that this great benefit to mankind will be killed aborning by unnecessary regulation." Other proponents of nuclear development shared those views. They realized that safety was indispensable to progress, as an accident could destroy the fledgling industry or at least set it back many years. At the same time, they worried that regulations that were too restrictive or inflexible would discourage private participation and investment in nuclear technology. The inherent difficulty the AEC faced was distinguishing between essential and excessive regulations.

As we enter the new millennium, eliminating unnecessary regulatory burden remains a major challenge for the NRC and the nuclear industry. This challenge is closely linked to another regulatory challenge we refer to as risk-informing our regulations. Some of our critics refer to our efforts in these areas as "regulatory retreat". In fact, at a recent Commission meeting, Jim Riccio from Public Citizen referred to our efforts as "the deregulation of nuclear safety standards". Now, while I respect Mr. Riccio for voicing his opinions, I strongly disagree with both assertions. I believe our initiatives in these areas in no way reflect less of a commitment to safety, but instead reflect a more informed commitment to safety. The NRC is simply capitalizing on a wealth of operating experience, extensive research, and well-developed risk insights to bring greater realism to our regulatory framework. Our initiatives should allow both licensees and the NRC to focus more attention on the truly risk-significant aspects of the plants and spend less time on regulatory burdens that contribute little or nothing to safety. They will also allow the NRC to utilize our limited resources more effectively and efficiently.

I and the other Commissioners remain committed to reducing unnecessary regulatory burden and to risk-informing our regulations. However, as we proceed along that course, neither the NRC staff nor our licensees should lose sight of the following 4 points:

1. First, the key word in the term "unnecessary regulatory burden" is "unnecessary". Regulation is by its very nature burdensome. Regulation that carries with it no burden, likely also carries with it no value. In order to achieve its mission, the NRC will impose the appropriate level of regulation it believes is necessary to protect public health and safety and the environment, irrespective of its popularity. Nonetheless, both the NRC and our licensees have a responsibility to the American people to understand where the line between necessary and unnecessary regulation is, and to respect it.

2. Second, our licensees must accept that risk-informed regulation is a double-edged sword. While our move toward risk-informing our regulations will likely provide many opportunities to reduce unnecessary regulatory burden, it would be foolish to think that risk-insights won't also identify areas where more regulation is needed. As long as the industry responsibly accepts the sharp edge of the sword representing additional regulatory burden, the NRC will continue down the path of risk-informing our regulations. Should that edge become dulled by irresponsible industry opposition, the integrity of risk-informed regulation will be compromised, and NRC progress will come to a screeching halt.

3. Third, risk-informed regulation should bring with it the promise of greater regulatory stability. Reactionary regulation is bad regulation. The beauty of a truly sound risk-informed regulatory framework is that it should be immune to the regulatory pendulum swings that have marred this industry's past. From my perspective, an unstable regulatory environment is in and of itself unnecessarily burdensome and is not in the best interests of the public, our licensees, or our staff.

4. Finally, as I stated at the Regulatory Information Conference, we must move forward deliberately, yet cautiously, in the area of risk-informed regulation. While I am optimistic that we can use risk insights to improve many aspects of Part 50, I am not yet convinced that there is sufficient stakeholder support to justify the cost of making a wholesale change to Part 50. Although I am willing to provide the resources necessary to take the important initial steps, I will not support additional resources if there is not sufficient interest in using these alternative regulations.

In summary, I agree with AEC Commissioner Willard Libby's position that the commercial nuclear power industry should not be killed by unnecessary regulation. I am committed to ensuring that this does not happen. I am equally committed to ensuring that the commercial nuclear power industry is not killed by the equally lethal hazards associated with insufficient regulation or a less than credible regulator.

High-Level Waste

History is also repeating itself in the area of high-level waste.

An issue that undermined confidence in the AEC and the nuclear industry in the early 1970s was the AEC's approach to high-level radioactive waste disposal. The growth of the nuclear power industry made the safe disposal of spent fuel rods and other waste materials an increasingly urgent matter. The AEC had investigated means of dealing with reactor wastes for years, but had not found a solution to the problem. As early as 1957, a scientific consensus had concluded that deep underground salt beds were the best repositories. In 1970, in response to increasing expressions of concern about the lack of a policy for high-level waste disposal from scientific authorities, members of Congress, and the press, the AEC announced that it would develop a permanent repository for nuclear waste in an abandoned salt mine near Lyons, Kansas. However, the AEC had not conducted thorough geologic and hydrologic investigations, and the suitability of the site was soon challenged. The uncertainties about the site generated a bitter dispute between the AEC and Congress. It ended in 1972 in great embarrassment for the AEC when the reservations of those who opposed the Lyons location proved to be well-founded.

The disposal of high-level radioactive waste remains a major challenge facing the nuclear industry. As you know, in April, President Clinton vetoed high-level waste legislation sent to him by Congress. Given that we are in an election year, I certainly do not expect any other waste legislation to move forward during this session of Congress. While it would be inappropriate for me to comment on the merits of that decision, I doubt that many would dispute that the nuclear industry is bearing the burden for the federal government's failure to provide a repository for high-level radioactive waste.

The NRC is responsible for licensing the repository after determining whether DOE's proposed repository site and design comply with EPA's environmental standards and with the NRC's implementing regulations found in 10 CFR Part 60. Currently, DOE is scheduled to issue its final Environmental Impact Statement for the Yucca Mountain site in early FY 2001 and its license application in early 2002. I am proud to say that the NRC has met all of its commitments to date and stands ready to fulfill its role associated with Yucca Mountain.

There is a continuing debate between ourselves and the EPA regarding appropriate environmental standards for protection of human health at Yucca Mountain. Although Congress gave EPA the responsibility for setting these standards, I and the other Commissioners have been very active in expressing our views about this matter to Congress. While the NRC believes that a 25 millirem all pathways standard is appropriate, the EPA disagrees stating that it should be 15 millirem with a separate standard for groundwater. Although logical people can disagree on these issues, the EPA is the only regulatory agency in the world that believes there should be a separate groundwater standard. I think that fact speaks volumes. I cannot overstate the national and international implications of this matter or the importance the Commission places on them.

Finally, I appreciate the fact that discussions about long-term milestones associated with Yucca Mountain are of little consolation to those of you facing the imminent loss of spent fuel pool storage capacity and the significant costs associated with dry cask storage. I assure you the Commission has a clear understanding of the spent fuel situation in the United States and is committed to ensuring that safe, technically- sound casks are certified in a prompt and thorough manner. While we have been successful in improving the timeliness and predictability of our cask certification process, we need to achieve further process efficiencies and resolve the generic technical issues like credit for high burnup fuel. Simply put, this is a regulatory responsibility in which we must not fail.

<u>**Public Confidence**</u>

Sam Walker's historical perspective clearly illustrated the swings in public perception and public confidence that have occurred throughout the history of the commercial nuclear power industry. In the early days of nuclear power development, public attitudes toward commercial use of the technology were highly favorable. Press coverage of nuclear power was also overwhelmingly positive. For example, an article in National Geographic in 1958, concluded that "abundant energy released from the hearts of atoms promises a vastly different and better tomorrow for all mankind." In the early 60s, the public became more alert to and anxious about the hazards of radiation, largely as a result of a major controversy over radioactive fallout from nuclear weapons testing. For the most part, however, during the 60s and to some extent the early 70s, America's support of nuclear power grew as the public viewed nuclear power as a potential solution to environmental concerns and the energy crisis. Since that time, America's confidence in nuclear power has been shaken by events like the Browns Ferry fire of 1975, the Three Mile Island accident of 1979, the Chernobyl accident of 1986, the plant licensing debacles of the 80s and early 90s, and finally the Millstone saga of the 90s. Despite these events, recent polls show that the nation's confidence in the safety of nuclear power is again on an upswing.

In his book entitled "Containing The Atom", Sam Walker quotes former AEC Chairman James Schlesinger as stating that although it "should be difficult to be other than bullish" about the long-term prospects for nuclear power, the pace of development would depend on two variables: "first, the provision of a safe, reliable product; second, achievement of public confidence in that product." While Mr. Schlesinger's comments were made in 1971, there is no question in my mind that they hold true today.

From my perspective, while the growing environmental concerns associated with fossil energy sources may have brought nuclear power back into the energy debate in the U.S. , the resurgence in public confidence that nuclear power is enjoying would not have been possible were it not for the industry's improved safety performance over the last few years. Public confidence must be earned, and the improved overall performance within the fleet has contributed to a demonstrable increase in public confidence. Nonetheless, let's face it, this confidence is fragile and thus the industry must always be vigilant in protecting it. The best way to do that is by operating plants safely, responsibly, and efficiently. The industry cannot tolerate performance lapses like those that have occurred at Indian Point 2 over the last year. Performance lapses like these not only undermine public, Congressional, and to some extent, regulatory confidence in Indian Point 2, but they also have the spillover effect of eroding confidence in each of the other 102 reactors operating throughout the U.S.

In its quest to improve public confidence, the nuclear industry must not lose sight of the clear nexus between a strong industry and a strong regulator. The industry should not underestimate the value of having a regulator that is tough, competent, and independent. I know that there are some in the industry who continue to call for further reductions in our staff, and others who call for us to dramatically reduce the scope of our regulations. I caution those individuals to be careful about what you ask for. The American public will simply not support or even tolerate a nuclear industry that it views is not overseen by a strong, credible regulator.

I believe the NRC and the nuclear industry have also underestimated the importance of communicating effectively with the public. From my perspective, many in the industry have done a poor job communicating with the public and as a result, public confidence has suffered. They have been reactive in their approach to communications, and have not taken the time to educate the public about nuclear power or to keep them informed about activities at the plants. The industry only has to look at examples such as Carolina Power & Light's Brunswick plant to understand the economic, social, and political benefits associated with effective public communications. During a recent visit to Brunswick, I met with a large group of local government and business leaders and was surprised by the amount of public support that the plant enjoys. It was clear to me that CP&L's efforts to reach out to the neighboring community and its leaders have resulted in significant tangible and intangible benefits associated with a high level of public confidence and trust. It is in the industry's best interests to learn from examples such as this and recognize that maintaining a continuing dialogue with the public makes good business sense.

Poor communication by the NRC has also served to erode public confidence in the agency and the nuclear industry. In the past, the NRC approached public confidence in much the same way the Maytag repairman approaches his job. We were passive in our communications with the public. We allowed our critics to define what our agency was, what its actions meant, and how these actions should be perceived. As a result, the agency frequently found itself in the difficult position of playing catch-up. This approach had its roots with the old AEC. The AEC's organizational philosophy simply did not recognize a role for the agency in enhancing public confidence. The agency paid a very heavy price for this passive approach.

I believe the NRC must become more proactive and forthright in its communications. We must be the first to communicate with the public about important regulatory decisions and must clearly articulate the reasoning behind them. We should change our organizational philosophy so that we no longer allow inaccurate or misleading assertions in the public arena to go unaddressed. When spent fuel casks are referred to as mobile Chernobyl's, I think we should clearly present the true basis for why we feel our regulations will assure that dry cask storage is safe. When opponents of the new oversight process or our decision on N+1 label them as regulatory retreat, we must accurately and promptly respond so that the public is not left with a mistaken understanding of our programs. How will the NRC ever enhance public confidence if we remain passive in the public arena? We simply won't. I sincerely believe that if we have a true and defendable story to tell, it is irresponsible for us not to tell it - a disservice to our licensees, our staff, and, most importantly, the American people.

Conclusion

In conclusion, managing the businesses of nuclear power and nuclear regulation brings with it many challenges and opportunities. In order for the nuclear industry and the NRC to successfully meet these challenges and seize these opportunities, our visions of the future must benefit from the lessons of the past.

George Bernard Shaw once said, "If history repeats itself, and the unexpected always happens, how incapable must man be of learning from experience." The nuclear industry and the NRC must learn from history so that we do not fall victim to the unexpected. To do otherwise would be irresponsible. As the industry reaps the benefits associated with improved performance, and as the NRC and the industry pursue greater efficiencies and regulatory reform, we must learn from the lessons of the past and be careful not to role back the safety improvements made over the last 20 years. We must ensure that the lessons of the past do not get "reformed out" or "budgeted out" of our programs. We cannot allow ourselves to lose sight of the fact that the performance and safety improvements that both the industry and the NRC are enjoying today came at a very high price -- a price that we cannot afford to repeat.

I want to thank you for giving me this opportunity to share some of my thoughts this morning. At this time, I'd be pleased to address any questions you may have.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs **Telephone: 301/415-8200**

Washington, DC 20555-001 **E-mail: opa@nrc.gov**

Web Site: http://www.nrc.gov/OPA

No. 00-120 August 4, 2000

NRC APPROVES TRANSFER OF OPERATING LICENSES FOR 20 NUCLEAR PLANTS

The Nuclear Regulatory Commission has approved the transfer of operating licenses for 20 commercial nuclear power plants from Commonwealth Edison and PECO Energy Company to Exelon Generation Company.

Exelon is being formed in connection with the proposed merger of Unicom Corporation (Unicom), the parent of Commonwealth Edison, and PECO.

The 13 Commonwealth Edison units affected are all located in Illinois. They are **Braidwood** 1 and 2, near Joliet; **Byron** 1 and 2, near Rockford; **Dresden** 1 (permanently shut down) and Units 2 and 3, near Morris; **LaSalle** 1 and 2, near Ottawa; **Quad Cities** 1 and 2, near Moline; and the permanently shut down **Zion** 1 and 2, in Zion.

The PECO units affected are **Peach Bottom** 1 (permanently shut down) and Units 2 and 3, near Lancaster, Pennsylvania; **Limerick** 1 and 2, in Limerick, Pa. Also affected are **Salem** 1 and 2, in Hancocks Bridge, New Jersey, which are partially owned by PECO but operated by Public Service Electric & Gas Co.

Last December PECO and Commonwealth Edison submitted applications to the NRC requesting approval for the license transfers. The key issues considered by the NRC's technical staff included decommissioning funding, insurance and Exelon's technical and financial qualifications.

Notices of the requests for approval and for an opportunity for a hearing were published in the *Federal Register* on March 9. The Commission received no comments or hearing requests. The technical staff's approval becomes effective immediately.

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NRC NEWS

UNITED STATES NUCLEAR REGULATORY COMMISSION

OFFICE OF PUBLIC AFFAIRS, REGION I

475 Allendale Road, King of Prussia, Pa. 19406

No. I-00-58 July 27, 2000

CONTACT: Diane Screnci (610)337-5330/ e-mail: dps@nrc.gov
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NOTE TO EDITORS:

The Nuclear Regulatory Commission staff has issued a letter to Consolidated Edison Company of New York detailing the preliminary findings of a special inspection to review the cause of the February 15 steam generator tube failure at Con Ed's Indian Point 2 nuclear power plant in Buchanan, N.Y. The letter is attached.

Separately, the NRC today issued an amendment to the Indian Point 2 technical specifications. The amendment allows Con Ed, among other things, to operate with the containment recirculation filters and charcoal adsorbers removed. The request for the amendment - submitted by ConEd in November 1999 - was intended to take advantage of updated research findings on estimated public radiation doses from reactor accidents. Copies of this amendment are available from the NRC's electronic reading room at accession number ML003727500. Copies are also available from the NRC's Office of Public Affairs.

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July 27, 2000

Mr. A. Alan Blind
Vice President - Nuclear Power Consolidated Edison Company of New York, Inc.
Indian Point 2 Station
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: PRELIMINARY RESULTS OF NRC SPECIAL INSPECTION 50-247/2000010- STEAM GENERATOR TUBE FAILURE

Dear Mr. Blind:

This letter transmits the preliminary results of a special inspection conducted to review the cause of the February 15, 2000, steam generator tube failure at your Indian Point 2 reactor facility. We are providing these preliminary results in advance of the full inspection report since the results have the potential to influence ongoing assessments of the most recent steam generator inspections and root cause analyses. These results are subject to NRC management final review. The overall significance determination for these findings remains under evaluation.

The NRC team members included personnel from the Office of Nuclear Reactor Regulation and Region I, as well as NRC-contracted specialists in steam generator eddy current testing. On July 20, 2000, the team leader discussed the preliminary results with you, Messrs. J. Groth and J. Baumstark, and other members of the Con Edison staff.

The team concluded that the overall technical direction and execution of the 1997 steam generator inspection program were deficient in several respects. Con Edison did not recognize and take appropriate corrective actions for significant conditions adverse to quality that affected eddy current data collection/analysis. This increased the likelihood that detectable flaws in low row U-bend tubes were not identified.

More specifically, Con Edison did not:

1. take appropriate corrective actions following identification of a new and significant tube degradation mechanism, i.e., inside diameter (ID) primary water stress corrosion cracking (PWSCC) at the apex of a low row U-bend tube. Operating experience indicates that apex cracking is more likely to result in tube failure than other U-bend cracks. The 1997 steam generator inspection program did not fully assess the implications of this new degradation mechanism and adjust, as appropriate, the inspection methods and analyses.

2. recognize the significance of, and fully evaluate, the flaw masking effects of the high noise encountered in the eddy current signal. In the case of the steam generator tube that failed, the magnitude of the noise was a problem that negatively impacted the probability of detection. The data analysis techniques were not adjusted to compensate for the noise to improve the identification of a flaw signal and ensure the appropriate probability of detection, particularly when conditions which increased susceptibility to tube degradation existed.

3. appropriately establish procedures and implement practices to address the potential for hour-glassing in the upper support plate flow slots. Hour-glassing in this location is indicative of increased stresses on the steam generator tubes, which increase the likelihood of tube cracks. Further, the potential existence and impact of upper support plate hour-glassing were not assessed following the identification in 1997 of eddy current probe restrictions at the upper support plate and the identification of a PWSCC indication at the apex of a steam generator tube.

4. ensure the use of properly qualified eddy current techniques. The U-bend plus-point eddy current probe was not set-up properly for use. Specifically, you did not use the proper calibration standard and phase rotation specified by the EPRI technique qualification standard. While this issue had a small effect on the probability of detection of low row U-bend indications, it was another example that reflected the deficiencies in the overall technical direction and execution of the 1997 steam generator program.

The team also concluded that Con Edison's root cause analysis for the tube failure, dated April 14, 2000, did not sufficiently address the above described deficiencies. While the root cause analysis attributed the tube failure to a flaw that was obscured by eddy current signal noise, it did not identify, nor address, deficiencies in the processes and practices that were implemented for the 1997 steam generator inspection.

At the exit meeting, Con Edison disagreed with the team's preliminary findings. Specifically, it is our understanding that Con Edison's position is that: 1) all 1997 steam generator inspection requirements were met; 2) the team had not identified any specific requirements, standards or guidelines that were not met; 3) no specific noise criteria existed relative to the probability of detection of flaws using eddy current examination; 4) the PWSCC indication was expected and no additional assessment was warranted after this discovery; 5) the root cause submitted was complete and accurate; and, 6) the NRC team's preliminary findings are not in agreement with NRC Inspection Report 50-247/97007, dated July 16, 1997. Many of these viewpoints had been discussed during the inspection. The NRC will continue to consider these points as part of our established regulatory process, which includes the significance determination process and inspection report finalization.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room). Should you have any questions regarding this letter, please contact Mr. David C. Lew at 610-337-5120.

Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 05000247

License No. DPR-26

cc w/encl:

J. Groth, Senior Vice President - Nuclear Operations

J. Baumstark, Vice President, Nuclear Power Engineering

J. McCann, Manager, Nuclear Safety and Licensing

B. Brandenburg, Assistant General Counsel

C. Faison, Director, Nuclear Licensing, NYPA

J. Ferrick, Operations Manager

C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law

P. Eddy, Electric Division, Department of Public Service, State of New York

T. Rose, NFSC Secretary

F. William Valentino, President, New York State Energy Research and Development Authority

J. Spath, Program Director, New York State Energy Research and Development Authority

County Clerk, West Chester County Legislature

Westchester County Executive

Putnam County Executive

Rockland County Executive

Orange County Executive

T. Judson, Central NY Citizens Awareness Network

M. Elie, Citizens Awareness Network

NRC NEWS

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No. 00-112 July 14, 2000

NRC INVITES PUBLIC TO SUBMIT NOMINATIONS FOR ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The U.S. Nuclear Regulatory Commission (NRC) is seeking qualified candidates for appointment to two vacancies on its Advisory Committee on Reactor Safeguards (ACRS).

The ACRS was established by Congress to provide the NRC with independent expert advice on matters related to licensing and the safety of existing and proposed nuclear power plants. The Committee's work currently emphasizes safety issues associated with the operation of 103 commercial nuclear power plants in the United States; the pursuit of a risk-informed, and performance-based regulatory approach; review of license renewal applications; digital instrumentation and control systems; and technical issues related to standard plant designs.

The ACRS membership is drawn from a variety of engineering and scientific disciplines needed to conduct the broadly based review for these facilities, as well as proposed standards and criteria and related research activities. At this time, the Commission is specifically seeking to fill two vacancies with expertise in structural mechanics/materials engineering and metallurgy applicable to nuclear power systems, and the application of risk methods related to nuclear safety issues. Candidates are selected to provide a balanced technical base consistent with the requirements of the Federal Advisory Committee Act.

Because conflict-of-interest regulations restrict the participation of members actively involved in the regulated aspects of the nuclear industry, the degree and nature of any such involvement will be weighed. Each qualified candidate's financial interests must be reconciled with applicable Federal and NRC rules and regulations prior to final appointment. This might require divestiture of securities issued by nuclear industry entities, or discontinuance of industry-funded research contracts or grants.

A resumé describing the educational and professional background of the candidate, including any special accomplishments, professional references, current address and telephone number should be provided. Criteria used to evaluate candidates include education and experience, demonstrated skills in nuclear safety matters, and the ability to solve problems. Candidates must be citizens of the United States. All candidates will receive careful consideration. An indication of the candidate's ability and willingness to devote the time required (approximately 60-100 days per year) should also be provided. Applications will be accepted until September 29, 2000.

Copies of resumés of nominees should be sent to the Office of Human Resources, ATTN: Robin Avent, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555-0001.

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The Washington Post
OUTLOOK
Sunday, August 6, 2000

Let's Get Real About Risk

By David Ropeik

Hundreds of thousands of Americans will die this year, deaths that can be prevented. Millions will get sick with preventable illnesses. Billions of dollars and countless hours of human effort will be wasted unnecessarily—all because we are afraid of the wrong things.

In a frenzy of fear we are pouring millions this summer into protecting ourselves from the West Nile virus, and spending only a fraction of that sum on public education encouraging people to wash their hands, which would eliminate far more disease transmission than killing every mosquito in America.

Public and private spending on the cleanup of hazardous waste in America is estimated at $30 billion a year. Hazardous waste is a real problem, but the number of people whose health is at risk because of it is actually quite low. Compare that $30 billion with only $500 million a year on programs to reduce smoking, one of the leading preventable causes of death in America.

Or consider the Boston suburb where parents were so terrified that a chemical was in the air of just one room in their children's elementary school that they pressured the town school board to close the school at midyear. As a result, 6- to 11-year-old children who had nearly all been able to walk to school were put on buses and sent on snow-covered streets through rush-hour traffic to classrooms in the local high school.

Egad! What happened to common sense?

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When It Comes to Risk, Let Facts Rule—Not Fears

RISK, *From B1*

All this, even though the chemical (a solvent used in a nearby plant) was present at levels of under 10 parts per billion—well within safety limits—and the most cautious public health experts, hired by the parents, said that putting a fan in the window was all that was needed to make the air in the school safe.

We could make decisions that are more rational and informed. In many areas, science can identify the physical hazards, tell us how many people are likely to be affected by each one, what various mitigations will cost and how effective we can expect them to be. We can rank risks and remedies and put things in perspective. But we don't. Instead, we make policy based more on fear than fact.

Let's be blunt. This irrational response kills people. In a world of finite resources, we can only protect ourselves from so many things. If we overspend on risks such as pesticides or asbestos, which are real but of relatively low magnitude, we have less to spend on greater threats such as bacterial food poisoning or fossil fuel emissions. As a result, thousands of the people exposed to those higher risks will die.

The usual suspects blamed for bad policy are politics, greed, the media, even the open, manipulable nature of democracy itself. True, these are all factors in a process that often becomes a battle between competing private agendas rather than an informed search for policies that will serve the greatest common good. But the principal underlying cause of wasteful choices that seek protection from the wrong bogeymen is fear.

By definition, fear is more emotional than rational. We fear before we think. Cognitive scientist Joseph LeDoux of New York University identified neural pathways that send information about possible hazards to the amygdala, the fear response center in the ancient core of the brain, before the same information is sent to the cortex, the newer, thinking, rational part of the brain. A hiker who comes upon a shape on the ground that could be either a snake or a stick jumps out of the way immediately—even while another part of his or her brain is trying to think rationally about which one it is.

But society, with limited resources, must be more rational than that. When individual fears become group fears, and when those groups, organized or not, become big enough or visible enough to put pressure on the government to provide protection from less dangerous threats, we can end up with policies that leave a lot of people in the way of harm from higher risks that we're doing less about.

It turns out there are some universal perception factors, identified by social psychologist Paul Slovic and others, that make many of us afraid of the same things and thus tend to turn individual fears into group fears that then foster irrational government policy. Among them are:

CONTROL VS. NO CONTROL You normally feel in control when you drive. Not so when you are an airplane passenger bumping through turbulence at 30,000 feet. When you feel you have control, you are less afraid.

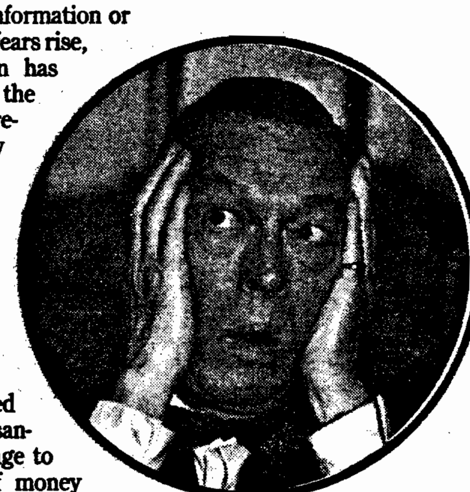
IMMEDIATE/CATASTROPHIC VS. CHRONIC We tend to be more afraid of what can kill a lot of us suddenly and violently, like a plane crash, than, say, lung cancer, which causes hundreds of thousands more deaths, but one at a time, over time.

NATURAL VS. HUMAN-MADE We're less afraid of radiation from the sun than of the radiation from power lines and cell phone towers. The risk from the sun is immensely greater, but no matter. Those power lines and cell phones are human-made. This one helps explain widespread fear of new technology and chemicals.

RISK VS. BENEFIT Medicines often have dangerous side effects, but the more we perceive a benefit from the drug, the less we fear its risks.

IMPOSED VS. VOLUNTARY Nonsmokers are often fearful of tobacco smoke. Smokers usually aren't.

TRUST VS. DISTRUST Experts in the field say this is often the most important risk perception factor, the fulcrum on which the whole seesaw of risk perception rests. If we trust the people informing us about a risk, and if we accept and trust that risk policies are determined in an open and reasonable process, our fears subside. If we don't trust the information or the process, our fears rise, as the Pentagon has discovered in the suspicious response of a few service members to its anthrax vaccination program.



So how do we make policymaking more rational? With a governmental process poisoned by selfish partisanship, often hostage to the influence of money and special interests, and spineless in the face of the latest media-fed fear frenzy, how can we get political leaders and government agencies to make wiser choices and protect us better? There is a model.

Some years ago, the Environmental Pro-

tection Agency and the automobile industry declared something of a truce in their war over the science of automobile emissions. Instead of each side spending millions on self-funded research the other side wouldn't accept, they each put in 50 percent of the money necessary—a total of $6 million—to create something called the Health Effects Institute. HEI was not created to make policy, but to give policymakers credible, trustworthy scientific information on which rational policy could be based. It was set up to be an impartial scientific review board—an agency of neutral arbiters, outside the government, beholden to nothing but the truth. To conduct its evaluations, it appoints panels of scientists, representing their various fields somewhat as a jury represents the community in a trial, so that no one with an ax to grind can control the process.

HEI's success and influence are growing. All the combatants in the air pollution fight, for example, have looked to HEI for "the" scientific opinion on the seriousness of particulate pollution.

Why not create such an independent, nongovernmental agency—let's call it the Risk Analysis Institute—to provide us with credible, trustworthy guidance on risks? The institute would rank the hazards we face, so we would know which ones are the most likely to occur; classify risks according to which ones have the most serious consequences; and conduct cost-benefit studies to help us rank mitigation choices by cost and effectiveness, so we would know which options will maximize resources to protect the most people.

In addition, it would identify the range of remaining uncertainties. The institute's analysts would also compare the policies of various agencies, to warn when a policy that reduces risk in one arena might increase it in another. For example, federal government standards to increase fuel efficiency reduce pollution, but encourage smaller, lighter vehicles, which are more fuel-efficient but more dangerous for passengers.

Supreme Court Justice Stephen Breyer suggested something like this in his 1993 book "Breaking the Vicious Circle" (written before his appointment to the high court). He proposed "a small centralized administrative group, charged with a rationalizing mission" within government. But bear in mind that trust is perhaps the most important of all the risk perception factors. An agency of government could not establish that trust.

An institute outside government might. To that end, it should have as much freedom as possible from the influence of politics, real or perceived. Its funding should come without strings, ideally from a mix of sources with competing agendas but willing to invest in credible, sound science. Funds should also be guaranteed, so no contributor can influence outcomes by threatening to cut off the cash. And its scientific work would have to be carried out by professionals who are chosen for their education and training, their expertise and reputations for integrity, neutrality and open-mindedness, not for who their political friends are.

Setting up the institute outside the government would serve another important goal: Final policymaking decisions would still be made by government agencies, preserving citizens' ability to voice their concerns and use the political process to help shape the outcome.

That means that lobbyists, politics, the media and money would also still have influence. The messy process of policymaking would not change dramatically. But a Risk Analysis Institute's credible analyses, supporting not a specific policy but rational policymaking in general, would incrementally move government decision making toward wiser, more informed choices.

Some conservatives have given "rational risk policy" and regulatory reform a bad name, often invoking a supposed "rational" response ostensibly in the public interest but actually on behalf of the special interests of corporate sponsors out to neuter the power of government oversight. Equally inflexible consumer groups and environmentalists resist rationality because the more fearful something sounds, the more it helps them advance their agenda.

But injecting rationality into the process is nothing more than good sense, in everyone's interest. It's time to create a vehicle to produce credible, reliable science to help develop policymaking that looks beyond our fears to what will do the most good.

The longer we wait, the more we risk. mi

②

Risk-Informed Part 50
Option 2

Overview of SECY-00-xxx "Risk-Informing Special Treatment Requirements"

475th ACRS Meeting
August 29, 2000

Timothy A Reed
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Objective of Option 2 SECY

- Provides preliminary views of ANPR comments
- Discusses conceptual approach to implementing Option 2 rulemaking plan
- Supports September 29, 2000 Commission brief

ANPR Comments

- Generally supportive of rulemaking
- Preliminary views on eight topics presented in SECY
- SECY attachment groups all ANPR comments into eight tables w/preliminary responses
- Final ANPR comment response -- proposed rule

ANPR Preliminary Views

Highlights of Significant Comments

- Selective Implementation -- identify all RISC-1 and 2 SSCs
- Impact on Other Regulations -- believe Part 54 should be risk-informed
- Need for Prior NRC Review -- objective continues to be little or no prior review
- PRA Quality -- Will consider other methods than consensus standards (NEI PRA certification)

ANPR Preliminary Views Cont'

Highlights of Significant Comments

- Approach -- believe can do all Option 2 rules in a single rulemaking (except §50.36)
- Part 21 -- may be necessary to modify Part 21 to remove RISC-3 SSCs from scope
- Part 21 should not apply to RISC-2 -- may be reporting but would be in §50.69

Option 2 Rulemaking Approach

- Consistent with SECY-99-256
- Robust categorization
- Licensees maintain functional capability of SSCs using existing or new programs
- RISC-2 SSCs -- control reliability, availability, capability per categorization process
- RISC-3 SSCs -- maintain design functions asdescribed in UFSAR
- Describe in UFSAR how meet requirements

Ongoing Tasks

- Review of NEI implementing guidance
  - Treatment and categorization guideline
  - PRA peer certification guideline
  - Industry pilot effort
- STP exemption review
- Contractor work -- commercial processes
- Continued interactions with stakeholders
  - Meeting with NEI in mid-September
  - Commission brief --September 29

Summary Of SECY

- ANPR comments generally supportive of effort to risk-inform special treatment requirements
- Rulemaking approach is consistent with SECY-99-256
- Review of STP exemption request continues
- Will continue interaction with stakeholders during development of new rule

(3)

Risk-Informed 10 CFR 50.44 "Standard for Combustible Gas Control System in Light-Water-Cooled Power Reactors"

Presented to
Advisory Committee on Reactor Safeguards

Presented by
Tom King, Mark Cunningham, Mary Drouin
Office of Nuclear Regulatory Research

August 29, 2000

U.S. Nuclear Regulatory Commission

OBJECTIVE

Risk-Informed Revisions to 10 CFR Part 50

- Enhance safety by focusing NRC and licensee resources in areas commensurate with their importance to health and safety
- Provide NRC with a framework to use risk information to take action in reactor regulatory matters
- Allow use of risk information to provide flexibility in plant operation and design, which can result in burden reduction without compromising safety

RISK-INFORMED 10 CFR 50.44

"Standard for Combustible Gas Control System in Light-Water-Cooled Power Reactors"

- ***Objective:*** control combustible gases (as a result of the design basis accident) that could challenge containment integrity, thereby, potential radionuclide release
- Rule specifies analytical requirements (e.g., accidents of concern, sources and amounts of combustible gases) and physical requirements to demonstrate analytical requirements are no challenge
- Work performed indicate no safety benefit or risk significance associated with parts of the regulation and some risk issues not addressed by regulation

50.44 TECHNICAL REQUIREMENTS

Analytical Requirements Imposed by the Rule

- The type of accident to be considered
  - Loss of coolant accident
  - Degraded core
- Type of combustible gas
  - Hydrogen
- Source of hydrogen
  - Fuel-cladding oxidation
  - Radiolytic decomposition of coolant
  - Corrosion of metal
- Hydrogen source term
  - 5% oxidation reaction over 2 minute period
  - 75% metal-water oxidation reaction for Mark III and ice condenser containments

50.44 TECHNICAL REQUIREMENTS

Physical Requirements Imposed by the Rule

- Measure the hydrogen concentration in containment
- Insure a mixed containment atmosphere
- Control combustible gas concentration in containment following a LOCA (recombiners)
- Install high point vents on all reactors
- Inert atmosphere in Mark I and II containments
- Provide hydrogen control system (igniters) in Mark III and Ice Condenser containments

RISK SIGNIFICANCE OF COMBUSTIBLE GASES

- Core damage/melt accident can potentially produce combustible gases (both hydrogen and carbon monoxide) from both fuel cladding oxidation and core-concrete interaction
- Control of post-LOCA hydrogen via a vent-purge methods can unnecessarily lead to radionuclide release to the atmosphere
- Depending on containment type and accident type, conditional large early release probability range from 0.1 to 1.0
- Hydrogen combustion not a significant challenge to containment integrity in the short term (~24 hours)
  - Large dry and subatmosphere due to large volume
  - Mark I and II due to inert atmosphere
  - Mark III and Ice Condenser due to igniters (except for station blackout)
- Combustible gas concentration may be sufficient to challenge containment integrity in long term
  - From core-concrete interaction for large dry, subatmosphere, ice condenser and Mark III
  - Oxygen generation from radiolysis can lead to de-inerted atmosphere in Mark I and II

50.44 RISK-INFORMED ALTERNATIVE

1. Concern	Combustion of gases poses challenge to containment integrity
2. Strategy	Relates to mitigative strategy of limiting radionuclide releases
3. Importance	Risk studies indicate conditional large early release probability for certain containment and accidents >0.1

⇛ ***not a candidate rule for elimination***

50.44 RISK-INFORMED ALTERNATIVE (continued)

Analytical Requirements ⇛ ***Enhance***

- Specify hydrogen source term based on realistic calculations
- Source term based on more likely severe accidents including both in-vessel and ex-vessel combustible gas generation
- Combustible gases include hydrogen and carbon monoxide
- Combustible gas control after 24 hours after onset of core damage be covered by Severe Accident Management Guidelines
- Similar to Mr. Christie's petition except that he requests a source term based on realistic calculations for accident with a high probability of causing severe reactor core damage

50.44 RISK-INFORMED ALTERNATIVE (continued)

Measure hydrogen concentration ⇛ ***Eliminate requirement***

- Hydrogen monitoring not needed to initiate or activate the hydrogen control systems for each of the containment types
- Hydrogen monitors have limited signifiance in mitigating threat to containment in early stages of a core melt accident
- Mr. Christie's petition also request elimination of this requirement

50.44 RISK-INFORMED ALTERNATIVE (continued)

Insure mixed atmosphere ⇛ ***Retain requirement***

- Needed to maintain defense-in-depth
- Needed to meet intent of GDC 50
- GDC 50 -- the containment and its compartments shall accomodate, with sufficient margin, the effects of potential energy sources including those from metal-water and other chemical reactions
- Current features that promote atmospheric mixing will not be degraded by any future plant modifications
- Mr. Christie's petition did not address this requirement

50.44 RISK-INFORMED ALTERNATIVE (continued)

Control H2 for postulated LOCA ⇛ ***Eliminate requirement***

- Type of accident not risk significant
- Means to control concentration (e.g., recombiners) do not provide any benefit
- Vent-purge method can result in unnecessary radionuclide releases to atmosphere
- Mr. Christie's petition included eliminating this requirement

50.44 RISK-INFORMED ALTERNATIVE (continued)

Inert Mark I/II containments ⇛ ***Retain requirement***

- Removal would result in integrity of Mark I and II containment being highly vulnerable to hydrogen combustion
- Mr. Christie's petition included retaining this requirement

50.44 RISK-INFORMED ALTERNATIVE (continued)

Install high point vents ⇛ ***Retain requirement***

- Combustible gases in RCS can inhibit flow of coolant to the core
- Capability to vent the RCS provides a safety benefit
- Mr. Christie's petition included retaining this requirement

50.44 RISK-INFORMED ALTERNATIVE (continued)

H2 control system (igniters) for Mark III and Ice Condensers
⇛ ***Enhance requirement***

- Modify to control hydrogen during risk significant core melt accidents
- Control system uses igniters which are AC dependent
- Under SBO conditions, igniters not available and containment vulnerable to hydrogen combustion
- SBO shown to be large contributor for some plants
- Mr. Christie's petition only proposes that the hydrogen control system be capable of meeting a specified performance level. Vulnerability under SBO conditions would still exist.

50.44 RISK-INFORMED ALTERNATIVE (continued)

⇛ ***Alternative (performance-based) requirement***

- Alternative that would allow licensee to use risk information
- Demonstrate plant meets specified performance criteria
  - e.g., maintain containment intergrity for at least 24 hours for all risk-significant events
- Attractive for future plants
- Mr. Christie's petition included a requirement that for facilities with other types of containments *"must demonstrate that the reactor containment .... can withstand, without any hydrogen control system, a hydrogen burn for accidents with a high probability of causing severe core damage."* Believe staff recommendation is equivalent.

50.44 RISK-INFORMED ALTERNATIVE (continued)

Alternative ⇛ ***"Long-term" recommendation***

- Long term control (greater than 24 hours after onset of core damage) be included as part of licensee's Severe Accident Managment Guidelines
- Combustible gases still pose challenge to containment integrity in the long term with the possibility of a large late radionuclide release
- Mr.Christie's petition did not address the concern of long-term combustible gas control

PHASE II

Upon Commission Approval

- Proceed with rulemaking

Risk-Informing NRC Regulations

August 29, 2000 ACRS Meeting

Adrian Heymer, NEI

NEI

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Option 2

- **Risk-informed regulatory regime**
  - Focus on SSCs and activities that are safety-significant
  - Significant interaction and requirements being imposed on RISC-3 SSCs
- **ASME Standard & PRA certification**
  - Peer review -- an acceptable methodology to assess PRA suitability for Option 2
  - Further interactions to resolve NRC comments

Option 3 -- Implementation

- **Regulations (mandatory or optional) should not place unnecessary resource burden on licensees or NRC staff**
- **NRC decision on including new regulatory elements should be based on:**
  - Up-to-date technical analyses and information
  - Estimates of licensee/NRC benefits & burden

NEI

Option 3 -- Implementation

- **NRC Framework -- document being revised**
- **§50.44 -- Must be sound technical basis for including or excluding optional requirements**
- **Estimate of additional burden?**
- **§50.46 -- Redefinition of Large-Break LOCA**
    - NEI interacting with NSSS Owners' Groups to develop a common approach
    - Follow-on activities

(5)

Advisory Committee on Reactor Safeguards
Full Committee

Combustible Gas Control

August 29, 2000
Two White Flint, Rockville, MD

Bob Christie

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SUMMARY
Change Hydrogen Control Regulations as of 8/28/00

10CFR50.12 Exemption Requests	Petition for Rulemaking	SECY-98-300 Option 3 Framework	SECY-98-300 Option 3 10CFR50.44
Submitted:	**Recent action:**	**Recent action:**	**Agreement:**
San Onofre 2&3 - 9/10/98 Approved 9/3/99	ACRS 6/29/00	ACRS 6/29/00	Delete post LOCA hydrogen requirements
Oconee - 7/26/00	Letter - Christie to Mike Snodderly (NRC), 7/3/00	ACRS 7/11/00	Containment air mixing unchanged
	ACRS 7/12/00	Letter - Christie to Ashok Thadani (NRC), 7/19/00	Reactor Coolant System high point vents unchanged
	Letter - Christie to Sam Collins (NRC), 7/14/00	Letter - Christie to Ashok Thadani (NRC), 8/24/00	Mark I's and Mark II's inerted unchanged
	Letter - Christie to Cynthia Carpenter (NRC), 7/20/00		**Disagreement:**
			NRC staff wants to add long term requirements for hydrogen monitors
			NRC wants igniters operable during Station Blackout for Mark IIIs and ice condensers
Future action:	**Future action:**	**Future action:**	**Future action:**
Other submittals in preparation	ACRS 8/29/00	ACRS 8/29/00	ACRS 8/29/00
	Recommendation by NRC staff to NRC Commissioners at end of August 2000		Recommendation by NRC staff to NRC Commissioners at end of August 2000
	Mike Snodderly (NRC) working on open purge valve - severe accident		

Duke Power.
A Duke Energy Company

W. R. McCollum, Jr.
Vice President

Duke Power
Oconee Nuclear Site
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July 26, 2000

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: Duke Energy Corporation
Oconee Nuclear Station, Units 1, 2 and 3
Docket Numbers 50-269, 50-270 and 50-287
Request for Exemption to 10CFR50.44, 10CFR50, Appendix A, General Design Criterion 41, and 10CFR50, Appendix E, Section VI.
Proposed Technical Specification Change Concerning Hydrogen Control System (TSCR 2000-05)

Pursuant to the provisions of 10 CFR 50.12, " Specific exemptions," Duke Energy Corporation (Duke) is requesting an exemption to the requirements of 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors," 10 CFR 50, Appendix A, General Design Criterion 41, "Containment atmosphere cleanup," and 10 CFR 50, Appendix E, Section VI, "Emergency Response Data System." The purpose of this exemption request is to remove requirements for hydrogen control systems (i.e., containment post-accident hydrogen monitors and recombiners) from the Oconee, Units 1, 2, and 3 (ONS) design basis. With this change, the consideration of hydrogen generation would no longer be included in the design basis of ONS. Accordingly, the enclosed Technical Specification (TS) Change Request 2000-05 would remove the post-accident hydrogen control systems from the ONS TS and provide the basis for deletion of a Selected Licensee Commitment concerning hydrogen recombiners.

Enclosure 1 provides the documentation supporting the exemption request. Enclosure 2 is a license amendment request, which consists of five attachments. Attachments A and B provide mark-up and new pages of the Oconee TS, respectively. The Description of Proposed Changes and Technical Justification is provided in Attachment C. Attachments D and E provide the No Significant Hazards Consideration Evaluation and Environmental Impact Analysis, respectively.

As described in the enclosures, approval of the requested exemption would improve the safety focus at Oconee and represent a more effective and efficient method for maintaining adequate protection of public health and safety. The requested changes would permit simplification of Emergency and Emergency Response Plan Procedures thereby reducing operators' post-accident burden. Such simplification would enable operators to give priority to more important safety functions following postulated plant accidents.

1/2

It is Duke's intention that, upon NRC approval of this request, the description of the hydrogen control systems, its bases and other associated discussions would be removed from the UFSAR and from the Emergency and Emergency Response Plan Procedures.

A similar request for an exemption to the requirements of 10 CFR 50.44, and 10 CFR 50, Appendix A, General Design Criterion 41, 42 and 43 was approved by the NRC for San Onofre Nuclear Generation Station, Units 2 and 3, by letter dated September 3, 1999.

Implementation of this amendment to the Oconee Technical Specifications will impact the Oconee UFSAR. Necessary changes will be made in accordance with 10 CFR 50.71(e). Duke requests a 90-day grace period for implementation of this exemption request and the associated changes.

The Duke Nuclear Safety Review Board and the Oconee Plant Operations Review Committee have reviewed and approved this proposed Technical Specification amendment.

A copy of this application is being forwarded to the South Carolina Department of Health and Environmental Control for their review and, as appropriate, subsequent consultation with the staff.

Please contact Robert C. Douglas at 864-885-3073 with any questions regarding this submittal.

Very truly yours,

W. R. McCollum, Jr.
Site Vice President
Oconee Nuclear Station

Enclosures

2/2

Performance Technology

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July 3, 2000

Mr. Mike Snodderly
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Dear Mr. Snodderly:

Following our conversations last week, I spent some additional time over the weekend considering the approach you might want to use for the evaluation of the Emergency Operating Procedures at the nuclear units for hydrogen control during severe accidents. It is my belief that your time could be best spent in the following manner.

I recommend that if you want to perform any evaluations, you first evaluate those plants that have movable hydrogen thermal recombiners that must be physically installed after accidents to control hydrogen from design basis accidents. If the operators at these nuclear units contemplate the use of any system for hydrogen control during severe accidents, it will be the hydrogen purge system. Due to the large amounts of hydrogen which would be produced in a short time frame in severe accidents, the operators will recognize that the only hydrogen control system they have is the hydrogen purge system. Whether they would activate the hydrogen purge system in severe accidents is the question. I believe that the operators would not activate the hydrogen purge system in severe accidents but I have not evaluated the situation in detail. As you know, I have a concern about this situation because the activation of the hydrogen purge system during severe accidents would be very detrimental to public health risk.

After the evaluation of those nuclear units with movable hydrogen thermal recombiners, if you still believe you have to continue, I would continue with an evaluation of the nuclear units with permanent hydrogen thermal recombiners but that have a hydrogen purge system as backup. At these nuclear units, the hydrogen thermal recombiners will be the first system called upon for hydrogen control and the hydrogen purge system will be the backup. Neither system will be effective in severe accidents for controlling hydrogen, but I believe there is less likelihood of using the purge system in these nuclear units than in nuclear units with movable hydrogen thermal recombiners but this is only my opinion. The operators will still have to evaluate the use of the hydrogen purge system during severe accidents at these units.

RJO
7/3/00

During these evaluations, I am not sure that one could put much weight on 10CFR100 radiation dose accident calculations to determine whether an operator would or would not activate the hydrogen purge systems during severe accidents as you have suggested. As I stated in the ACRS Subcommittee on Probabilistic Risk Assessment meeting on June 29, 2000, I believe that 10CFR100 radiation dose accident calculations are not appropriate for severe accidents. There is also the matter of timing for 10CFR100 calculations It is generally assumed that the 10CFR100 calculations for the activation of the hydrogen thermal recombiners for design basis events would take place days after the design basis accident. In severe accidents, large amounts of hydrogen can be produced in hours, not days and I doubt that anyone will have the time to perform 10CFR100 calculations. We should not be performing 10CFR100 dose calculations after severe accidents. RJC

As I have indicated to you in our previous conversations, your evaluation of the Emergency Operating Procedures is a matter of concern for the NRC in the immediate future. In my opinion, the best that we could hope for from your effort would be some "band aid" solutions to possible problems with the Emergency Operating Procedures. The permanent solution to the problem is to eliminate the requirements for the hydrogen thermal recombiners and the hydrogen purge systems following design basis events from the nuclear units. This permanent solution can be quickly achieved either by the approval of my petition for rulemaking or by the approval of 10CFR50.12 exemption requests. Personnel at the nuclear plants would like to solve the problem in a permanent fashion and I agree with them completely. In my opinion, the optimum solution would be to approve the petition for rulemaking in an expedited manner and allow the nuclear units to quickly eliminate the requirements for the hydrogen thermal recombiners and the hydrogen purge system from the Technical Specifications, Emergency Operating Procedures, Final Safety Analysis Reports, and any other place such requirements exist.

Please let me know of your progress on the evaluation of the Emergency Operating Procedures. Please contact me if you have any questions or desire further assistance.

Sincerely,

Bob Christie

Bob Christie

cc: Cynthia A. Carpenter
Anthony W. Markley

Performance Technology

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July 14, 2000

Mr. Sam Collins
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20872-2738

Dear Mr. Collins:

By my letter dated 10/7/99 to the NRC Commissioners, I raised safety issues regarding existing regulations for hydrogen control following postulated accidents. My letter to the Commissioners indicated that, based on the San Onofre Task Zero Safety Evaluation Report, strict compliance with existing regulations was detrimental to public health and safety. My letter was sent to you for action. Following discussions with your staff, I sent a letter to Mr. Frank Akstulewicz of your staff, dated 11/9/99, and agreed to treat part of my letter as a petition for rulemaking concerning 10CFR50.44 and 10CFR50, Appendix A, Criterion 41. Your letter to me dated 1/4/00, confirmed your staff was going to process my petition for rulemaking using the usual NRC practices.

As explained to me last year by your staff, I understand the usual NRC practices for rulemaking include consideration of "adequate protection" and consideration of 10CFR50.109, Backfitting. The usual practices also require that the petition for rulemaking be noticed for public comment, which occurred January 12, 2000. It was my understanding that my petition for rulemaking was to be considered on its own merits per these usual procedures. On June 29, 2000, your staff stated in an ACRS meeting that my petition was not being considered on its own merits but was being incorporated into "Option 3" of SECY-98-300. Later, your staff told me that my petition was not likely to be the recommended rulemaking from Option 3. This was the first time I heard of this decision by your staff.

In my 11/9/99 letter, I stated that it would be advantageous to make sure that Dr. Tom King and the people responsible for SECY-98-300, "Option 3" were aware of the actions of <u>your staff</u> in this matter. In your letter to me dated 1/4/99, you indicated that in addition to my petition for rulemaking being evaluated on its own merits, my letter had been sent to NRC Office of Research for consideration as part of NRC Research activities concerning "Option 3." I did not take this "addition" as meaning my petition would be evaluated by Option 3 standards and I do not believe in your letter that you meant that my petition be incorporated into the Option 3 evaluation.

My recommendation for changes to the regulations applies to all nuclear electric power units in the United States. I believe all the nuclear units are subject to the same

RJe
7/14/00

detrimental impact from the existing regulations. My petition for rulemaking is premised on the fact that existing hydrogen control regulations make all the nuclear units less safe than the units would be if the regulations were changed as I proposed. I believe the NRC staff Safety Evaluation Report for San Onofre is applicable to all the nuclear units. I believe that implementation of the proposed changes at all nuclear electric power units is necessary to improve safety. My petition for rulemaking should not be evaluated in Option 3 because my petition is not a "voluntary" effort applicable only to those nuclear units which "volunteer" for Option 3. The criteria used for evaluation in Option 3 go far beyond "adequate protection" and the backfit rule.

I have informed your staff that I do not believe my petition should be incorporated into Option 3 for evaluation and I have also informed the ACRS about this position in their meeting on July 12, 2000. There is no basis for treating my petition in a manner other than "standard practice." Approval of my petition for rulemaking will make the nuclear units "safer," therefore meeting the adequate protection criteria. My petition meets the requirements of 10CFR50.109, Backfitting. My petition has undergone the required period of public comment. My petition "risk-informs" the regulations and makes the regulations more effective and efficient. I believe it should be possible to make a decision on my petition on its own merits in short order. It has already been nine months since I brought this matter to the attention of the NRC Commissioners and they referred my 10/7/99 letter to you for action. This is nine months in which I believe the plants have been less safe.

To summarize, it is my understanding that your staff is not presently processing any approval or disapproval of my petition for rulemaking. Your staff is waiting for something to come out of Option 3. Without approval of my petition, the utilities cannot implement changes to make the nuclear electric power units "safer" and more economic with respect to hydrogen control except by the 10CFR50.12 exemption request process. As your staff is aware, some utilities are pursuing the 10CFR50.12 process following the pattern approved in the San Onofre Task Zero. These actions by other utilities are believed necessary because there is no visible action on my petition for rulemaking in spite of your staff's granting the hydrogen control exemptions to San Onofre.

I would like to meet with you to discuss these issues further. I will contact your office to arrange an appointment for such a meeting. In the meantime, please contact me if you have any questions.

Sincerely,

Bob Christie

Bob Christie

Cc: Ashok Thadani

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July 20, 2000

Ms. Cynthia Carpenter
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Dear Ms. Carpenter:

I appreciate Tony Markley and you taking the time to talk to me yesterday about my letter to Sam Collins, dated 7/14/00.

My summary of our conversation yesterday is as follows. Sam Collins has now designated you as the individual in NRR that I am to talk to in all matters concerning my petition for rulemaking noticed in the Federal Register on 1/12/00. You indicated Sam Collins does not want to meet with me to discuss my letter of July 14, 2000.

In the telephone conversation, you stated that my petition for rulemaking is a "risk-informed" matter. You indicated that, as stated in the ACRS meeting on June 29, 2000, the evaluation of my petition has been incorporated into Option 3 of SECY-98-300. You believe that Option 3 people are the appropriate people to judge the technical basis of my petition for rulemaking and the Option 3 criteria are the appropriate criteria for evaluation. You do not believe that my petition for rulemaking will be the recommended approach coming out of Option 3 for hydrogen control and therefore personnel from NRR are not evaluating my petition separate from Option 3. When asked what the process would be if Option 3 did not exist, you indicated my petition would have been sent to Research for evaluation.

You indicated that you are constrained by the rules of the NRC with respect to rulemaking and have no other option to follow except the path chosen. When asked, you indicated that there is no benefit for a public meeting for me to discuss this issue with you since you have my letter to Sam Collins. You indicated that Tony Markley is drafting a reply to my letter to Sam Collins and that I will receive this letter after it goes through the concurrence process in NRC. You would give me no schedule for when such a letter would be issued.

In the telephone conversation, I explained that I did not believe the NRR position was the appropriate position to be taken and reiterated my concerns expressed in my letter to Sam

RJC
7/20/00

Collins. My petition for rulemaking is a safety concern first expressed in my letter to the NRC Commissioners, dated 10/7/99. I asked if you had had any discussions recently with Mike Snodderly concerning my safety concerns and you indicated that you had not. I reiterated that my petition was sent to NRR for resolution by the Commissioners and that I agreed to make my letter a petition for rulemaking on the basis of the existing procedures for rulemaking. Again, my petition is not a "voluntary" initiative to be considered in Option 3.

I pointed out the petition for rulemaking was a follow-up to Task Zero at Arkansas Nuclear One and Task Zero at San Onofre and the rulemaking was a better alternative than the exemption request process. You indicated no concern about the licensees having to submit exemption requests under 10CFR50.12, similar to the San Onofre submittal, to make the plants safer and obtain the same decision that would be gained by the approval of the petition for rulemaking.

All in all, it is clear that months ago NRR personnel determined a course of action for evaluation of my petition for rulemaking and that this course of action involved Option 3 rather than usual practices. It does not appear that there is anything that I can say or do for you to change this position. I assume that you are taking this course of action with the complete approval of you supervisors.

As I indicated in the telephone conversation, I am very dissatisfied with the course of action taken by NRR. My petition addresses a safety concern that is documented in the Task Zero at Arkansas Nuclear One and the Task Zero at San Onofre and in public meetings and letters to the NRC. Every day that the NRC delays the approval of my petition is another day in which I believe the nuclear electric power units are less safe. My petition for rulemaking should be evaluated by the usual practices of the NRC for rulemaking which is what NRR staff and I agreed to last year. There are much better ways to "risk-inform" the regulations than Option 3. One of these better ways is to use the usual practices.

I am now waiting for a reply for my letter to Mike Snodderly, dated 7/3/00, and a reply to my letter to Sam Collins, dated 7/14/00..

Sincerely,

Bob Christie

Bob Christie

cc: William D. Travers
Samuel J. Collins
Ashok Thadani

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July 19, 2000

Dr. Ashok Thadani
Office of Research
U. S. Nuclear Regulatory Commission
11545 Rockville Pike
Rockville, MD 20852-2738

Dear Dr. Thadani:

During your staff's presentation on the "Risk-Informed Part 50 Framework" to the Advisory Committee on Reactor Safeguards on July 12, 2000, your staff identified four "issues" that are to be sent to the NRC Commissioners for guidance as part of your report to the Commissioners due in August, 2000. These are.

1 Should selective implementation within a regulation of the technical requirements be allowed?
2. Should safety enhancements be required to pass backfit rule?
3. Should there be a reverse backfit test for burden reduction?
4. Role of Safety Goals? (not on slides used but added by Dr. King in presentation).

I wish to comment on issues #2, #3, and #4 because I believe you should accurately describe these issues to the Commissioners. In this vein, I recommend that you read the transcript of the discussion I had with the ACRS Subcommittee on Probabilistic Risk Assessment during their meeting on July 11, 2000.

Issue #2 (backfit) and issue #4 (Safety Goals) are to me the same issue and my comments on these two issues are contained in Attachment 1. My comments on issue #3 (reverse backfit) are contained in Attachment 2.

The following is a summary of my comments.

With respect to issues #2 (backfit) and #4 (Safety Goals).

The Part 50 Framework document called Draft, Revision 0, dated April 2000:

a. Ignores the standard of "adequate protection" which is the legal basis for the licensing of existing plants.

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b. Ignores the direct instruction from the NRC Commissioners to consider the backfit rule in any attempt to change the regulations to implement the Safety Goals at existing nuclear units. See Attachment 1, Section A.

c. Uses a partitioned objective to risk-inform the regulations for the existing nuclear electric power units that the NRC Commissioners stated was to be used for evolutionary design. See Attachment 1, Section B.

d. Uses partitioned objectives to change the regulations that in effect require risk-informed regulations to be written to a level below "how safe is safe enough." See Attachment 1, Section C.

With respect to issue #3, the NRC staff is asking the Commissioners for direction on the "issue of reverse backfit" when there is no issue of reverse backfit. See Attachment 2.

I believe that it would be worth while for me to discuss my comments on this subject with you in person before you send your report to the NRC Commissioners in August, 2000. I will be contacting your staff in the near future to arrange such a meeting.

Sincerely,

Bob Christie

Bob Christie

cc: Sam Collins (Office of Nuclear Reactor Regulation)
Dana Powers (Advisory Committee on Reactor Safeguards)

Attachment 1
Letter from Bob Christie to Dr. Ashok Thadani dated 7/19/00

The use of Safety Goals and Backfitting in enhancing existing regulations

All of my following comments are based on the Framework Document that is designated Draft, Revision 0, April 2000, with the NRC authors listed as Mary Drouin and Alan Kuritzky and a host of people from Sandia National Laboratories and Brookhaven National Laboratory.

The Framework for Risk-Informing the Technical Requirements of 10CFR presently being used by the NRC staff indicates that "Established quantitative health objectives (QHOs) and related subsidiary quantitative objectives will be used to guide the development of risk-informed regulatory requirements." While the NRC staff state that these quantitative objectives will not appear in the regulations, the NRC staff states that these quantitative objectives will be used to write deterministic regulations that will achieve the levels defined by these quantitative objectives.

As you well know, I have been advocating for a number of years that we make use of the Quantitative Health Effects Objectives (QHOs) from the 1986 NRC Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants" to make the regulations more effective and efficient in providing "reasonable assurance of adequate protection of public health and safety." My effort has become known as the "Whole Plant Study." On the surface, the NRC staff Framework Document appears to have the same objectives that I have been advocating. However, as always, "the devil is in the details." The details of the Framework Document are incompatible with my program and also incompatible with the direction specified by the NRC Commissioners in the use of the Safety Goals to enhance regulations.

The Framework for Risk-Informing the Technical Requirements of 10CFR50 claims to be following a "top down" approach to enhance the regulations but in reality the Framework Documents is a "bottom up" approach based on partitioned objectives that are not related to either "adequate protection" or "how safe is safe enough." The NRC staff claims that the partitioned objectives they want to use are based on the Quantitative Health Effects Objectives and the directions the staff received from the NRC Commissioners in the Staff Requirements Memorandum dated June 15, 1990. As I pointed out to the ACRS Subcommittee on Probabilistic Risk Assessment on July 11, 2000, these claims of the NRC staff are not accurate.

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A. The NRC staff ignores the issue of "adequate protection" and the backfit rule.

In the Framework Document, in Section 2.1, the NRC staff states that the Atomic Energy Act requires the NRC to ensure that nuclear power plant operation provides adequate protection to the health and safety of the public. The staff notes that this requirement is called the "adequate protection" standard or the " no undue risk" standard. What Section 2.1 fails to note is that the NRC can enhance the standard of "adequate protection" by the use of 10CFR50.109, Backfitting. After this description in Section 2.1, defining what is required by the Atomic Energy Act, you find that the NRC staff no longer use the term "adequate protection of public health and safety" but rather the term "protecting public health and safety." The NRC staff in the rest of the Framework Document effectively ignores the concept of "adequate protection" and the backfit rule.

This deliberate action is taken by the NRC staff involved in the writing of the Framework Document in spite of the direct instructions by the NRC Commissioners in the Staff Requirements Memorandum of June 15, 1990, covering implementation of the Safety Goals.

> "...6) In order to enhance our regulatory process for the current generation of plants, the Commission believes the staff should strive for a risk level consistent with the safety goals in developing or revising regulations. In developing and applying such new requirements to existing plants, the Backfit Rule should apply."
>
> "...11) The Commission agrees that it must not depart from or be seen as obscuring the arguments made in court defending the Backfit Rule.
>
> These arguments clearly established that there is a level of safety that is referred to as "adequate protection." This is the level that must be assured without regard to cost and, thus, without invoking the procedures required by the Backfit Rule. 1/ Beyond adequate protection, if the NRC decides to consider enhancements to safety, costs must be considered, and the cost-benefit analysis required by the Backfit Rule must be performed. The Safety Goals, on the other hand, are silent on the issue of cost but do provide a definition of "how safe is safe enough" that should be seen as guidance on how far to go when proposing safety enhancements, including those to be considered under the Backfit Rule.

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B. The application of quantitative objectives for evolutionary design to existing nuclear units.

The Framework Document uses a value of less than or equal 0.1 for the Conditional Probability of Early Containment Failure. The NRC Commissioners directed in the June 15, 1990, Staff Requirements Memorandum that this value apply to evolutionary designs, not existing designs.

> 4) ..."The Commission has no objection to the use of a 0.1 Conditional Containment Failure Probability objective for the evolutionary design, as applied in the manner described above.

C. Partitioned Objectives.

It has been demonstrated through analysis (Probabilistic Risk Assessment) of each nuclear unit in the United States that the public health risk of each nuclear unit is unique to each unit. Each nuclear unit has a unique public health risk profile that is impacted by each unit's personnel, equipment, procedures, maintenance, operation, site location, meteorology, population density, etc. Each unit, through its Probabilistic Risk Assessment, knows a lot about its risk profile but it is very difficult to generalize such knowledge to all the nuclear units. Because of this unique profile of each nuclear unit, it is very difficult to partition any overall standards. Each nuclear unit has a unique way of meeting the standard of adequate protection or meeting the standard of how safe is safe enough.

The NRC Commissioners were very aware of the unique characteristic of public health risk from nuclear power plants both when they published the 1986 Policy Statement on Safety Goals and when they issued the June 15, 1990 Staff Requirements Memorandum.

In the 1986 Policy Statement, the NRC Commissioners deliberately defined only two Quantitative Health Effects Objectives. The Commissioners deliberately did not set any performance guideline for core damage frequency or containment conditional failure probability. In the 1986 Policy Statement the Commissioners directed the Staff to investigate the possibility of setting a performance guideline such that the overall mean frequency of a large release of radioactive material to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation. The NRC staff later determined that this performance guideline was not compatible with the Quantitative Health Effects Objectives.

Goal allocation of higher tier objectives to lower tier objectives is very difficult. Goal allocation can be successful if the lower level objectives are derived directly from the higher tier objectives and do not create a new higher tier level objective. If done

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correctly, these lower tier objectives can simplify the process and still lead to the correct decisions. If done incorrectly, these lower tier objectives lead to logical inconsistencies which complicate the decision process and lead to incorrect decisions.

In the June 15, 1990, Staff Requirements memorandum, the NRC Commissioners gave some direction for the use of "partitioned" objectives.

> "Implementation of the safety goal may require development and use of 'partitioned' objectives. In general, the additional objectives should not introduce additional conservatisms. The staff should bring its recommendations on the use of each such subsidiary objective to the Commission in the context of the specific issue for which it would be useful and appropriate, and explain its compatibility with the safety goals. Based upon the NRC's review of a sample of plant PRAs, it appears that these plants not only meet the quantitative health effects objectives but exceed them. This may or may not reflect excessive conservatism in regulations. While there have been improvements in PRA techniques, uncertainties in the summary results are still such that quantitative PRA objectives should not be used as licensing standards or requirements.
>
> The Commission believes that the safety goal objectives should be applied to all designs, independent of the size of containment or character of a particular design approach to the release mitigation function. Accordingly, for the purpose of implementation, the staff may establish subsidiary quantitative core damage frequency and containment performance objectives through partitioning of the Large Release Guideline. These subsidiary objectives should anchor, or provide guidance on 'minimum' acceptance criteria for prevention (e. g. core damage frequency) and mitigation (e.g. containment or confinement performance) and thus assure an appropriate multi-barrier defense-in-depth balance in design. Such subsidiary objectives should be consistent with the large release guideline, and not introduce additional conservatism so as to create a *de facto* new Large Release Guideline.
>
> A core damage probability of less than 1 in 10,000 per year of reactor operation appears to be a very useful subsidiary benchmark in making judgments about that portion of our regulations which are directed toward accident prevention.
>
> ...The Commission has no objection to the use of a 0.1 Containment Conditional Failure Probability for the evolutionary design, as applied in the manner described above.
>
> ...These partitioned objectives are not to be imposed as requirements themselves but may be useful as a basis for regulatory guidance."

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In Section 3.3 of the Framework Document, it is stated. "...The quantitative health objectives are the highest-level quantitative goals. The QHOs were originally set as a measure of 'safe enough.' Given this position of the Commission, there are no risk arguments for setting subsidiary quantitative objectives more stringent than the QHOs."

However, the Framework Document uses three partitioned quantitative objectives for guidelines for writing new regulations for the existing nuclear units.

1. Core damage frequency less than or equal 1.0 E-4.
2. Conditional Probability of Early Containment Failure of less than or equal 0.1.
3. Large Early Release Frequency of less than or equal 1.0 E-5.

All of the existing nuclear electric power units in the United States have been licensed to the standard of "reasonable assurance of adequate protection of public health and safety." All existing nuclear electric power units meet this standard. As stated in the June 15, 1990 Staff Requirements Memorandum issued by the NRC Commissioners, it is believed that all of the existing nuclear electric power units in the United States are lower on a risk scale than the Quantitative Health Effects Objectives though no one knows this for sure. What is known, is that many of the existing nuclear electric power units in the United States do not meet one or more of the three quantitative objectives being used in the Framework Document. It is clear that using these partitioned quantitative objectives for guidelines for writing new regulations would be requiring nuclear units to go below "how safe is safe enough"

For example: as stated in the ACRS letter from R. L. Seale to Shirley Ann Jackson, May 11, 1998, "Elevation of CDF to a Fundamental Safety Goal, and Possible Revision of the Commission's Safety Goal Policy Statement."

> "...Observation 2. Results of analyses indicate that a CDF of 1.0E-4 per reactor year, if applied to all plants with their current level of containment performance, in many cases would be more conservative than the QHOs. This would, therefore, be a new *de facto* fundamental safety goal."

I believe that the same statement could be made of the other two partitioned quantitative objectives. The Framework Document states the proper use of the QHOs in Section 3.3 and then violates the statement with the choices for the partitioned objectives.

-1- (Attachment 1)

Attachment 2
Letter from Bob Christie to Dr. Ashok Thadani dated 7/19/00

"Reverse Backfitting"

In the discussion with the NRC Commissioners on June 20, 2000, Mr. James P. Riccio stated that "if the staff of the NRC tries to impose new requirements, they have to go through a cost/benefit analysis commensurate with the backfit rule. To deregulate, you don't have to do that." Mr. Riccio indicates that this is a disparity. Mr. Riccio states "...when the regulator sees something that is important to safety, that they should be able to act upon it without having to go through the machinations (I assume he means the backfit rule), especially if you're going to allow them (I assume he means the licensees) to deregulate. I (Mr. Riccio) believe in equal treatment. If you are going to allow the deregulation to occur without any safety analysis - sorry, cost/benefit analysis, then the same should be said for imposing new regulations under this rubric."

Some of the staff of the Nuclear Regulatory Commission have started to call this position "Reverse Backfitting." I call it "Avoiding Backfit Analysis."

I also believe in equal treatment. Any change to the NRC regulations that is imposed on licensees, started by anyone, should go through a detailed safety evaluation. Any change to the NRC regulations that is imposed on licensee, started by anyone, should go through a detailed cost/benefit evaluation.

What Mr. Riccio sometimes sees and complains about is a 10CFR50.109, Backfitting analysis for changes to the regulations that the NRC staff initiates. This backfit regulation exists because all the NRC regulations are predicated on "reasonable assurance of adequate protection of public health and safety" and the use of 10CFR50.109 if the NRC staff wishes to go beyond adequate protection. The NRC regulations are not predicated on zero risk. The courts in the United States have made this clear.

The NRC staff has never liked the backfit rule and has always tried to avoid backfit analysis. See the "Report on Backfitting and Licensing Practices at the U. S. Nuclear Regulatory Commission," by James R. Tourtelotte, Chairman, Regulatory Reform Task Force, U. S. NRC, March 11, 1985. A sample from the Report on Backfitting: "The primary purpose of the backfit rule when it was passed on March 31, 1970, was to improve the stability of the licensing process by minimizing the alterations of structures, systems or components of a nuclear power plant after the construction permit has been issued. The rule has been selectively ignored by the staff for nearly 15 years. There is a substantial amount of evidence suggesting that the staff's backfitting practices which have cost consumers billions of dollars have made nuclear plants more difficult to operate and

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maintain, have injected uncertainty and paralyzing delay into the administrative process, and in some instances may have reduced rather than enhanced public health and safety."

After the backfit rule was strengthened in 1988, the impact was not that the staff performed backfit evaluations for changes that the NRC staff initiated. Rather, the impact was that the staff proposed fewer direct changes to the regulations. Over the years since 1988, the NRC learned to avoid the strengthened backfit rule by claiming the change initiated by the NRC staff was necessary to meet "adequate protection," or by getting the licensee to "voluntarily" commit to the change, or by issuing a Regulatory Guide. A regulatory guide is a NRC staff document that is "voluntary." Of course, the fine print in the "voluntary" regulatory guide says that this guide is an acceptable method and other methods are also acceptable to meet NRC staff requirements but any other method must meet at least the requirements of the NRC regulatory guide.

In my opinion, the attempt by the NRC staff to avoid having to perform a backfit analysis when the NRC staff proposes additional requirements to the regulations in Option 3 (voluntary), in spite of the direct NRC Commissioner direction to perform such an analysis (See Attachment 1), is a clear example of how the NRC tries to get around the backfit rule.

When a licensee suggests a change to the regulations, there is a detailed and protracted process of safety evaluations done under either rulemaking or 10CFR50.12, Specific exemptions. I wish it were as easy as Mr. Riccio appears to believe for a licensee to make a change to the regulations. I know about these processes because I have spent the last three years performing these safety evaluations. The process of safety evaluation is a very rigorous and expensive process.

What Mr. Riccio doesn't see is the cost/benefit analysis done by the utilities if a licensee suggests changes to the regulations. From my own experience over the last three years, I can tell you personally that the costs to the licensees of changing the regulations, even when everyone (licensee and NRC staff) agrees that the change will result in a safer nuclear unit, are substantial. Every step of the way through the "Whole Plant Study" has had some cost to the licensees. The meetings with the NRC staff, the analyses performed, the reviews performed, the paperwork submitted, the responses to NRC questions, and all the other actions required to satisfy the staff of the NRC are real costs which are borne by the licensees. If not directly, then indirectly, all NRC staff review of the requested change is paid for by the licensees. For all this expenditure, the licensee has no assurance of success.

From the licensee's perspective, the benefits expected if the change is approved has to be more than all the costs of obtaining NRC approval. In past licensee attempts to "risk-inform" NRC requirements (not even the regulations), the benefits of changes made to the requirements sometimes did not outweigh the costs which is why not many licensees propose risk-informed changes to the regulations.

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Some examples concerning unsuccessful licensee attempts to change the regulations are as follows. At Arkansas Nuclear One, there were two submittals with two negative NRC safety evaluation reports written for changing the time for hydrogen monitoring before the NRC approval letter of September, 28, 1998, for Task Zero at ANO. Before personnel from San Onofre were successful in the San Onofre Task Zero exemption request from hydrogen control requirements, personnel from Waterford 3 tried to obtain the same change and were turned down by the staff of the NRC.

Another example: it has been over seven months since my petition for rulemaking to change 10CFR50.44, was started. This petition for rulemaking is an extrapolation based on the Arkansas Nuclear One Task Zero and the San Onofre Task Zero of the Whole Plant Study both of which were approved by the NRC staff. The latest word from the NRC staff is that my petition will not be the recommended course of action coming out of the Option 3 effort and therefore the NRC staff is not processing my petition for either approval or disapproval. In the meantime, licensees cannot implement changes to make the nuclear electric power units "safer" and more economic with respect to hydrogen control except through the 10CFR50.12 Specific exemption process.

Mr. Riccio has it backward. Even if the regulations didn't require a 10CFR50.109 Backfitting analysis, equal treatment would require that the staff of the NRC perform a detailed backfit cost/benefit analysis for any change in the regulations initiated by the NRC staff, even "voluntary" changes.

Performance Technology

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August 24, 2000

Dr. Ashok Thadani
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U. S. Nuclear Regulatory Commission
11545 Rockville Pike
Rockville, MD 20852-2738

Dear Dr. Thadani:

I appreciate the time that you and other NRC personnel took to talk to me on August 18, 2000, about Option 3 of SECY 98-300 in response to my letter to you dated July 19, 2000. The meeting was very valuable to me because it allowed me to recognize the differences between what the nuclear industry has been proposing in the Whole Plant Study and what the NRC staff is now proposing in Option 3. Clearly, there are major differences between the respective approaches. My summary of the respective positions and differences as discussed in the meeting is as follows.

The objective of Option 3 is for NRC personnel to write a set of deterministic regulations for existing nuclear electric power units in a manner that will assure that the public health risk to individuals and society from these nuclear units is below (more restrictive), on a risk graph, the risk level defined by the Quantitative Health Effects Objectives ("how-safe-is-safe-enough") of the 1986 NRC Policy Statement on Safety Goals for Operating Nuclear Power Plants. The key principles are "defense-in-depth," "safety goals," and "uncertainty." The implementation of the Option 3 objective is accomplished by writing regulations that are based on separate "partition factors" (defense-in-depth) that, when taken in the aggregate, guarantee that the public health risk is below the Quantitative Health Effects Objectives (safety goals) by a substantial margin (uncertainty). This program is "voluntary" except that if regulations are added to achieve the Option 3 objective and the added regulations meet the criteria of 10CFR50.109, Backfitting; then the added requirements may be mandatory.

The objective of the Whole Plant Study is to use insights from Probabilistic Risk Assessments to change the existing regulations for existing nuclear electric power units to achieve "reasonable assurance of adequate protection of public health and safety" in a more effective and efficient manner (regulations will address significant risk items by cost effective means). The key principles are "adequate protection;" 10CFR50.109, Backfitting; and the Quantitative Health Effects Objectives ("how-safe-is-safe-enough"). The implementation of the Whole Plant Study objective is accomplished by retaining

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portions of existing regulations that are effective and efficient (adequate protection); deleting portions of existing regulations that are not effective and efficient; and, where appropriate, adding regulations that meet the criteria of 10CFR50.109; except that no regulations are added below the risk level of "how-safe-is-safe-enough."

To me it is clear that there are major differences between the two approaches. The objectives are different, the key principles are different, and the implementation strategies are different. The only common element may be the use of insights from Probabilistic Risk Assessments. The Quantitative Health Effects Objectives of the 1986 NRC Safety Goal Policy Statement and 10CFR50.109, Backfitting, are used in each program but their use is drastically different in such a manner that I hesitate to say these items are common to each program. In my opinion, the most important difference in the programs is that Option 3 does not accept the concept that substantial compliance with the existing regulations provides "reasonable assurance of adequate protection of public health and safety" while this concept is the starting point for the work in the Whole Plant Study. The implementation of regulations based on the recommended Option 3 "partition factors" would represent a "ratcheting" of the level of safety of nuclear electric power units to a standard more restrictive than that which the Commission has defined as "safe enough."

I believe the discussion we had on August 18, 2000, was very beneficial to all concerned. Again, thank you for taking the time to discuss this matter with me.

Sincerely,

Bob Christie

Bob Christie

cc: Samuel J. Collins, NRR
Dr. Dana Powers, ACRS

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Proceedings of PSAM-3 Meeting - Crete, Greece (1997)

An Assessment of the Risk-Impact of Reactor Power Upgrade for a BWR-6 MARK-III Plant

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and

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1. Introduction

The Leibstadt nuclear power plant, a General Electric (GE) Boiling Water Reactor/Model 6 (BWR6) with MARK III containment, is located on the Swiss bank of the Rhine River, in the canton Aargau. The plant is operated by the Kernkraftwerk Leibstadt AG (KKL) utility, and began commercial operation on December 15, 1984, at a power rating of 3,012 MW(t). The reactor power was later uprated to the current level of 3,138 MW(t). The utility is planning to upgrade the Leibstadt power by an additional 14.7% to 3,600 MW(t). The KKL utility is also in the process of gradually replacing the existing GE8 reactor fuel with the ABB/SVEA96 fuel design.

An independent regulatory Probabilistic Safety Analysis (PSA) model has been developed to assess the severe accident risk implications of the proposed 14.7% power upgrade, and the current modifications in the fuel design. The objective of this paper is to discuss the methodological aspects of the study within the level-2 PSA framework.

2. Regulatory PSA Model

The approach to regulatory PSA studies in Switzerland is discussed in [1]. In this approach, the utility PSA model is reviewed, and requantified using alternative models, data, and procedures.

The present utility level-1 PSA for Leibstadt is being performed by Electrowatt Engineering (EWI) Services (UK) Ltd in cooperation with RELCON of Sweden. The study, when completed, will include the core damage potential of various internal and external event initiators. However, to date, only the level-1 PSA results for internal events have been independently confirmed, since the utility is in the process of completing the external-events part of the study.

The starting point for the present regulatory PSA model is the utility supplied study.

This study was used by the Inspectorate and contractors to develop an independent set of system fault trees and event trees, using the SPSA computer code [2]. The various independent, common cause, human error, and maintenance unavailability data used in the original utility study was compared with generic data and other plant-specific studies, to arrive at the basis for quantification of the present regulatory PSA model.

In addition, an independent level-2 PSA model was developed and used for the present regulatory evaluation.

3. Impact on Core Damage Profile

The impact of the power upgrade on core damage frequency (i.e., results of the level-1 PSA study) have not been quantitatively assessed. The main level-1 PSA issues that can be impacted by the proposed power upgrade include (1) the decay heat removal success criteria, (2) the dynamic operator actions, and (3) the reduced design safety margins for the important mitigating systems.

The effect of power level and fuel design on decay heat removal "success criteria" is minimized by the Inspectorate requirements that the existing decay heat removal success criteria be maintained for the planned power upgrade condition at Leibstadt.

There is expected to be some influence on the success probability of the dynamic human actions at the proposed uprate power conditions, because one major factor that affects the probability of operator errors is the time available to respond to an event. However, a review of all important operator actions in all the regulatory PSA level-1 results, shows that those actions would not be substantially affected by the expected reduction in the available response time.

The system design safety margin for the important mitigating systems, particularly High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) systems, are expected to provide sufficient margins that the increased power level would not affect the overall decay heat removal capabilities.

Therefore, no discernable impacts resulting from the power upgrade on the internal events mean core damage frequency of 4.4×10^{-6} per reactor year is expected. The contribution of various initiated events to the mean core damage frequency consists of 11% due to ATWS, 17% due to LOCAs, 11% due to transients with loss of decay heat removal, 61% to all other transients, and <<1% due to ISLOCAs. The calculated uncertainties in the core damage frequency ranges from about 7×10^{-8} to about 1.5×10^{-5} per reactor year.

4. Impact on Progression of Severe Accidents

The impact of the proposed reactor power upgrade and the fuel design modifications on the severe accident progression, fission product releases, and containment challenges applicable to the Leibstadt nuclear power plant is described in this section. Table 1 lists the issues that are expected to be impacted by the power upgrade and the fuel design changes, including a qualitative ranking of their intrinsic uncertainties.

Table 1 Intrinsic uncertainties for the issues impacted by reactor power and fuel

Issues Impacted		Intrinsic Uncertainty
1. Core Radiological (Isotopic) Inventory		Medium
2. Decay Heat		Low
3. Time of Core Uncovery		Low
4. Core and Structural Heat up Rates		Low
5. Metal Oxidation/Hydrogen Generation		Medium
6. Fuel Damage and Melt Relocation		High
7. Time of RPV Failure		High
8. Extent of MCCI and Non-condensible Gas Generation		Medium
9. In-Vessel Fission Product Release		High
10. In-Vessel Retention of Fission Products		High
11. Fission Product Retention in Pressure Suppression Pool		High
12. Ex-Vessel Fission Product Release		High
13. Fission Product Retention in Drywell and Wetwell Compartments		Medium
14. Time of Containment Failure/Containment Filtered Vent		Medium
15. Early Containment Loads	Combustion	Medium
	Direct Containment Heating	High
	Ex-Vessel Steam Explosions	High
16. Late Containment Loads	Combustion	Low
	Slow Pressurization	Low
	Basemat Penetration	Low

The assignment of low, medium and high ranks to various uncertainty issues is intended to guide the degree by which the impact of the power upgrade and fuel design changes can be characterized and quantitatively assessed. Specifically:

Low Uncertainties - The intrinsic uncertainties are small relative to the expected changes resulting from the reactor power level and fuel design. Therefore, the expected impact of the power and fuel modifications on the characterization of the issue can be quantified with confidence, as guided by relatively good knowledge of the governing physical phenomena associated with the issue.

Medium Uncertainties - The intrinsic uncertainties are not small relative to the expected changes resulting from the reactor power level and fuel design. Therefore, only trends associated with the impact of the planned changes can be quantified with confidence, as guided by relatively incomplete knowledge of the governing physical phenomena associated with the issue.

High Uncertainties - The intrinsic uncertainties are large relative to the expected changes resulting from the reactor power level and the fuel design. Therefore, assessment of the expected trends of the impact of the changes on the characterization of the uncertain severe accident issue is difficult under all conditions of interest.

Analysis of various issues listed in Table 1 has demonstrated [2] that the most significant impact of the power upgrade result from the increased radioactive inventory, and the time acceleration of events due to the increased decay heat level at the uprated power conditions. The issues of medium uncertainty were assessed and it was concluded that even though some trends could be established in terms of the influence of reactor power and fuel design changes, nevertheless, the overall impact of power and fuel design is not significant.

On the other hand, given the large degree of uncertainty associated with some of the severe accident issues in Table 1 exemplified by Figure 1, quantitative assessment of the potential impact of the relatively small increase in power level (as compared with the large intrinsic uncertainties) is very difficult, within the current state of the art. Figure 1 shows that the accident sequence variabilities are greater than the variations resulting from power and fuel changes.

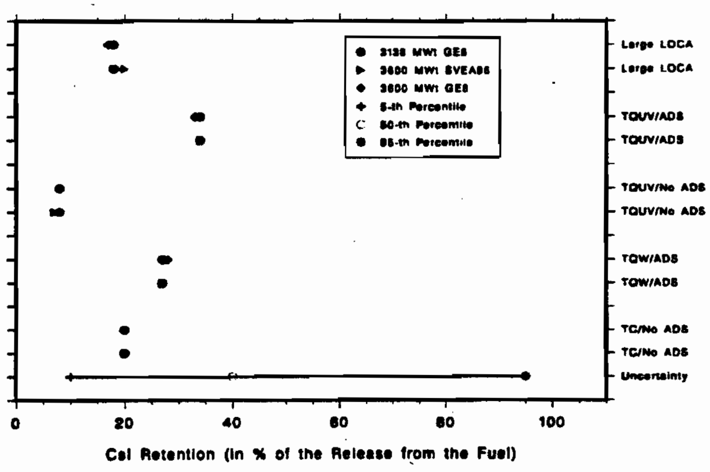


Figure 1 Calculated impact of power and fuel changes on CsI retention for various accident sequences and comparison with uncertainty ranges

TIME TO CORE UNCOVERY
TIME TO CONTAINM. FAILURE
TIME TO FISSION PROD. RELEASE
INVERS. PROP TO POWER LEVEL

The influence of reactor power on the time evolution of accidents can be demonstrated through a simple analysis of loss of coolant inventory during a transient event not involving ADS activation. Assuming that the decay power can be represented by a simple polynomial, it can be easily shown that the time to core uncovery follows [2]:

$$\tau_{uc} = [\ 3.5\ \frac{m_{taf}\, h_{fg}}{Q(0)} + t_o^{0.7}\ ]^{1.43} \qquad (1)$$

where $Q(0)$ is the initial reactor power level in MW(t), t_0 is the time since reactor shutdown in seconds, and m_{taf} and h_{fg} represent the water mass above the top of active fuel, and heat of vaporization, respectively.

Equation (1) clearly demonstrates that the time to core uncovery is inversely proportional to reactor operating power. This inverse relationship is seen to be weakly non-linear. The time to core uncovery is shorter by about 20% at the 14.7% power uprate condition (assuming the water inventory is the same, even though the reactor water inventory is slightly smaller at the uprated power condition due to the increased void formation as compared with the present power). A similar power-dependence can be shown for containment failure time (for accidents involving slow over-pressurization of the containment), and thereby, the time of fission product release to the environment.

The level-2 PSA model includes the quantitative impact of uprated power on low uncertainty issues; while, the model also includes the quantitative impact of the trends associated with severe accident progression issues of medium uncertainty in Table 1. However, the model assumes that the impact of power and fuel design changes cannot be quantified for those issues that are classified with high relative uncertainties. In addition, the PSA model also incorporates the impact of time evolution of accidents on radioactive decay, and transmutation in arriving at the risk of activity released to the environment.

5. Results

In the present study, risk is defined as a product of the release activity and the release class frequency (i.e., activity per reactor year), integrated over all possible release classes. Activity is defined as the sum of the fractions of the total core inventory which are released for each release class, at the time of release (to account for transmutation and radioactive decay). These results may be interpreted as the risk of activity of release in the immediate vicinity of the plant.

Table 2 shows the comparison of risk of activity of release (excluding noble gases) for the existing power and fuel (3,138 MW, GE8), the uprate power and existing fuel (3,600 MW, GE8), and the uprate power and new fuel design (3,600 MW, SVEA96). It should be noted that noble gases decay very quickly, and that their contribution to offsite risk is minor (i.e., they can contribute only to the inhalation dose). It is seen that the mean risk of activity of release increases by about 30% due to the 14.7% increase in reactor power. In addition, the risk impact of the proposed ABB fuel (i.e., SVEA96) is relatively small. The main reason for

$RISK = \sum_{\text{All Rel. Classes}}$ Release Activity × Release Class frequency

↑

$\sum$ fractions of total core inventory any impact on risk due to the change in fuel results from an approximately 30% increase in Zircaloy surface area for the SVEA96 fuel as compared with GE8; even though, the total Zircaloy mass is smaller by 20% for the SVEA96 fuel viz-a-viz the GE8 fuel (This has a small impact on the probability of early containment failure due to hydrogen combustion).

Table 2 Impact of power and fuel design on the estimated mean risk

Power, Fuel Design	Risk of released activity (Bq/yr)	Change in risk relative to 3,138 MW and GE8 Fuel
3,138 MW(t), GE8	6.27×10^{11}	NA
3,600 MW(t), GE8	8.02×10^{11}	28%
3,600 MW(t), SVEA96	8.14×10^{11}	30%

Figure 2 shows the effect of uncertainties on the risk of activity of release (excluding noble gases) at the existing power and fuel versus the proposed power upgarde (including the new fuel).

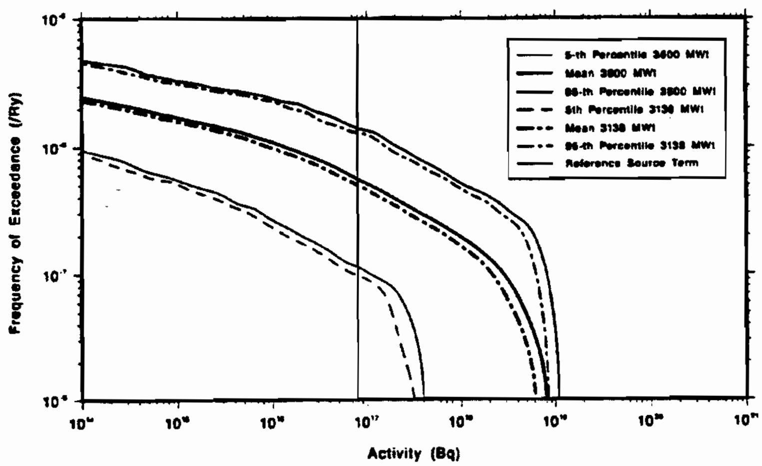


Figure 2 Impact of power uprate and fuel design on the risk of released activity

The uncertainties for the released activity range from three decades at high frequencies to nearly a decade at very low frequencies. On the other hand, the uncertainties in frequency of releases is seen to range from about a factor of 5 for low releases, extending to several decades at very high releases.

References

1. U. Schmocker, et al., "Approach to Regulatory Review of Swiss Probabilistic Safety Assessments," Proceedings of Use of Probabilistic Safety Assessment for Operational Safety, PSA'91, 125-133, IAEA-SM-321/10, Vienna (1991).

2. M. Khatib-Rahbar, et al, "Regulatory Evaluation of Leibstadt Probabilistic Safety Assessment," Energy Research, Inc. ERI/HSK 95-303, HSK 12/144, Volumes I & II (1996).

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
475th MEETING

DRAFT REPORT:
"CAUSES AND SIGNIFICANCE OF DESIGN-BASIS ISSUES AT U.S. NUCLEAR POWER PLANTS"

RONALD LLOYD

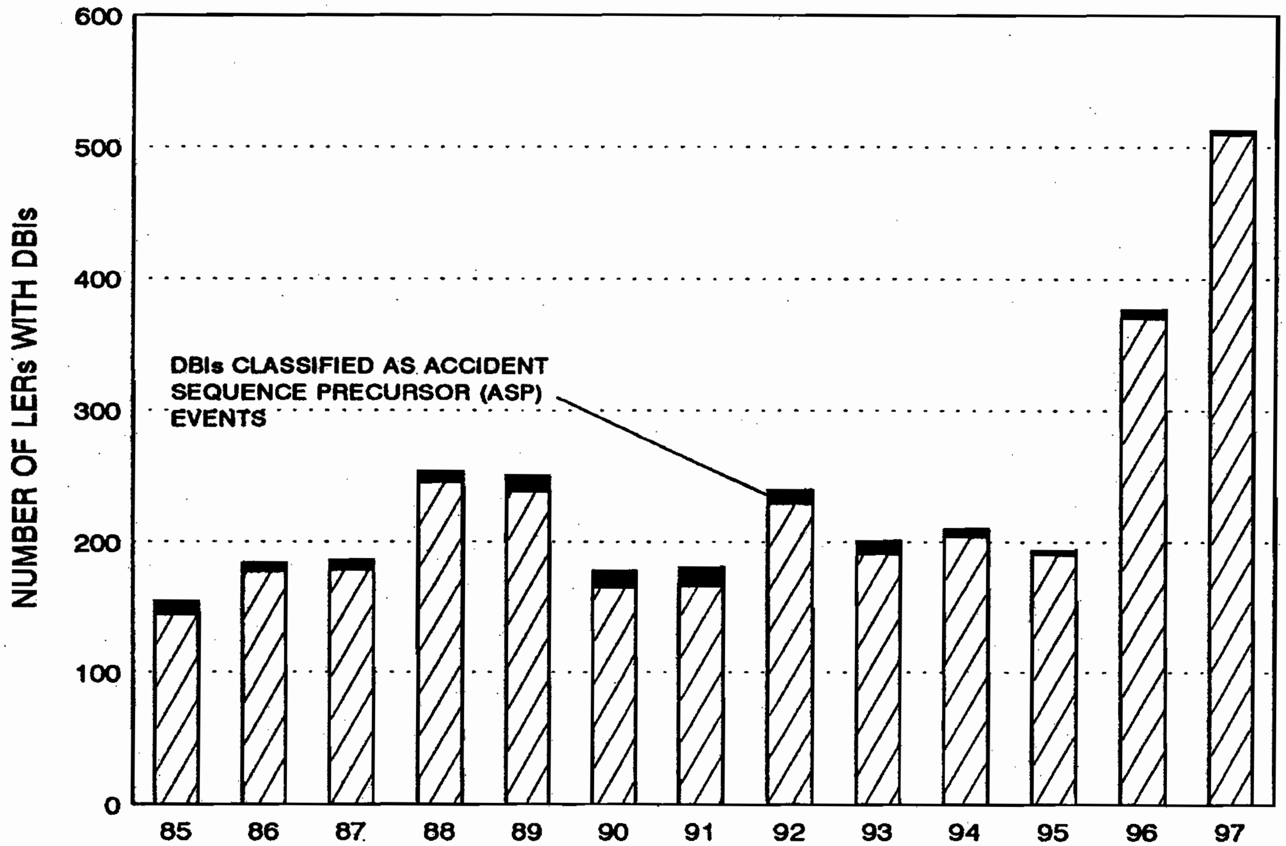
OFFICE OF NUCLEAR REGULATORY RESEARCH

AUGUST 29, 2000
ROCKVILLE, MARYLAND

7

Reporting Trends for DBIs from 1985 Through 1997

1. Over 3100 Total LERs with DBIs
2. Increases in the Number of Reported DBIs Coincided with NRC Initiatives
3. Over 500 LERs with DBIs in 1997 (Focus Area)
4. Small Percentage of DBIs Classified as Accident Sequence Precursor (ASP) Events
5. Over 80 Percent of DBIs Reported as "Unanalyzed Conditions"



NUMBER OF LERs WITH DBIs
600
500
400
300
200
100
0
DBIs CLASSIFIED AS ACCIDENT SEQUENCE PRECURSOR (ASP) EVENTS
85
86
87
88
89
90
91
92
93
94
95
96
97

DBI Risk-Informed, Deterministic Significance Framework

LERs Were Assessed in Four Different Areas

1. DBI Risk Category
   - Potential
   - Minimal
   - None

2. Safety Demand Present
   - Yes
   - No

3. Effect Type
   - Actual Event
   - Potential Event

4. Effect Extent
   - Failed System
   - Degraded System
   - Degraded or Failed Train

DBI Risk-informed, Deterministic Significance Framework

GROUP	DBI SAFETY SIGNIFICANCE CATEGORY		DBI RISK CATEGORY			DBI DETERMINISTIC SIGNIFICANCE CLASSIFICATION						
						Safety Demand		Effect Type		Effect Extent		
			Potential	Minimal	None	Yes	No	Actual	Potential	Failed System	Degraded System	Degraded or Failed Train
I	1	a	x			x		x		x		
		b	x			x		x			x	
		c	x			x		x				x
	2	a	x				x	x		x		
		b	x				x	x			x	
		c	x				x	x				x
	3	a	x				x		x	x		
		b	x				x		x		x	
		c	x				x		x			x
II	4	a		x		x		x		x		
		b		x		x		x			x	
		c		x		x		x				x
	5	a		x			x	x		x		
		b		x			x	x			x	
		c		x			x	x				x
	6	a		x			x		x	x		
		b		x			x		x		x	
		c		x			x		x			x
	7	-			x							

OBSERVATIONS

The Most Common Causes[1] of DBIs Were:

• Original Design Error	72 %
• Procedure Deficiency	28 %
• Human Error	22 %

[1]More than one cause was generally listed for each DBI

There Was a Significant Variation Among Plants in the Number of Reported DBIs

1997 Data

	Range
• DBIs Reported	0 - 37
• Engineering Inspection Hours	90 - 3700
• Engineering Inspection Hours per DBI	15 - 630

A Few Safety-related Systems Accounted for about Half of the DBIs

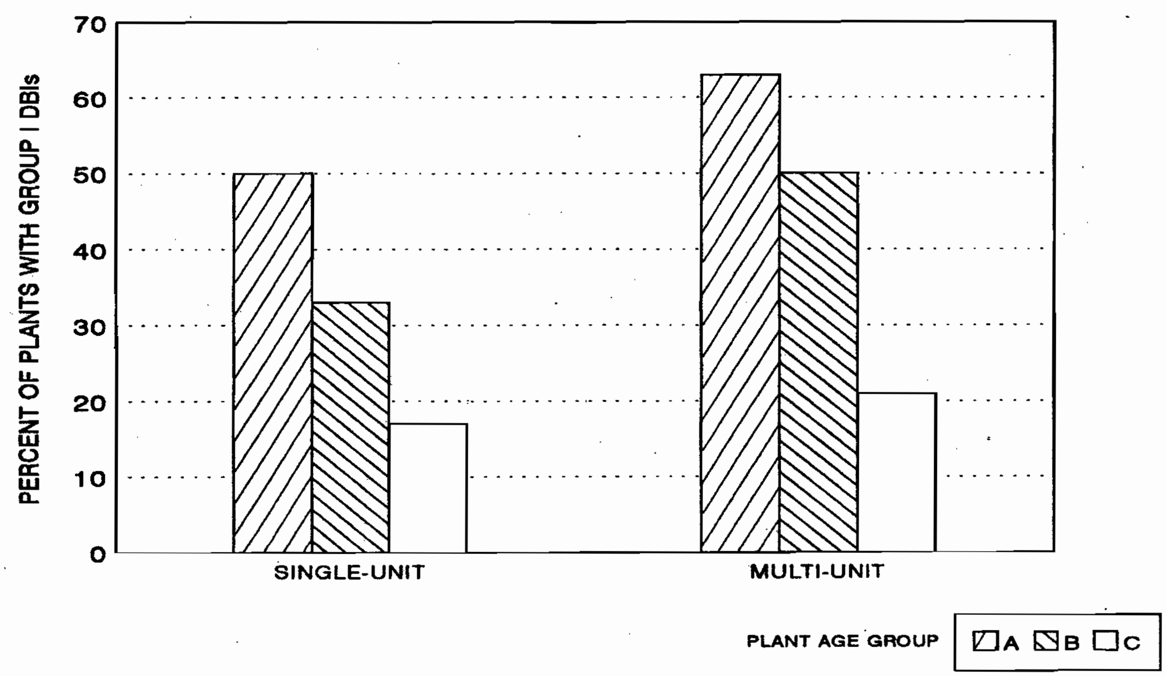
System	Group I and Group 2 DBIs	Group I DBIs Only
Emergency Core Cooling	13%	34%
Emergency ac/dc Power	11%	18%
Containment and Containment Isolation	7%	12%
Primary Reactor	7%	2%
Auxiliary/Emergency Feedwater	6%	3%
Emergency Service Water	6%	4%

- Group 1: Potentially Risk Significant (19%)
- Group 2: Minimal or no Risk Significance (81%)

Older Plants Generally Reported More DBIs Than Newer Plants

• Group A: Licensed Between 1964 and 1974 (44 Units)	5.6 DBIs/Unit
• Group B: Licensed Between 1975 and 1984 (35 Units)	4.7 DBIs/Unit
• Group C: Licensed Between 1985 and 1997 (31 Units)	3.1 DBIs/Unit

Group I DBIs Were More Likely at Multi-Unit Sites than Single-Unit Sites



The Percent of LERs with DBIs That Were ASP Events Steadily Decreased, While the Number of DBIs Increased

During 1990–1997:

- The Percent of DBIs Classified as ASP Events Decreased From Approximately 8% to less than 1%

- The Total Number of ASP-DBI Events Decreased from 13 to 3

- The Total Number of ASP Events from All Causes Decreased From 28 to 5

Important accident sequence precursor events (1992-1997)

Plant	Event Date	Description	Involved DBI	BWR/ PWR	CCDP
Ft. Calhoun	07/03/92	Reactor Trip Due to Invertor Malfunction and Subsequent Pressurizer Safety Valve Leak	No	PWR	2.5×10^{-4}
Robinson 2	08/22/92	Unusual Event Due to Loss of Off-Site Power and Reactor Trip	No	PWR	2.1×10^{-4}
Turkey Pt. 3, 4	08/24/92	Loss of Offsite Power Due to Hurricane Andrew	No	PWR	1.6×10^{-4}
Oconee 2	10/19/92	Loss of Off-site Power and Unit Trip Due to Management Deficiencies, Less than Adequate Corrective Action Program	No	PWR	2.1×10^{-4}
Sequoyah 1, 2	12/31/92	Reactor Trip as a Result of a Switchyard Power Circuit Breaker Fault and a Unit 2 Entry Into Limiting Condition for Operation [LCO] 3.0.3 when Both Centrifugal Charging Pumps were Removed from Service	No	PWR	1.8×10^{-4}
Catawba 1, 2	02/25/93	Technical Specification 3.0.3 Entered Due to Inoperable Pump Discharge Valves	Yes	PWR	1.5×10^{-4}
Perry	04/19/93	Excessive Strainer Differential Pressure Across the residual heat removal (RHR) Suction Strainer Could Have Compromised Long Term Cooling During Post-LOCA Operation	No	BWR	1.2×10^{-4}
LaSalle 1	09/14/93	Unit 1 Scram and Loss of Off-Site Power Due to Bus Duct Water Intrusion	No	BWR	1.3×10^{-4}
Haddam Neck	02/16/94	Automatic 480 Volt Bus Transfer Failure Due to Circuit Breaker Malfunction	No	PWR	1.4×10^{-4}
Wolf Creek	09/17/94	Reactor Coolant System Blows Down to Refueling Water Storage Tank During Hot Shutdown	No	PWR	3.0×10^{-3}
St. Lucie 1	08/02/95	Failed PORVs, Reactor Coolant Pump, Seal Failure, Relief Valve and Subsequent Shutdown Cooling System Unavailability, Plus Other Problems	Yes	PWR	1.1×10^{-4}
Wolf Creek	01/30/96	Loss of Circulating Water Due to Icing on Traveling Screens Causes Reactor Trip	No	PWR	2.1×10^{-4}
Catawba 2	02/06/96	Loss of Off-Site Power Due to Electrical Component Failures	No	PWR	2.1×10^{-3}
Haddam Neck	08/01/96	Potential for Inadequate RHR Pump Net Positive Suction Head During Sump Recirculation	Yes	PWR	1.1×10^{-4}

Group I DBIs Varied by NRC Region

Percent of Plants with Group I DBIs

Region I	52%
Region II	36%
Region III	59%
Region IV	19%

For 1995–1997, DBIs Appeared to Correlate with NRC Engineering Inspection Effort

- An Increase in the Number of Inspection Hours Generally Resulted in an Increase in the Number of Reported DBIs

Plants with the largest total number of DBIs (1990-1997)

Plant Name	Number of DBIs	Number of ASP-DBIs
Crystal River 3	93	0
Millstone 1	85	0
Indian Point 3	59	0
Millstone 3	55	0
Palisades	55	0
Fort Calhoun	45	2
Millstone 2	43	1
Maine Yankee	41	1
Dresden 2	41	0
Haddam Neck	36	3
Salem 1	36	1

The Importance and Applicability of DBIs Discussed in NRC Generic Communications Occasionally Takes Several Years for Licensees to Recognize and Address

Generic communications on pressurized-water reactor containment sump strainer and boiling-water reactor emergency core cooling system strainer clogging

Date Issued	Information Notice/ Bulletin Number	Title
05/88	IN 88-28	Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage
11/89	IN 89-77	Debris in Containment Emergency Sumps and Incorrect Screen Configurations
01/90	IN 90-07	New Information Regarding Insulation Materials Performance and Debris Blockage of PWR Containment Sumps
09/92	IN 92-71	Partial Plugging of Suppression Pool Strainers at a Foreign BWR
04/93	IN 93-34	Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment
05/93	IEB 93-02	Debris Plugging of Emergency Core Cooling Suction Strainers
10/95	IEB 95-02	Unexpected Clogging of a RHR Pump Strainer While Operating in Suppression Pool Cooling Mode
10/95	IN 95-47	Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage
05/96	IEB 96-03	Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors
10/96	IN 96-059	Potential Degradation of Post Loss-of-Coolant Recirculation Capability as a Result of Debris
05/97	IN 97-027	Effect of Incorrect Strainer Pressure Drop on Available Net Positive Suction Head

CLARIFYING THE DEFINITION OF DESIGN BASES

Presentation to the Advisory Committee on Reactor Safeguards
August 29, 2000
Stewart Magruder
Office of Nuclear Reactor Regulation
(301) 415-3139

OBJECTIVE

- Develop guidance that provides a clearer understanding of what constitutes design bases information as defined in 10 CFR 50.2

10 CFR 50.2 DEFINITION

Design Bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.

RELEVANCE OF DESIGN BASES

- "Design Bases" used in following regulations:
  - 50.34 (FSAR content)
  - 50.59 (Changes - Effective early 2001)
  - 50.72, 50.73 (Reporting - Until early 2001)
  - Appendix A to Part 50 (GDC)
  - Appendix B to Part 50 (QA)
- Useful for evaluating degraded and nonconforming conditions

BACKGROUND

- Engineering team inspections (Late 1980s)
- Industry guidance (NUMARC 90-12)
- NUREG-1397 (February 1991)
- Commission Policy Statement (August 1992)
- Millstone/Maine Yankee (1996)
- Nine Mile Point - reporting issue (1997)
- Revised industry guidance (NEI 97-04)
- Staff committed to develop regulatory guidance

Draft Guidance (DG-1093)

- Endorsed Appendix B of NEI 97-04 with two exceptions
- Briefed ACRS on 10/1/99 and 11/5/99
- Published for comment 4/12/00
- 11 comment letters - supportive of effort
  - NEI
  - Utilities (9)
  - NRC Region III

DG-1093 EXCEPTIONS

- Defense-in-Depth
  - Important aspect of principal design criteria
  - Provides standard for judging design bases

- Relationship to UFSAR
  - Design bases may change as a result of plant modifications to ensure compliance with current requirements
  - Supporting design information is required to be included in UFSAR

PROPOSED FINAL REGULATORY GUIDE

- Endorses Appendix B of NEI 97-04 with no exceptions
- NEI modifications addressed staff concerns

GENERAL GUIDANCE

- *Design bases functions:* Functions performed by systems, structures and components that are (1) required, or otherwise necessary to comply with, regulations, license conditions, orders or technical specifications, or (2) credited in licensee safety analyses to meet NRC requirements.

- *Design bases values:* Values or ranges of values of controlling parameters established as reference bounds for design to meet design bases functional requirements. These values may be (1) established by NRC requirement, (2) derived from or confirmed by safety analyses, or (3) chosen by the licensee from an applicable code, standard or guidance document.

SUMMARY

- Staff and industry have reached a common understanding of the term
- Public comments support guidance document
- Request ACRS letter approving publication of final Regulatory Guide endorsing industry guidance

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
AUGUST 29, 2000

BRIEFING ON AP1000 STANDARD PLANT DESIGN
by
Jerry N. Wilson - Senior Policy Analyst /RLSB/NRR

Background

- NRC certified the AP600 design on January 24, 2000

- Westinghouse
  - Designing a 1000 Mwe version of the AP600
  - Considering applying for design certification
  - Requested a pre-application review to determine the scope & cost of a design certification review for the AP1000 standard plant design

AP1000 Design

- Increase power to reduce cost/KW
- Minimize changes to AP600 design
- Increase number & length of fuel assemblies
- Increase height of reactor vessel
- Increase capacity of reactor coolant pumps
- Increase pressurizer volume
- Increase size of Steam Generators
- Increase containment volume & design pressure
- Increase capacity of ADS

AP1000 Application

- ➤ Retain ~ 80% AP600 design control document
- ➤ Rely on the AP600 test program for AP1000
- ➤ Use AP600 analysis codes with minor modifications
- ➤ Use portion of the AP600 PRA, Level 1
- ➤ Defer selected design activities to combined license
- ➤ Use the same (AP600) industry codes & standards

AP1000 Pre-application Review

- Phase one - complete
  - NRC met with Westinghouse on April 27, 2000
  - Westinghouse requested start with May 4th letter
  - Westinghouse identified issues with May 31st letter
  - ACRS identified issues with June 21st letter
  - NRC - issues, information & estimates - July 27th

- Phase two - requested
  - Westinghouse requested start in their August 28th letter and identified deliverables and schedule
  - NRR will use PBPM process to determine workload priority

- Phase three - Design Certification Review?

10

Westinghouse
Electric Company LLC

Advanced Plant Development
Box 355
Pittsburgh Pennsylvania 15230-0355

DCP/NRC1465
Project 711

August 28, 2000

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Samuel J. Collins

Dear Mr. Collins,

Thank you for your letter of July 27, 2000 that provided the results of the NRC staff's Phase 1 assessment of the AP1000 pre-application review. Westinghouse desires to proceed at this time with a portion of Phase 2 review as indicated below. The staff estimates exceeded our available budget, and we have prioritized the review tasks to remain within our budget limitations. It is possible that additional tasks may be added to the review if we can obtain additional financial support. These tasks would be added by a separate future letter request and a schedule will be determined at that time. At this point we would like the NRC staff to plan to proceed with resolving the following items:

- Applicability of AP600 Test Program to AP1000
- Applicability of AP600 Analysis Codes to AP1000
- AP1000 Design Acceptance Criteria
- AP1000 Exemptions

On the two items that were deferred, the following comments are provided. After considering the staff comments on Item 4 - AP1000 Probabilistic Risk Assessment, it is our belief that this item will not meet our criteria for Phase 2 work of contributing significantly to the efficiency of the Design Certification Review. Therefore, Westinghouse proposes to defer the Probabilistic Risk Assessment to Phase 3. Westinghouse remains interested in performing Item 1 - Scope of NRC Review, but will defer this task temporarily based upon the funding available.

Westinghouse desires to initiate the NRC review of the selected items on November 1, 2000 and we plan to provide the appropriate deliverables to you prior to that time as requested in your letter. Westinghouse has provided in Enclosure 1, the information requested in your letter to assist the staff to prioritize this requested pre-application against the four NRC goals. Westinghouse requests that a target schedule be established for the pre-review and requests that a target completion be established of February 2001. Enclosure 2 provides a description of our deliverables for Phase 2.

The test reviews are the most important reviews of the Phase 2 program. We renew our request to NRC to make every effort to assign the reviewers who performed the test assessment for the AP600 Design Certification Review to make NRC activities and decisions as effective, efficient, and realistic as possible.

Westinghouse has in Enclosure 3, provided comments on the NRC staff "Phase 1 Results" and on the ACRS letter on the pre-review. For the most part, the comments are clarifications of whether certain items are in the scope of the Phase 2 pre-review. There are also a few technical clarifications or comments provided in the interest of efficient work process. Westinghouse is willing to meet to discuss any of the comments but does not consider a meeting necessary unless the staff would like to discuss the topic further. Westinghouse has in Enclosure 4 provided comments on the ACRS issues related to the review of the AP1000 design.

Very truly yours,

W E Cummins

W. E. Cummins, Director
Advanced Plant Development

cc: J. N. Wilson

/Enclosures

1) "Westinghouse Assessment of Phase 2 Versus NRC Performance Goals"
2) "Descriptions of Westinghouse Submittals for AP1000 Phase 2 Goals for Application Review"
3) "Westinghouse Comments on Phase 1 Results"
4) "Westinghouse Categorization of ACRS Issues Related to the Review of the AP1000 Design"

Enclosure 1
Westinghouse Assessment of Phase 2 Versus NRC Performance Goals

Westinghouse is pleased to provide the following qualitative assessment of the AP1000 pre-application review (Phase 2) effort for the purpose of assisting NRR staff in prioritizing the Phase 2 review against the NRC's projected FY2001 workload. Phase 2 of the pre-application review will provide the NRC staff's evaluation of several key issues that Westinghouse has requested be reviewed to determine the optimum process for and the feasibility of a design certification application. As requested in the NRC letter of July 27, 2000, Westinghouse's assessment is presented in the context of the NRC's four performance goals as amplified in the NRC's Strategic Plan [NUREG-1614, Vol.2, Part 2]. Westinghouse believes that the measures that the NRC has selected to demonstrate the performance goal achievement are essentially keyed to licensing actions related to Operating Reactors and thus do not generally apply directly to this advanced reactor review process. Nevertheless, we have identified many aspects of the Phase 2 review that fulfill the descriptions of the underlying measures identified in the Strategic Plan. We have also provided Westinghouse's assessment (high, medium, low) of the degree to which we believe the Phase 2 evaluation meets the intent of each performance goal.

PERFORMANCE GOAL #1: Maintain safety, protection of the environment, and the common defense and security.

Westinghouse Performance Goal Ranking: High

This is the preeminent performance goal that takes precedence over all other performance goals. To achieve it, NUREG-1614 states that the NRC will give priority to those licensing actions and exemptions that provide the greatest safety benefit to the public. Phase 2 is geared to meet this expectation in the fullest measure.

While current operating reactors have proven very safe, the NRC-approved detailed probabilistic safety analysis associated with the AP600 ALWR design shows a safety factor improvement of two orders of magnitude over typical operating reactors. However, the AP600 was designed and certified prior to the deregulation of the US electricity market. In the deregulated market, each generator must compete favorably against alternative sources strictly on financial merits. In order for new nuclear power plants to be viable in this deregulated market, Westinghouse believes that further cost reductions must be achieved. The AP1000 utilizes the passive safety features certified on the AP600 but will be constructed at a much lower cost per kilowatt of generating capacity. By certifying a cost competitive design, Westinghouse believes that the NRC would be providing a very substantial benefit to the public safety and to protecting the environment. While the direct application is in the future, the magnitude of the safety benefit, in Westinghouse's opinion, compares favorably to the NRC's planned activities of which Westinghouse is aware. This benefit, however, will not be achieved without first performing the Phase 2 assessment.

NUREG-1614 also states that the NRC will encourage applicants, vendors, and others to inform the NRC at the earliest opportunity of planned future reactor activities so that the NRC will be prepared to respond. Phase 2 also meets the goal of giving the NRC the maximum advanced notification of the certification effort, and will give the NRC an excellent assessment of the key issues, the effort, and expertise that will be necessary for the certification effort, if Westinghouse and the U.S. nuclear industry elect to pursue design certification.

PERFORMANCE GOAL #2: Increase public confidence.

Westinghouse Performance Goal Ranking: Medium

10 CFR Part 52 was designed to make public participation more meaningful by affording the public the opportunity to interact with the NRC and the applicant at a stage prior to the commencement of construction activities. Some of the strategies to increase public confidence are clearly implemented in a plant Design Certification Program. The program includes processes that recognize public interests and concerns. For elements of the public that support a nuclear power option in electricity generation, an efficient approach to the safety evaluation of the AP1000 will enhance the public confidence in the NRC. For elements of the public that do not support the nuclear power option, the open and inclusive Design Certification safety evaluation process should enhance the perception of the NRC as a strong, fair regulator interested in timely public involvement. The dramatic simplification of the passive plant safety systems increases the potential for public understanding and involvement in the process.

It is clear that the public confidence in the NRC was positively impacted by completion of the AP600, System 80+ and ABWR design certification efforts. It is expected that similar positive impacts would be achieved from a successful AP1000 review.

PERFORMANCE GOAL #3: Make NRC activities and decisions more effective, efficient , and realistic.

Westinghouse Performance Goal Ranking: High

The proposed AP1000 Pre-Application Review is an excellent opportunity to demonstrate effective, efficient and realistic regulation. Several of the implementation strategies are applicable to the review and the measure to complete two key process improvements that increase efficiency, effectiveness, and realism could be applied to the AP1000 pre-application review. As discussed in the Westinghouse meetings with the staff on the licensing process for the AP1000, the entire objective of a phased approach to the review is to increase the efficiency and effectiveness of both the applicant (Westinghouse) and the staff review process. Westinghouse and the NRC staff agreed upon the multi-stage review process for the express purpose of leveraging the value of the AP600 Design Certification effort and increasing the efficiency and reducing the required resources for the AP1000 review by efficiently retaining the appropriate portions of the AP600 DCD. In addition, the process of estimating the cost, schedule, and resource needs of the pre-review prior to initiating the review is believed to be a significant process improvement over the AP600 Design Certification process. This estimating and planning phase may potentially be adopted by the NRC staff as a process improvement applicable to other tasks.

Specifically, the following implementing strategies are all achieved with the AP1000 Pre-Application review:

1. *To use risk information to improve effectiveness and efficiency*

   The design and licensing of the AP600 extensively used risk information to improve both the design, and to improve regulatory efficiency. Westinghouse used risk as a design tool to select features of the plant to effectively minimize the risk associated with an AP600. Working together, the NRC and the AP600 stakeholders used risk information to improve the efficiency of the regulatory oversight associated with an AP600 with the Regulatory Treatment of Non-Safety Systems (RTNSS) process. The AP1000 builds on the efficiencies attained by the AP600 Design Certification, and an AP1000 Design Certification would employ the same risk measures as AP600. The phase 2 pre-application review process is essential for a successful AP1000 Design Certification.

2. *To make decisions based on technically sound and realistic information*

   The Phase 2 submittals will provide the NRC the technical information needed to determine whether the approach proposed by Westinghouse is feasible for certifying the AP1000 design. The phased approach is likely to identify and address key issues and concerns of the staff and the ACRS at the earliest opportunity, resulting in a more efficient process for the complete safety assessment of the AP1000.

3. *To anticipate challenges posed by the introduction of new technologies and changing regulatory demands and to take steps to ensure that the agency's regulatory process does not impede the use of new technology to improve safety, increase productivity, or reduce costs.*

   The design certification application for the AP1000 is in direct response to the economic deregulation of the electric power industry for which the agency has proposed to modify its regulatory processes in order to keep pace. Phase 2 is a modification of the standard design certification process and its timely completion is essential to keep pace with the economic deregulation of the industry. The Phase 2 effort provides the NRC with the ability to determine very definitively what challenges await the agency during an AP1000 design certification effort.

4. *The effectiveness of the NRC will also be enhanced by the continued utilization and honing of the staff skills necessary to conduct integrated plant reviews of advanced reactor designs.*

   Endeavors such as the AP1000 pre-application review provides challenges to the industry as well as the NRC to improve processes and contributes to the long-term viability of the industry.

PERFORMANCE GOAL #4: Reduce unnecessary regulatory burden on stakeholders.

Westinghouse Performance Goal Ranking: High

10 CFR Part 52 currently does not contemplate a pre-application review and an applicant would normally be required to submit an entire application to obtain an NRC acceptance review. The Phase 2 effort will enable Westinghouse to determine whether the design certification effort as proposed is technically feasible and will prevent the unnecessary expenditures and resource diversion involved with a certification application prior to the resolution of the key issues that would need to be addressed for Design Certification. This reduction in regulatory burden could potentially save Westinghouse millions of dollars and also provides for optimum utilization of NRC staff resources.

ADDITIONAL CONSIDERATIONS:

Chairman Meserve has expressed his intention that the NRC will be fully positioned to support future applications for advanced plants. Over the past decade, both the NRC and the industry invested significant effort and resources in attempting to fulfill this objective. However, Westinghouse believes that the currently certified ALWR designs will not be able to compete in a deregulated market unless the price of natural gas increases substantially or becomes unstable and/or the fossil fuel generators are assessed financial penalties for environmental considerations. Since those are uncontrollable and unpredictable factors, the AP1000 design certification application is necessary to position the nuclear option to compete in the deregulated market. Phase 2 is essential to achieving that objective for the nuclear industry and to achieving Chairman Meserve's stated intention for the NRC.

Enclosure 2
Description of Westinghouse Submittals for AP1000 Phase 2 Pre-Application Review

Applicability of AP600 Test Program to AP1000

In NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," the NRC states that the requirements of 10 CFR Part 52 have been interpreted to require that a passive plant vendor must develop and perform design certification test programs of a sufficient scope. This includes both separate-effects and integral-systems experiments to provide data to assess the computer codes used to analyze plant behavior over the range of normal operating conditions, transient conditions, and accident sequences.

Westinghouse will submit a report titled AP1000 Analysis Plan and Scaling Assessment of AP600 Test Program. Its format will be based on WCAP-14141, AP600 Test and Analysis Plan. The purpose of the report will be to provide the information necessary for the NRC staff to determine whether the AP600 test programs are sufficient to meet the requirements of 10 CFR Part 52 for an application for Design Certification of an AP1000. This report will include the following:

- Description of the AP1000 plant focusing on design changes to AP600 that are potentially important with respect to the performance of the passive safety systems
- An overview description of the AP600 test programs and their applicability to AP1000
- Discussion of important thermal-hydraulic phenomenon for modeling AP1000 performance
- Scaling assessment of the AP600 tests to the AP1000 plant
- Justification of the use of validated AP600 analysis codes for AP1000
- Results of AP1000 safety performance assessments using AP600 analysis codes
- Description of changes to AP600 analysis codes to be implemented as part of AP1000 design certification

This report will address the issues identified in the letter from the NRC as well as the issues identified in the letter from the ACRS. An outline of the proposed report is included. The resolution of this issue will be a determination by NRC of whether the AP600 test program meets the requirements of 10 CFR Part 52 for the AP1000.

Outline of AP1000 Analysis Plan and Scaling Assessment

1.0 Introduction
2.0 AP1000 Design Description
  2.1 Overall Plant Description
  2.2 Reactor Coolant System Design
    2.2.1 Reactor Design
    2.2.2 Steam Generator Design
    2.2.3 Reactor Coolant Pump Design
    2.2.4 Pressurizer and Loop Arrangement
  2.3 Passive Core Cooling System Design
    2.3.1 Passive Core Cooling System Design Margins Assessment
  2.4 Containment and Passive Containment Cooling System Design
    2.4.1 Containment and PCS Design Margins Assessment
3.0 Important Thermal-Hydraulic Phenomenon for Modeling the AP1000 System Performance
4.0 AP600 Test Program
5.0 Scaling Assessment of the AP600 Test Program
  5.1 Passive Core Cooling System
    5.1.1 Separate Effects Phenomenon
      5.1.1.1 Automatic Depressurization
      5.1.1.2 Core Makeup Tank
      5.1.1.3 Passive Residual Heat Removal
    5.1.2 Integral System Phenomenon
      5.1.2.1 LOCA Phenomenon
        5.1.2.1.1 Automatic Depressurization
        5.1.2.1.2 Transition to IRWST Injection
        5.1.2.1.3 Sump Injection
      5.1.2.2 Integral System Test Matrices
  5.2 Containment Performance Phenomenon
  5.3 Departure from Nucleate Boiling Tests
6.0 Safety Analysis Results Comparisons
  6.1 Transient Performance Assessment
  6.2 Loss of Coolant Accident Performance Assessment
    6.2.1 Small Break LOCA
      6.2.1.1 Two-Inch Cold Leg Break
      6.2.1.2 Direct Vessel Injection Line Break
      6.2.1.3 Long-Term Cooling
    6.2.2 Large Break LOCA
  6.3 Containment Performance Assessment
    6.3.1 Steam Line Break
    6.3.2 LBLOCA
7.0 Applicability of AP600 Safety Analysis Codes to AP1000
  7.1 WCOBRA/TRAC Code Validation for AP1000
    7.1.1 LBLOCA
    7.1.2 SBLOCA
    7.1.3 Long-Term Cooling
  7.2 NOTRUMP Validation for SBLOCA
  7.3 LOFTRAN_AP Code Validation
  7.4 WGOTHIC Containment Code
8.0 Conclusions
9.0 References

Applicability of AP600 Analysis Codes to AP1000

As part of the design certification application for the AP600, Westinghouse performed extensive code development and validation activities to develop analysis tools suitable for performing Chapter 15 accident analyses for the AP600. The NRC and the Advisory Committee on Reactor Safeguards (ACRS) have performed extensive reviews of the code development and validation programs for the computer codes developed for the AP600. It is recognized that certain limitations of the codes were identified in NUREG-1512. In these cases, the acceptability of the codes for the AP600 is based, in part, on the large safety margins provided by the AP600. Westinghouse will address the limitations identified in NUREG-1512 for the AP600 computer codes used for safety analysis and will demonstrate the appropriateness of their use for the AP1000.

Westinghouse will provide safety analysis assessments of the AP1000 using the AP600 analysis codes. These assessments will be included in the AP1000 Analysis Plan and Scaling Assessment. These assessments will not be a complete set of Chapter 15 accident analyses, but will be a representative sampling of analyses demonstrating the performance of the AP1000 safety systems.

Based on the conclusions of NUREG-1512 as well as the scaling assessments and analysis, Westinghouse will provide an assessment of the applicability of each code to the AP1000, and will identify any code changes necessary. This assessment will be provided as part of the AP1000 Analysis Plan and Scaling Assessment Report.

In addition, Westinghouse will provide an assessment of the AP1000 passive core cooling system design margins with respect to safety injection performance characteristics. The relative margin between the performance of the AP600 and the AP1000 passive core cooling system features will be assessed during the minimum core inventory time period at the start of IRWST injection following a small LOCA. This assessment will address the relative performance margins in the IRWST injection paths and the ADS stage 4 vent paths. The line resistances of these paths will be used together with consistent boundary conditions to provide a simple calculation of the comparative injection and venting flow rates. The purpose of this evaluation is to provide a simple estimate of the relative margin of the AP1000 as compared to the AP600. The assessment will present the important inputs, boundary conditions and calculated results and will discuss the meaning and significance of the results. This assessment is not meant to replace any of the Chapter 15 accident analyses that would be provided as part of the AP1000 Application for Design Certification. It is provided for informational purposes to assist the staff and ACRS to assess the margin of safety that will be provided by the AP1000 passive safety systems for the particular phases of the LOCA events that are most sensitive to the code limitations outlined in NUREG-1512.

The resolution of this issue will be an evaluation by the NRC of the acceptability of the AP600 analysis codes, including proposed changes, for performing accident analysis for the AP1000.

Design Acceptance Criteria

The AP1000 Design Certification application is expected to include less design detail than that provided in the AP600 Design Certification application. The General Arrangement, structural configuration, equipment and piping layout of the AP1000 are substantially the same as the AP600. However, qualification analyses will be deferred to the Combined License applicant. This affects the design detail available during Design Certification in the following areas:

Seismic analyses	(DCD Sections 2 and 3.7)
Structural design	(DCD Section 3.8)
Piping design	(DCD Section 3.6 and 3.9)

The objective in phase two for this issue is for Westinghouse and NRC to agree on the level of detail to be provided in an application for Design Certification for the AP1000. In phase two, Westinghouse will provide markups of the above listed sections of the AP600 DCD. These markups will show the level of information proposed for the AP1000 DCD. The AP1000 DCD will retain the methodology and design criteria for the COL applicant that references an AP1000 plant. Where the AP600 DCD contained results of analyses, the AP1000 DCD will identify information to be provided by the Combined License applicant. COL requirements and DAC will be proposed similar to those employed in the DCD for other certified standard plant designs (i.e. System 80+).

Analyses and evaluation will be provided in phase two for a hard rock site. The results of these analyses are intended in phase two to provide NRC with an understanding of the effect of the AP1000 configuration changes on the seismic results previously provided for the AP600. Additional review of these analyses should be deferred to the review of the AP1000 Design Certification Application.

In addition, Westinghouse will provide a draft Design Acceptance Criteria (DAC) for the AP1000 piping design, seismic design, and structural design. Resolution of this issue will be a determination by the NRC that the AP1000 can utilize Design Acceptance Criteria in the areas of piping design, seismic design, and structural design.

Exemptions

The purpose of this item is to identify which exemptions granted for the AP600 design certification can be retained for the AP1000 application. Westinghouse will identify the exemptions that will be requested for the AP1000 application and will provide justification for these exemptions in accordance with the requirements for 10 CFR 50.12.

Enclosure 3
Westinghouse Comments on Phase 1 Results
(Enclosure to NRC letter of July 27, 2000)

Item 1 – Scope of NRC Review

There are no comments on this item at this time.

Items 2 and 3 – Test Program and Analysis Plan

The test reviews are the most important reviews of the Phase 2 program. Westinghouse notes that the NRC staff estimates for the review of the applicability of the AP600 tests to the AP1000 assumes that a single staff member with no prior AP600 experience will need "to spend a significant amount of time reviewing the AP600 test program to prepare for the Phase 2 assessment." Westinghouse requests that the NRC make every effort to assign the reviewers who performed the test assessment for the AP600 Design Certification Review. Using the original reviewers would demonstrate application of NRC performance goal number three, "Make NRC activities and decisions more effective, efficient and realistic." The test and analysis review effort for the AP600 Design Certification involved a significant cross-section of the NRC technical staff and included input from the Office of Nuclear Regulatory Research. It is expected that experienced AP600 reviewers from the staff and research would require significantly less hours than would be required for a new reviewer unfamiliar with the AP600. We believe that a successful review will result if a group of NRC experts experienced in the Design Certification of the AP600 can form a consensus opinion on the test program issues.

Westinghouse has revised our deliverable for these tasks. The detailed outline of the AP1000 Analysis Plan and Scaling Assessment of AP600 Test Program shows that results of safety analysis assessments will be provided. In addition, the AP1000 Passive Core Cooling System Design Margins Assessment will be provided as section 2.3.1 of the report.

<u>Separate Effects Tests</u>

The outline of the report has been revised based on the NRC staff feedback to explicitly address the separate effects tests of the PRHR, ADS, and CMT. These components for AP1000 are the same (scale, geometry, and configuration) as used in the AP600. Therefore, it is not expected that formal scaling assessments are necessary to justify the use of these separate effects tests databases. The report will address the basis of the acceptability of these test programs for the AP600, and will discuss the basis that these test programs are acceptable for an AP1000 application.

<u>Integral Systems Tests</u>

The NRC letter proposes that the test matrices for the integral systems test be reviewed to demonstrate that the test matrices adequately cover the AP1000 design. Westinghouse will address this issue in section 5.1.2.2 of the report by performing a scaling assessment that compares the range of break sizes tested for the AP600 to an equivalent range of break sizes for the AP1000. The results should demonstrate that the size of breaks tested for AP600 provide a sufficiently large range of equivalent AP1000 break sizes.

Formal scaling assessments will be performed for the ADS phase, transition to IRWST injection phase, and sump injection phase.

Critical Heat Flux

This issue will be addressed in section 5.1.4 of the AP1000 Analysis Plan and Scaling Assessment Report. Westinghouse will demonstrate the adequacy of the DNB correlations that will be used for AP1000.

WCOBRA-TRAC

In discussions with the NRC staff reviewer for this item, there are three major issues associated with the use of WCOBRA-TRAC that need to be addressed for AP1000. These three issues are:

1. The use of WCOBRA-TRAC for Long-Term Cooling analyses was acceptable based on the validation of the code against four specific OSU tests. Westinghouse must demonstrate that the scaling of the OSU test facility is sufficient for the AP1000, and that the tests used to validate WCOBRA-TRAC for AP600 are sufficient for the purposes of code validation for AP1000 long term cooling. This issue will be addressed in section 5.1.2 of the AP1000 Analysis Plan and Scaling Assessment.
2. The analysis of Long-Term Cooling for AP600 was performed using WCOBRA-TRAC in a "windows" mode. This approach was acceptable for the AP600 design certification and Westinghouse must demonstrate that this approach is still valid for the AP1000. This issue will be addressed in section 7.1.3 of the report.
3. Westinghouse should consider the performance of the passive safety systems with respect to the issue of boron precipitation in the core during long-term core cooling post-LOCA. In the AP600, Westinghouse addressed this issue by performing calculations that demonstrated that boron precipitation in the core would not occur due to the moisture carryover of liquid out the 4^{th} stage ADS valves. Results of these calculations demonstrated significant margin such that the amount of liquid leaving the core via the 4^{th} stage ADS valves would have to be reduced by more than an order of magnitude before boron precipitation in the core could occur. Westinghouse agrees that this issue will need to be addressed and intends to address it as part of an application for Design Certification of an AP1000 (not in current review phase).

LOFTRAN/LOFTTR2

Westinghouse will address the issues raised regarding the use of LOFTRAN for AP1000 in section 7.3 of the report.

NOTRUMP

The issues identified in the letter and the subsequent phone call with the cognizant staff reviewer will be addressed in section 7.2 of the report. In addition, the issues raised by the ACRS in the letter on AP1000 will be addressed.

WGOTHIC

Westinghouse will address the issues regarding heat and mass transfer and water coverage characteristics raised in the opening paragraphs of the discussion on WGOTHIC in section 5.1.3 of the report. Scaling of these phenomenon with respect to the test facility will be formally assessed. The following refers to the numbered items in the NRC letter.

1. A PIRT evaluation will be provided as section 3 of the report.
2. Evaluations performed by Westinghouse and the NRC for the AP600 utilized a lumped parameter nodalization. Justification for the use of a lumped parameter nodalization was based on:
   - Froude number scaling
   - Mixing studies (boundary layers, buoyant plumes)
   - Scaled test facilities
   - Lumped parameter nodalization studies

   In addition, AP600 containment performance for LBLOCA exhibited a large margin with respect to design limits for analysis of accidents using realistic assumptions. Westinghouse will perform Froude number scaling for the AP1000 in section 5.2 of the report. Westinghouse will provide its rationale as to the applicability of the AP600 mixing studies, scaled test facilities, and lumped parameter nodalization studies to the AP1000 in the Analysis Plan and Scaling Assessment report. In addition, the AP1000 containment design margin for LBLOCA will be demonstrated such that the lumped parameter nodalization approach could be found to be acceptable by the NRC.
3. The limitations and restrictions for WGOTHIC will be addressed in section 7.4 of the report.

Item 4 – AP1000 Probabilistic Risk Assessment

As discussed in the cover letter, Westinghouse will defer this activity to Design Certification. The issues raised in the NRC letter would be addressed as part of the review of the basis for the AP1000 PRA acceptance criteria.

Item 5 – Defer Selected Design Activities

The following clarifies our position with regards to the seven issues raised in the NRC letter. The issues identified in the NRC letter are issues that will be addressed either as part of the AP1000 Design Certification (Phase 3), or by the COL applicant. The impact of these issues on the phase 2 scope of work is an assessment by the staff of the acceptability of the draft Design Acceptance Criteria (DAC).

1. In Phase 3, Westinghouse will demonstrate dynamic stability for the rock sites. A COL applicant information item will be added requiring dynamic stability to be confirmed by each COL applicant for sites that are not covered by the rock site demonstration.

2. In Phase 3, Westinghouse will provide an assessment of the foundation mat acceptability. Westinghouse will define interface loads and acceptance criteria in the Design Acceptance Criteria (DAC) (a draft of the structural DAC will be provided in Phase 2). The COL Applicant will perform a basemat design to complete the structural DAC.

3. In Phase 3, Westinghouse will provide assessments of critical regions of the structural design. This will include the modular walls of the in-containment refueling water storage tank. The structural design of these critical regions will be performed by the COL applicant in completing the structural DAC.

4. In Phase 3, Westinghouse will provide justification for use of any newer editions of the design codes (e.g. ASME 1999 Addenda for the containment vessel).

5. In Phase 3, Westinghouse will perform an assessment of the subcompartment pressurization analyses. The piping and structural designs will be performed by the COL applicant in accordance with the piping and structural DACs. These DACs will be provided in phase 2 and will address subcompartment pressurization loads. Westinghouse does not expect significant changes to the thermal and pressure loads because there was significant margin in these analyses for AP600.

6. In Phase 3, Westinghouse will describe the containment vessel design and its performance under severe accident conditions.

7. Westinghouse will incorporate these comments in the draft DCD sections.

Exemptions

Westinghouse has no additional comments on this issue.

Project Management for Phase 2

Westinghouse requests that project management of Phase 2 enable us to monitor costs on a monthly basis. We would envision monthly reports that would identify manpower resources expended on the project. In addition, we request that the costs be tracked separately as follows:

- Tests and Analysis Review
- Design Acceptance Criteria
- Exemptions / Project Management

The purpose of this request is to help us better manage our total costs on the project.

Enclosure 4

Westinghouse Categorization of ACRS Issues Related to the Review of the AP1000 Design

The following table summarizes our categorization of the items raised in the ACRS letter as either Phase 2 issues, Design Certification issues, or issues to be addressed by the COL applicant.

Item	Categorization	Comments
1. Application for Design Certification		
a	Phase 2 / DC	There are two parts to this item. The scope of the SSAR Chapter 15 accident analyses will be addressed at Design Certification. The issue of whether the AP600 analysis codes need to be revalidated will be addressed in Phase 2.
b	Phase 2	Section 7.2 of the AP1000 Analysis Plan and Scaling Assessment
c	Phase 2	Section 5 of the AP1000 Analysis Plan and Scaling Assessment
d	DC	Based on our current plan, this issue would be addressed during Design Certification.
e	DC	Evaluation of core performance will be part of Design Certification, although some aspects of core performance (DNB correlation) will be addressed as part of Phase 2.
f	Phase 2	An evaluation of the impact of performance ratings (i.e. design pressures and temperatures, system design capacities, etc.) will be addressed in Phase 2.
g	DC	Detailed seismic analysis of the containment is addressed as part of Design Certification.
2. Probabilistic Risk Assessment (will be submitted for Design Certification)		
a	DC	This will be addressed in the level 2 PRA
b	DC	This will be addressed in the level 2 PRA
c	DC	This will be addressed in the level 2 PRA
d	DC	This will be addressed in the level 1 PRA
e	DC	This will be addressed in the level 2 PRA
f	DC	This will be mainly addressed in the level 2 PRA. Aspects of containment mixing will be addressed in Phase 2.

HIGH-LEVEL GUIDELINES
FOR
PERFORMANCE-BASED ACTIVITIES

PRESENTATION TO ACRS FULL COMMITTEE

AUGUST 30, 2000

OFFICE OF NUCLEAR REGULATORY RESEARCH

N. Prasad Kadambi, RES/REAHFB

Robert Youngblood, ISL, Inc.

Christopher Smith, ISL, Inc.

OUTLINE

- OVERVIEW
- CASE STUDIES INVOLVING APPLICATION OF THE GUIDELINES
  - CASE STUDY OF HYPOTHETICAL REGULATORY FRAMEWORK FOR "CONTROL OF COMBUSTIBLE GASES"
  - CASE STUDY ON SUBPART H TO 10 CFR PART 20, "RESPIRATORY PROTECTION AND CONTROLS TO RESTRICT INTERNAL EXPOSURE IN RESTRICTED AREAS"
- INTERRELATIONSHIP AMONG REGULATORY INITIATIVES
- STAFF'S PLANS
- CONCLUSION

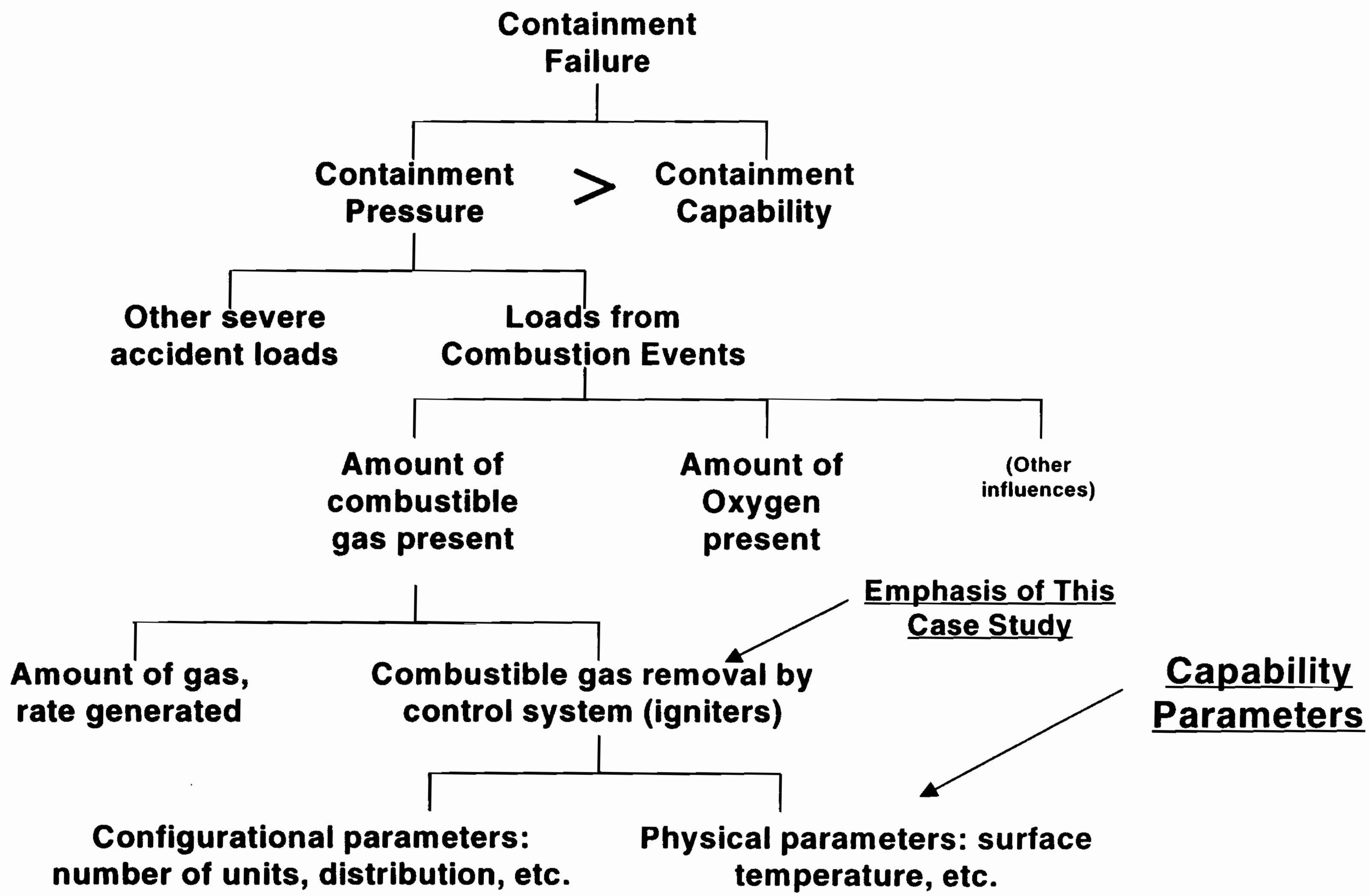
OVERVIEW

- THE STAFF BRIEFED ACRS ON JUNE 8, 2000 AND ACNW ON JULY 25, 2000.
- COMMISSION PAPER PROVIDES FINAL GUIDELINES, RESPONSES TO PUBLIC COMMENTS AND CASE STUDIES.
- THE CASE STUDIES ARE DESIGNED TO TEST WHETHER THE GUIDELINES ARE USEFUL.
- STAFF CONCLUDES THAT GUIDELINES ARE READY FOR AGENCY-WIDE APPLICATION
- MEANWHILE STAFF HAS CONTINUED SUBSTANTIAL EFFORTS ON RISK-INFORMED REGULATIONS, E.G.,
    - UPDATE TO THE RISK-INFORMED REGULATION IMPLEMENTATION PLAN.
    - PUBLIC WORKSHOP ON RISK-INFORMED APPROACHES TO NUCLEAR MATERIALS REGULATORY APPLICATIONS.
- THE STAFF IS DEVELOPING THE BASIS FOR INTEGRATING THE ACTIVITIES PURSUANT TO REGULATORY INITIATIVES.

Case Study 1
Combustible Gas Control

- This case study applies the viability guidelines to a hypothetical regulatory framework on combustible gas control in certain containment types
- Risk information can be used to establish what the requirement needs to accomplish:
    - Safety mission (what is important)
    - What the reliability / availability needs to be
    - Conditioning on the characteristics of the functional challenge (support availability, phenomenology)
- In this case study, risk information has established the following:
    - Uncontrolled combustion of gases evolved in accidents can lead to containment failure and large radiological release
    - Potentially important sequences involve
        - Station blackout, affecting availability of power sources
        - Core melt phenomenology, affecting operability of systems in containment
        - Severe accident loads from phenomena other than combustion, influencing the impact of loads from combustion
- High-level statement of requirement: Prevent containment failure from uncontrolled combustion of gases in risk-significant scenarios.
- Begin application of guidelines by searching for monitorable parameters
    - Capability parameters (flowrates, heat removal rates, ...)
    - Reliability / Availability parameters

Establishing Capability Parameters



Guideline IA: Measurable (or calculable) parameters to monitor acceptable plant and licensee performance exist or can be developed

- Capability (of igniters)
  - Surface temperature
  - Distribution and number of units
    - Not related to ongoing performance; fixed property of design
  - Environmental qualification parameters
    - Not amenable to performance monitoring
- Reliability / Availability:
  - Functional reliability
  - Division reliability
  - Division availability
  - Unit reliability
  - Unit availability

- Note: support systems need to be considered

Guideline IB: Objective criteria to assess performance exist or can be developed

- Capability (of igniters)
  - Surface temperature, distribution and number of units
    - Parameters are established through model evaluations
  - Environmental qualification
    - Criteria can be developed from phenomenology
- Reliability / Availability:
  - Functional reliability is determined in light of functional challenge frequency and (e.g.) LERF guidelines
  - Given the functional reliability, and the design configuration, criteria can be established for division and unit level reliability & availability parameters

Guideline IC: Licensee flexibility in meeting the established performance criteria exists or can be developed

- Capability (of igniters)
  - Within a given technology, some limitations on flexibility would be implicit (Needed surface temperature determined by phenomenology, etc.)
  - Choice of technology could be allowed
- **Reliability / Availability:**
  - Flexibility exists in that there are different ways to achieve needed functional reliability
    - More redundancy in design means more (igniter) unit outages can be tolerated, different levels of unit reliability can be tolerated
    - Specifying availability averaged over a specified time period is in some ways more flexible than specifying an allowed outage time

Guideline ID: A framework exists or can be developed such that performance criteria, if not met, will not result in an immediate safety concern

- Capability parameters: For typical testing frequencies, degradation in monitored aspects of capability would be detectable within a short time
- Reliability / Availability Parameters: The reliability and availability needed in this function at most plants could be confirmed by monitoring (testing)
- The risk accepted when performance criteria are not met depends on
  - the length of time over which they are not met,
  - the likelihood of a functional challenge, and
  - the consequences of functional failure
- For this function, the combination of analysis, frequency of challenges to this function, and the LERF guidelines would be used to support acceptable time scales for detecting and addressing performance issues

Case Study 1
Summary

- Capability parameters
  - Aspects of capability such as environmental qualification are not amenable to performance-based treatment
    - Parameters and criteria exist, but it is not practical to confirm performance
  - Some capability parameters satisfy guidelines other than flexibility.
    - To achieve licensee flexibility, choice of technology needs to be allowed
- Reliability / Availability parameters satisfy all four viability guidelines
- This regulatory framework could be performance-based to a significant degree
- The guidelines were useful in evaluating the viability of a performance-based approach in this regulatory framework

CASE STUDY 2

- THIS CASE STUDY IS FUNDAMENTALLY DIFFERENT FROM THE FIRST CASE STUDY - THE PURPOSE IS ASSESSMENT RATHER THAN IDENTIFICATION
- PERFORMANCE-BASED GUIDELINES ARE APPLIED TO A RECENTLY REVISED RULE
- ASSESSMENT FOR THIS CASE STUDY IS LIMITED TO THE RULE LEVEL OF THE REGULATORY FRAMEWORK

CASE STUDY 2

- FOCUS APPLICATION OF THE GUIDELINES ON THE RECENT **CHANGES** MADE TO THE RESPIRATORY PROTECTION REQUIREMENTS (SUBPART H OF 10 CFR 20)
- VIABILITY GUIDELINES WERE THOROUGHLY APPLIED TO THREE (3) SPECIFIC CHANGES TO THE SUBPART H REQUIREMENTS
- THE REMAINING GUIDELINES WERE APPLIED TO ALL THE CHANGES TO THE SUBPART H REQUIREMENTS
- DO THE GUIDELINES SUPPORT THE CHANGES MADE TO THE REQUIREMENTS?

CASE STUDY 2

APPLICATION OF THE GUIDELINES IS ONLY MADE AT THE RULE LEVEL

RULE CHANGE	RULE FUNCTIONALITY	GUIDELINE APPLICATION
REQUIREMENT TO INCLUDE NON-RADIOLOGICAL SAFETY FACTORS IN ALARA ANALYSES INCREASES LICENSEE FLEXIBILITY	MINIMIZE WORKER RISK DUE TO AIRBORNE HAZARDS	VIABLE FOR PERFORMANCE-BASED APPROACH INCREASE IN FLEXIBILITY MAKES THE REVISION MORE AMENABLE TO A PERFORMANCE-BASING
REQUIREMENT TO MEET QUANTITATIVE FIT TEST CRITERIA AND TESTING FREQUENCY ADDS PRESCRIPTIVE REQUIREMENTS TO THE RULE	ENSURE PROPER EQUIPMENT FUNCTION	LIMITED VIABILITY FOR PERFORMANCE-BASED APPROACH POTENTIAL FOR AN IMMEDIATE SAFETY CONCERN IF PROPER FIT FAILS DURING USE PRESCRIPTIVE REQUIREMENTS NECESSARY TO ENSURE ACCURATE DOSE CALCULATIONS
REVISED EXPLICIT CONSIDERATIONS FOR RESPIRATORY EQUIPMENT SELECTION NEUTRAL IMPACT ON LICENSEE BURDEN	ENSURE SELECTION OF PROPER EQUIPMENT	LIMITED VIABILITY FOR PERFORMANCE-BASED APPROACH POTENTIAL FOR AN IMMEDIATE SAFETY CONCERN IF WRONG EQUIPMENT SELECTED

CASE STUDY 2

- THE REMAINING GUIDELINES WERE APPLIED TO THE CHANGES TO THE SUBPART H REQUIREMENTS AND SUPPORT THE CHANGES MADE TO THE REQUIREMENTS

- CONCLUSION: THE RESULTS OF APPLYING THE PERFORMANCE-BASED GUIDELINES WERE CONSISTENT WITH THE CHANGES MADE TO THE SUBPART H REQUIREMENTS

- THIS CASE STUDY DEMONSTRATED THAT PRESCRIPTIVE REQUIREMENTS ARE SOMETIMES NECESSARY TO ENSURE THE ACCURACY OF PERFORMANCE INFORMATION

INTERRELATIONSHIPS AMONG REGULATORY INITIATIVES

- REGULATORY INITIATIVES ARISE FROM COMMISSION DIRECTION, OPERATING EXPERIENCE, STAKEHOLDER INPUT, STAFF INITIATIVES

- SCREENING PROCESS DETERMINES WHETHER TO PURSUE INITIATIVE, AND IF SO, WITH WHAT PRIORITY

- ELEMENTS OF THE REGULATORY FRAMEWORK CONSIDERED FOR CHANGE AS PART OF THE INITIATIVE ARE IDENTIFIED

- REGULATORY APPROACH IS SELECTED - (1) RISK-INFORMED AND PERFORMANCE-BASED, (2) RISK-INFORMED, (3) PERFORMANCE-BASED, AND (4) TRADITIONAL

    - THIS SELECTION RELIES ON GUIDELINES DEVELOPED AS PART OF THIS PERFORMANCE-BASED INITIATIVE AND THE RISK-INFORMED INITIATIVE

    - THE REGULATORY APPROACH MAY DIFFER FROM ONE LEVEL OF THE REGULATORY FRAMEWORK TO ANOTHER

    - BLEND OF APPROACHES WILL BE APPROPRIATE IN MANY AREAS

- PROCEDURES APPLICABLE TO ISSUANCE OF REGULATORY PRODUCTS (BACKFIT ANALYSIS, REGULATORY ANALYSIS, RULEMAKING PROCESS) REMAIN UNCHANGED

STAFF'S PLAN

- STAFF WILL APPLY GUIDELINES IN ONGOING AND FUTURE CHANGES TO THE REGULATORY FRAMEWORK AS APPROPRIATE:
  - GUIDELINES WILL BE APPLIED TO OPTION 3 EFFORTS UNDER RISK-INFORMED INITIATIVE
  - GUIDELINES WILL BE APPLIED TO SUITABLE CANDIDATE IDENTIFIED AS BEING NOT APPROPRIATE TO BE RISK-INFORMED
  - MANAGEMENT DIRECTIVES WILL BE DEVELOPED TO SUPPORT AGENCY-WIDE IMPLEMENTATION OF GUIDELINES
  - COMMUNICATION PLANS WILL BE DEVELOPED TO ENCOURAGE STAKEHOLDER IMPLEMENTATION OF GUIDELINES
- STAFF WILL PROVIDE A REPORT TO THE COMMISSION AT THE END OF FY-2001.
- THE PBPM PROCESS WILL TAKE INTO ACCOUNT THE RESOURCE IMPACTS OF APPLYING THE GUIDELINES FOR EACH CHANGE TO THE REGULATORY FRAMEWORK.

CONCLUSIONS

- STAFF HAS RESPONDED TO THE ELEMENTS OF THE SRM AND HAS GONE BEYOND IT TO DEMONSTRATE USEFULNESS OF THE GUIDELINES.
- INTERNAL AND EXTERNAL STAKEHOLDER INPUTS HAVE BEEN CONSIDERED IN THE FINAL GUIDELINES.
- ADVISORY COMMITTEES CAN OBSERVE APPLICATION OF THE HIGH-LEVEL GUIDELINES FOR PERFORMANCE-BASED ACTIVITIES THROUGH THEIR OVERSIGHT OF THE REGULATORY PROCESS.
- IMPROVEMENTS TO GUIDELINES WILL BE CONSIDERED AS EXPERIENCE DICTATES.
- IMPROVED INTEGRATION AMONG REGULATORY INITIATIVES WILL BE ACCOMPLISHED AS MORE EXPERIENCE IS GAINED.

Wed 8/30
@ 8:35

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
475TH	9	9.1

Title: High-Level Guidelines for Performance-based Activities [Predecisional]

Authors: NRC Staff

List of Documents Attached

Predecisional Draft Received by the ACRS on August 25, 2000, from RES to the Executive Director of Operations, SUBJECT: High-Level Guidelines for Performance-Based Activities

9

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

8- 25-00

PROVIDED FOR INTERNAL ACRS USE ONLY

PREDECISIONAL

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: HIGH-LEVEL GUIDELINES FOR PERFORMANCE-BASED ACTIVITIES

PURPOSE:

This paper is to inform the Commission of the development of the high-level guidelines consistent with the direction in the Staff Requirements Memorandum (SRM) to SECY-99-176, "Plans for Pursuing Performance-Based Initiatives." The guidelines, their relationship to the risk-informed program, and the results of test applications of the guidelines are provided. These guidelines can be applied to regulatory activities to identify and assess the use of performance-based regulatory approaches instead of prescriptive criteria to assure safe performance, and as such, should help to increase reliance on performance-based regulatory approaches throughout the agency.

SUMMARY:

The staff has developed and tested high-level guidelines (Attachment 1) to identify and assess the viability of making components of the regulatory framework performance-based. The guidelines are intended to promote the use of a performance-based regulatory framework throughout the agency. In general, a performance-based regulatory approach focuses on results as the primary basis for regulatory decision-making and as such allows licensee flexibility in meeting a regulatory requirement. This in turn, can result in a more efficient and effective regulatory process.

Internal and external stakeholders have commented on the guidelines and their comments have been addressed in the development of the guidelines. Specifically, the staff has addressed concerns among some stakeholders that a performance-based regulatory framework would focus only on reductions in regulatory burden and that public health and safety would lose

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emphasis. The staff notes that a performance-based approach is intended to focus the regulatory framework on desired outcomes and would be applied in conjunction with the agency's defense-in-depth principles as articulated in the Commission's White Paper, "Risk-Informed and Performance-Based Regulation," SRM to SECY-98-144 (White Paper).

Based on feasibility testing of the guidelines, the staff concludes that they can be used to effectively focus the regulatory framework to be more performance-based by:

(A) Identifying the components of the regulatory framework which can be made more performance-based. Note, the regulatory framework consists of the regulation and its supporting regulatory guides, standard review plans, technical specifications, NUREGs, and inspection guidance.

(B) Selecting or formulating performance parameters and associated performance criteria appropriate to the regulatory issue being addressed. For example, they facilitate identifying the level (i.e., component, train, system) at which performance criteria should be set.

Having established the feasibility of the guidelines, the staff plans to develop implementing guidance to incorporate the guidelines into internal NRC procedures, and to apply the guidelines to future regulatory initiatives, including those that are identified through risk-informed activities.

BACKGROUND:

In the SRM to SECY-99-176, issued on September 13, 1999, the Commission directed the staff to develop high-level guidelines to identify and assess the viability of candidate performance-based activities. The staff published a set of proposed guidelines in the Federal Register on January 24, 2000. The Commission was provided with a copy of the guidelines for information prior to the Federal Register publication.

In the SRM to SECY-99-176 the Commission directed that:

(A) The guidelines should be developed with input from stakeholders and the program offices.

(B) The guidelines should include discussion on how risk information might assist in the development of performance-based initiatives.

(C) The guidelines should be provided to the Commission for information.

(D) The staff should periodically update the Commission on its plans and progress in identifying and developing performance-based initiatives.

DISCUSSION:

The staff has used definitions from the White Paper for terminology such as "deterministic analyses," "risk insights," and "performance-based approach" in developing the guidelines.

Consistent with the NRC's Strategic Plan and the White Paper, the guidelines are to be applied across the full spectrum of materials, processes, and facilities regulated by the NRC.

Program Office and Stakeholder Input

In response to the SRM, the staff solicited the following program office and stakeholder input:

The staff established a Performance-Based Regulation Working Group (PBRWG) to ensure broad NRC program office participation in the development of the guidelines. The PBRWG has representation from RES, NRR, NMSS, and Region III. The PBRWG was instrumental in developing consensus among the offices on this initiative. Once these guidelines are incorporated into internal NRC procedures, the PBRWG will cease to exist and line management will assume responsibility for applying the guidelines.

A facilitated workshop was held on March 1, 2000 with a number of internal and external stakeholders representing the reactor, materials, and waste areas. This workshop solicited comments on an initial draft of the proposed guidelines and on a set of specific questions which were posed in two Federal Register Notices. Revised guidelines were published on May 9, 2000, and an on-line workshop was held on June 8, 2000. Comments were received at the workshops and in response to the Federal Register Notices, and the guidelines contained herein have been modified in response to public comments. The majority of the comments were supportive of the guidelines and staff efforts to make NRC regulatory requirements more performance-based. The staff's response to all comments appears in Attachment 2.

In addition, the staff briefed the Advisory Committee on Reactor Safeguards (ACRS) and the Advisory Committee on Nuclear Waste (ACNW). The Advisory Committee on Medical Uses of Isotopes (ACMUI) was provided briefing material.

Interrelationships Among Regulatory Initiatives

Initiatives to change the regulatory framework arise from various sources such as Commission direction, operating experience, stakeholder suggestions and staff initiatives. These proposed initiatives are normally subjected to a screening process that include identification of the specific modification of the regulatory framework and an initial prioritization utilizing the NRC's performance goals to determine whether the proposed initiative should be pursued and with what priority. A determination will then be made as to whether to pursue a "Risk-Informed and Performance-Based," "Risk-Informed," "Performance-Based," or "Traditional" approach based on guidelines described in this paper and in the Risk-Informed Regulation Implementation Plan (RIRIP). The staff would use the guidelines to assess the viability (discussed below) to make this determination. When feasible, it is preferable to use a risk-informed and performance-based approach. The staff is coordinating the guidelines in both areas to assure that no inconsistencies exist between them. A separate paper on RIRIP will be presented to the Commission. Once a decision is made to pursue a performance-based approach, the staff will apply the guidelines to assess the change (as described below) to further develop the approach. If the staff finds that a performance-based approach is not feasible, then the staff will assess what other methods can be used.

Overview of Guidelines

The guidelines are structured under three main groupings:

(i) Guidelines to Assess Viability: These guidelines rely on the four attributes of a performance-based approach as discussed in the White Paper. These are: measurable or calculable parameters; objective performance criteria; flexibility; and a performance failure not resulting in an immediate safety concern. These guidelines assess whether a more performance-based approach is feasible for any given new regulatory initiative. This assessment would be applied on a case-by-case basis and would be based on an integrated consideration of the individual guidelines within this grouping. In applying the guidelines, the staff must be cognizant of circumstances when implementation of a performance-based approach, in a manner inconsistent with the intent or objective, may have a negative or unacceptable effect on safety. For example, postponing needed maintenance in order to meet an availability goal would not be an acceptable way to use flexibility. However, it would be appropriate to revise the availability goal, reflecting considerations of safety significance, and expand flexibility if a sound technical basis is demonstrated.

(ii) Guidelines to Assess Change: If a performance-based approach is deemed viable based on the guidelines in (i) above, then the regulatory activity would be evaluated against guidelines that assess whether a more performance-based approach results in opportunities for regulatory improvement (by which is meant a positive contribution to the NRC's performance goals and achieving a net societal benefit). The performance goals are: maintain safety; increase public confidence; increase effectiveness, efficiency and realism; and reduce unnecessary regulatory burden. Additional guidelines in this group include a net benefit test, the ability of the proposal to be incorporated in the regulatory framework, and the ability to accommodate new technology. This evaluation is to be based on an integrated assessment of the individual guidelines within this grouping.

(iii) Guidelines to Assure Consistency with Other Regulatory Principles: These guidelines assess consistency and coherence with overriding NRC goals and principles (e.g., the defense-in-depth principle). It only needs to be applied if the candidate activity passes the first two sets of guidelines.

Use of Risk Information Relative to Performance-Based Initiatives

Consistent with the definition of a "risk-informed, performance-based approach" provided in the White Paper, risk information will be used to assist in the development of performance-based initiatives so that the staff will accomplish the following:

- Focus attention on the most important activities;
- Establish objective criteria for evaluating performance;
- Develop measurable or calculable parameters for monitoring system and licensee performance;

- Provide flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes; and

- Focus on the results as the primary basis for regulatory decision-making.

The staff has identified risk information to be relevant with respect to performance-based initiatives in three ways:

(1) A Basis for Establishing Appropriate Level of Performance:

A performance-based approach will assist in ensuring that important systems, functions, and other elements of regulated activity provide the requisite level of performance. In effect, the high level performance-based guidelines, and specifically the viability guidelines, provide a framework to search for the appropriate performance parameter and the level of performance necessary to achieve the safety objective. For example, for a given activity, the guidelines can help determine if performance goals should be set at the component, system or function level.

(2) To Provide Metrics, Thresholds and/or Regulatory Response:

The staff is using risk considerations to select performance metrics in several contexts. The reactor oversight program uses performance indicators which rely on risk information such as reliability and availability of certain systems, trains and components. The risk significance of performance changes can be evaluated directly where performance indicators are based on risk information. Performance thresholds and appropriate regulatory responses could then be determined in a straightforward manner. The guidelines are useful to characterize the appropriate performance attributes that might be monitored using risk insights. For example, risk information can be used to set reliability and availability goals for critical safety equipment.

(3) Unavailability of Quantitative Risk Evaluation Models:

On February 11, 1999, the Commission issued the SRM to SECY-98-132 in which the staff was directed to pursue performance-based initiatives that are not amenable to probabilistic risk assessment. Although many regulated activities may not be easily related to a quantitative risk model, they should not be precluded from being made more performance-based. Therefore, the staff is planning to apply the guidelines to suitable candidates in this category. In these instances, risk information of a less quantitative or non-quantitative nature, such as that available from an integrated safety assessment, should be relied upon. In some or all of these areas, a performance-based approach may present opportunities for regulatory improvements.

Testing of the High-Level Guidelines

Application of the guidelines requires that the nature of the regulated activity and the safety issues be defined with specificity. To explore how such challenges can be met in practice, the staff selected two issues to test the guidelines. For each issue, an NRC panel was formed consisting of experts on the specific regulatory issue. The first issue is related to the ongoing effort to risk-inform 10 CFR 50.44 (Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors). Although the hypothetical regulatory change is thought to be plausible, it must be considered purely illustrative at this time while the alternatives that will be proposed for revisions to 10 CFR 50.44 are still under consideration. The second issue involves a recent change that was made to Subpart H (Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas) of 10 CFR Part 20. In this case, the guidelines were applied retrospectively for illustrative purposes. The results of tests clearly support the utility of the high-level guidelines. A detailed description of these tests and the results appears in Attachment 3.

On the basis of the two test cases, the staff identified two issues concerning generic application of the guidelines. First, for a given regulatory activity, it appears that, in order to maximize the performance-based potential, one must apply the guidelines to the entire regulatory framework as it relates to that activity. This is because there typically exists a hierarchy of information pertaining to a regulated activity which encompass the more general provisions of the rule language to the relatively detailed supporting documents. Thus, opportunities to make an activity more performance-based could occur anywhere along the hierarchy. Further, an assessment that fails to apply the guidelines to the full regulatory framework could result in partial or ineffectual results, where, for example, a rule is made more performance-based but remains supported by unnecessarily prescriptive regulatory guidance.

Second, in most instances, performance will not be dependent on a single parameter. Rather, the guidelines will have to be applied to a combination of performance parameters each of which contributes to attaining the performance goals. For example, the first case study in Attachment 3 uses the combination of capability, reliability, and availability to provide the basis for setting performance criteria.

PLANS FOR PERFORMANCE-BASED INITIATIVE:

The staff plans to:

- Apply the guidelines in ongoing or future approved rulemakings, as appropriate.
- Apply the guidelines to ongoing regulatory efforts under Option 3 of SECY-98-300, "Options for Risk-Informed Revisions to 10 CFR Part 50."
- Apply the guidelines to suitable candidates identified as being not appropriate to be risk informed pursuant to the "Risk-Informed Regulation Implementation Plan" (SECY-00-0062, March 15, 2000).
- Develop a management directive to support agency-wide implementation of the guidelines in ongoing or future approved rulemakings and other regulatory activities as appropriate (e.g., the inspection process). Supporting guidance at the office level will occur through office letters;
- Develop a communications plan to promote broader awareness of performance-based approaches on the part of external stakeholders. Wider acceptance of the guidelines should lead to efficiencies and an overall increased level of performance-based activities.
- Provide a report to the Commission on the above activities at the end of FY-2001.

RESOURCES:

For FY 2001, RES currently has 1 FTE to: (1) apply the guidelines to a candidate regulation identified as not appropriate to be risk-informed; (2) develop a management directive; and (3) develop a communication plan. Resources requirements for developing specific performance-based changes to the regulatory framework as a result of implementing the high-level guidelines will be addressed as appropriate by the performing office(s). Future requirements will be addressed through the Planning, Budgeting, and Performance Management process.

COORDINATION:

The Office of the General Counsel has no legal objection to this paper. The Office of the Chief Financial Officer has reviewed this Commission Paper for resource implications and has no objection. The Office of the Chief Information Officer has reviewed this Commission Paper for information technology and information management implications and concurs in it.

William D. Travers
Executive Director
for Operations

Attachments: 1. High-Level Guidelines for Performance-Based Activities
2. NRC Response to Public Comments
3. Process and Case Studies Applying High-Level Guidelines

High-Level Guidelines for Performance-Based Activities

The proposed guidelines to identify and assess performance-based activities are shown below. They are substantially the same as those published in the Federal Register on May 9, 2000, with modifications based on internal and external stakeholder input. These guidelines are based on the four attributes in the Commission's White Paper, "Risk-Informed and Performance-Based Regulation," SRM to SECY-98-144. The nature of the regulated activity and the safety issues for which regulatory requirements are to be developed need to be defined with specificity before the guidelines are applied. Generally, an integrated assessment from a set of guidelines will provide the basis for any conclusion.

I. Guidelines to Assess Viability

The staff will apply the following guidelines to assess whether a more performance-based approach is viable for any given new regulatory initiative. This assessment would be applied on a case-by-case basis and would be based on an integrated consideration of the individual guidelines. Risk information provides the basis for identifying systems, functions or other elements of regulated activity which should be targeted for application of these guidelines so that the appropriate performance parameters are chosen and the level of performance is set to achieve the safety objective. The assessment for viability will ensure that sufficient information (data) and analytical methods exist or can be developed. The guidelines are listed below:

A. Measurable (or calculable) parameters to monitor acceptable plant and licensee performance exist or can be developed.

   (1) Directly measured parameter related to safety objective will typically satisfy this guideline.

   (2) A calculated parameter may also be acceptable if there is a clear relationship to the safety objective.

   (3) Parameters which licensees can readily access, or are currently accessing, in real time will typically satisfy this guideline. Parameters monitored periodically to address postulated or design basis conditions may also be acceptable.

   (4) Acceptable parameters should be consistent with defense-in-depth and uncertainty considerations.

B. Objective criteria to assess performance exist or can be developed.

   (1) Objective criteria consistent with the desired outcome are established based on risk insights, deterministic analyses and/or performance history.

C. Licensee flexibility in meeting the established performance criteria exists or can be developed.

   (1) Programs and processes used to achieve the established performance criteria would be at the licensee's discretion.

(2) A consideration in incorporating flexibility to meet established performance criteria will be to encourage and reward improved outcomes provided inappropriate incentives can be avoided.

D. A framework exists or can be developed such that performance criteria, if not met, will not result in an immediate safety concern.

(1) An adequate safety margin exists.

(2) Time is available for taking corrective action to avoid the safety concern.

(3) The licensee is capable of detecting and correcting performance degradation.

II. Guidelines to Assess Performance-Based Regulatory Change

If a more performance-based approach is deemed to be viable based on the guidelines in I. Guidelines to Assess Viability above, then the consequences of adopting a more performance-based approach would be evaluated based on an integrated consideration of this second group of guidelines. This assessment would compare the start up and implementation costs of the regulatory change relative to the NRC's performance goals and other desirable outcomes. The outcomes would be considered applicable to the public, the applicant or licensee, and the NRC staff. The guidelines are listed below:

A. Maintain safety, protect the environment and the common defense and security.

(1) Safety considerations play a primary role in assessing any change arising from the use of performance-based approaches.

(2) Adequate safety margins are maintained using realistic safety analyses, including explicit consideration of uncertainties.

B. Increase public confidence.

(1) An emphasis on results and objective criteria (characteristics of a performance-based approach) can help NRC to be viewed as an independent, open, efficient, clear, and reliable regulator.

(2) A performance-based approach helps with providing the public clear and accurate information about, and a meaningful role in the regulatory programs.

(3) A performance-based approach helps explain NRC's roles and responsibilities and how public concerns are considered.

C. Increase effectiveness, efficiency and realism of the NRC activities and decision-making.

(1) An assessment would be made of the level of conservatism existing in the currently applicable regulatory requirements considering analysis methodology and the applicable

assumptions. Any proposal to use realistic analysis would take into account uncertainty factors and defense-in-depth relative to the scenario under consideration.

(2) An assessment would be made of the performance criteria and the level in the performance hierarchy where they have been set. In general, performance criteria should be set at a level commensurate with the function being performed. In most cases, performance criteria would be expected to be set at the system level or higher.

D. Reduce unnecessary regulatory burden.

(1) A performance-based approach enables NRC to impose regulatory burden which is commensurate with the safety benefit, and which effectively focuses resources on safety issues.

(2) A performance-based approach will enable the costs associated with NRC activities to States, the public, applicants and licensees to be focused on areas of highest safety priority and avoid burden imposed by overly prescriptive regulatory requirements.

E. The expected result of using a performance-based approach shows an overall net benefit.

(1) A reasonable net benefit test would begin with a qualitative approach to evaluate whether there is merit in changing the existing regulatory framework. When the net benefit test is approached from the perspective of existing practices, stakeholder input may be sought.

(2) Unless imposition of a safety improvement or other societal outcome is contemplated, expending resources for a change in regulatory practice would be justified in most cases only if NRC or licensee operations benefit from such a change. The primary source of initial information and feedback regarding potential benefits to licensees would be the licensees themselves.

(3) For the limited purpose of screening potential performance-based changes, consideration of a specific result (such as net reduction in worker radiation exposure) may be sufficient for weighing the immediate implications of a proposed change.

F. The performance-based approach can be incorporated into the regulatory framework.

(1) The regulatory framework may include the regulation in the Code of Federal Regulations, the associated Regulatory Guide, NUREG, Standard Review Plan, Technical Specification, and/or inspection guidance.

(2) A feasible performance-based approach would be one which can be directed specifically at changing one, some, or all of these components.

(3) The proponent of the change to the components of the regulatory framework would have the responsibility to provide sufficient justification for the proposed change; all stakeholders would have the opportunity to provide feedback on the proposal, typically in a public meeting.

(4) Inspection and enforcement considerations would be addressed during the formulation of regulatory changes rather than afterwards. Such considerations could include reduced NRC scrutiny if performance so warrants.

G. The performance-based approach would accommodate new technology.

(1) The incentive to consider a performance-based approach may arise from development of new technologies as well as difficulty stemming from technological changes in finding spare components and parts.

(2) Advanced proven technologies may provide more economical solutions to a regulatory issue without compromising safety, hence justifying consideration of a performance-based approach.

III. <u>Guidelines to Assure Consistency with Other Regulatory Principles</u>

A. A proposed change to a more performance-based approach is consistent and coherent with other overriding goals, principles and approaches involving the NRC's regulatory process.

(1) These principles are provided in the Principles of Good Regulation, the Probabilistic Risk Assessment (PRA) Policy Statement, the Regulatory Guide 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and the NRC's Strategic Plan.

(2) Consistent with the high-level at which the guidance described above has been articulated, specific factors which need to be addressed in each case (such as defense-in-depth and treatment of uncertainties) would depend on the particular regulatory issues involved.

NRC Response to Public Comments:

The Federal Register Notice (FRN), 65 FR 3615 on January 24, 2000, requested comments on the proposed high-level guidelines with particular interest in a set of specific questions. Comments were provided at the March 1, 2000 workshop and in writing. The workshop was conducted as a facilitated discussion among stakeholders representing a wide variety of interests, including NRC representatives from various program offices. Revised guidelines were published in the Federal Register on May 9, 2000 (65 FR 26772), reflecting comments to that point. In addition, an on-line workshop, held on June 8, 2000, provided another opportunity for public comment. Limited comments were received as a result of this workshop.

In the January 24, 2000, FRN, the NRC specifically requested comments on a number of key questions concerning the proposed guidelines. The NRC's response to comments has been structured within the framework of the questions published in the January FRN. Comments not associated directly with any of the questions are shown under the heading "Other Comments."

The NRC's response to the comments and any indication as to how the guidelines have changed in response to the comments follows:

A. <u>Clarity and Specificity of the Guidelines</u>

1. Are the proposed guidelines appropriate and clear?

   Comment: Overall, favorable opinions were expressed regarding appropriateness and clarity of the guidelines. However, two commenters who were generally opposed to any shift to a more performance-based approach provided unfavorable responses. Specifically, those clearly opposed to the performance-based regulatory approach are concerned that its primary purpose is to reduce regulatory requirements and licensee burden thereby compromising the safety standard for overseeing regulated activity. Additionally, there is concern that under a performance-based approach, one would not be able to prevent accidental releases of radioactive material.

   Response: In the NRC's view, the performance-based approach has the potential of making the regulatory decisions more effective and efficient by reducing unnecessary regulatory burden, and do so without compromising overall safety. Further, the guidelines require that in order for an activity to be a viable performance-based candidate, failure to meet its performance criteria will not result in an immediate safety concern. Amplifying guidelines specify that a sufficient safety margin exists, time is available to take corrective action, and the licensee is capable of detecting and correcting performance degradation. Active consideration of all these factors can lead to superior safety standards while avoiding unnecessary regulatory burden. At the same time, the guidelines focus attention on the factors which prevent release of unsafe amounts of radioactive materials.

2. Are there additional guidelines that would improve clarity and specificity?

   Comment: One comment proposed a guideline to increase safety and another comment proposed a guideline to prevent incentives to "perverse" outcomes.

Response: As discussed below, a framework and process to increase safety by adding to regulatory requirements (subject to the Backfit Rule) exists and it would not be efficient to duplicate this through additional guidelines. No changes were made in the main guidelines because safety and beneficial outcomes are generally desirable goals which form parts of normal staff considerations. However, the amplifying guidelines under "Maintain Safety" have been modified to emphasize that safety considerations will play the primary role in NRC's assessments. Since the Commission addressed the matter of encouraging and rewarding improved outcomes in the White Paper (SRM to SECY-98-144, "White paper on Risk-Informed and Performance-Based Regulation)," an amplifying guideline to this effect has been added. This amplifying guideline under overall net benefit generated a comment indicating a misunderstanding that cost would be given a greater emphasis than safety. A revision has been made regarding the considerations related to a simplified net benefit test.

3. How does the "high-level" nature of the guidelines affect the clarity and specificity of the guidelines?

   Comment: The comments provided did not indicate any need to change any of the guidelines due to this factor. One commenter specifically endorsed the "high-level" approach to the guidelines, while also suggesting a graded approach incorporating a minimum acceptable risk.

   Response: The NRC interpreted "minimum acceptable risk" to mean a level of risk consistent with adequate protection considerations. The NRC agrees that a graded approach is appropriate for regulatory changes above and beyond adequate protection. The NRC maintains that the guidelines, as currently formulated, allow for this; thus, no changes were made to address this comment.

B. <u>Implementation of the Guidelines</u>

1. What guidelines, if any, are mandatory for an activity to qualify as a performance-based initiative?

   Comment: Commenters stated that none of the guidelines should be mandatory.

   Response: The viability guidelines must be satisfied for an activity to qualify as a performance-based initiative. In this sense, they may be considered mandatory. For example, a sufficient safety margin must exist. Also, the "Guidelines to Assure Consistency with Other Regulatory Principles" could be considered mandatory because they cover principles which the NRC would not knowingly violate.

2. What is the best way to implement these guidelines?

   Comment: An issue of considerable interest was whether a performance-based approach should be voluntary or not. Certain commenters believed that voluntary changes negatively affect the NRC's inspection and enforcement role whereas others maintained that changes must be voluntary to ensure flexibility on the part of licensees.

   Response: It is anticipated that voluntary implementation will often be proposed, and where mandatory implementation is proposed, such a change would be subject to the Backfit Rule.

Additionally, the NRC has decided to implement the guidelines to new initiatives. Initiatives proposed by stakeholders, such as in petitions for rulemaking, would thus be considered as potential candidates.

3. How should the Backfit Rule apply to the implementation of performance-based approaches?

   Comment: Most commenters indicated that reliance on a performance-based approach would have no bearing on whether or not the Backfit Rule applied. One commenter expressed the view that the Backfit Rule should apply to reductions in regulatory burden.

   Response: The NRC concurs that increased reliance on a performance-based approach poses no unique considerations relative to the Backfit Rule. The NRC fully expects that all new requirements, including those made performance-based, will be subject to existing NRC procedures which include backfit considerations as well as formal regulatory analysis requirements. This comment goes well beyond the scope of these guidelines as currently envisaged.

4. Should these guidelines be applied to all types of activity, e.g., should they be applied to petitions for rulemaking?

   Comment: To the extent that commenters favored application of the guidelines, they also supported application to all activities directed at improving the effectiveness of regulations. One commenter acknowledged that it may not be appropriate for some regulations, such as the Fitness for Duty Rule.

   Response: The NRC intends to apply the guidelines to all activities including responding to and resolving petitions for rulemaking. The commenter who indicated that they were not appropriate for all regulations did not provide a rationale for that position.

5. Should these guidelines only be applied to new regulatory initiatives?

   Comment: A number of commenters from industry preferred wider implementation. For example, one suggestion was to use the guidelines as a screen against existing regulations and to propose changes to the rules based on the potential for significant benefit.

   Response: NRC's current plans are to only implement the guidelines for new initiatives primarily because of NRC resource constraints. However, it should be noted that other mechanisms would continue to exist to identify potential changes to the regulatory framework.

6. Will these guidelines be effective in determining whether we can make a regulatory initiative more performance-based?

   Comment: In general, to the extent that any comments were offered in this regard, the response was in the affirmative.

C. <u>Establishment of Objective Performance Criteria</u>

1. In moving to performance-based requirements, should the current level of conservatism be maintained or should introduction of more realism be attempted?

   Comments: Commenters expressed the view that the appropriate level of conservatism depends on the analysis methodology and the applicable assumptions. Defense-in-depth and uncertainty factors also need to be considered. One commenter stated that it should not be assumed that the level of defense-in-depth remain the same in a performance-based approach.

   Response: The NRC agrees with the commenters and amplifying guidelines have been modified or added under main guidelines associated with "Measurable (or calculable) parameters to monitor acceptable plant and licensee performance exist or can be developed" and "Increase effectiveness, efficiency and realism of the NRC activities and decision-making."

2 What level of conservatism (safety margin) needs to be built into a performance criterion to avoid facing an immediate safety concern if the criterion is not met?

   The comments and response from (C.1) above are also applicable here.

3. Recognizing that performance criteria can be set at different levels in a hierarchy (e.g., component, train, system, release, dose), on what basis is an appropriate level in the hierarchy selected for setting performance-based requirements, and what is the appropriate level of conservatism for each tier in the hierarchy?

   Comment: Oral and written comments expressed the view that performance criteria are best set at the function or system level.

   Response: Some amplifying guidelines which address this issue have been added under the main guideline of "Increase effectiveness, efficiency and realism of the NRC activities and decision-making".

4. Who would be responsible for proposing and justifying the acceptance limits and adequacy of objective criteria?

   Comment: A commenter suggested that the proponent of a change should bear the responsibility for justifying the criteria and the adequacy of acceptance limits.

   Response: The NRC agrees with the commenter. Some amplifying guidelines have been added under the main guideline of "The performance-based approach can be incorporated into the regulatory framework".

5. What are examples of performance-based objectives that are not amenable to risk analyses such as PRA or Integrated Safety Assessment?

   Comment: Examples offered were cross-cutting issues, including fitness-for-duty, safety conscious work environment and management effectiveness.

Response: The NRC agrees with the commenter's examples and they are included in the Commission Paper.

6. In the context of risk-informed regulation, to what extent should performance criteria account for potential risk from beyond-design-basis accidents (i.e., severe accidents)?

   Comment: A commenter stated that risk-informed regulation reaches beyond design basis events by its nature.

   Response: The NRC agrees that risk-informed regulation needs to consider beyond-design-basis accidents.

D. Identification and use of measurable (or calculable) parameters

1. How and by whom are performance parameters to be determined?

   Comment: Comments were presented expressing concern that the NRC would be entirely dependent on licensees' own reports regarding performance. One commenter has stated that information collection at nuclear facilities may require changes to better measure performance. Another commenter raised concerns about licensee honesty and full disclosure.

   Response: The NRC would be responsible for setting the performance parameters with input from stakeholders. Further, the NRC would always maintain vigilance over performance observations. If information collection requirements need to be changed to implement a performance-based approach, such proposals will be addressed in the context of the specific regulatory requirement under consideration. No changes were made in the guidelines based on these comments.

2. How do you decide what a relevant performance parameter is?

   Comment: Some commenters expressed reservations with the use of performance parameters such as core damage frequency as a calculable parameter. Other comments cautioned against drawing broader conclusions (such as overall level of safety or lack thereof) from performance measures than may be justified.

   Response: As these considerations are context specific, and the merits of specific performance parameters are explicitly considered by the guidelines, no changes are proposed in the guidelines. However, on the basis of the experience gained from the limited testing of the guidelines, the scope of what is meant by "performance parameter" has been expanded. It was found that a number of relevant parameters may be required to address the guidelines relative to a given regulatory issue.

3. How much uncertainty can be tolerated in the measurable or calculated parameters?

   Comment: Comments indicate a strong connection between consideration of uncertainty and the level of conservatism in establishing the performance parameters and acceptance criteria.

Response: Changes made in response to (C.1) above are also applicable to this issue.

E. Pilot projects

1. Would undertaking pilot projects in the reactor, materials, and waste arenas provide beneficial experience before finalizing the guidelines?

Comment: Some commenters stated that pilot projects would be useful, and others stated that they were not needed. One commenter suggested that it was important to learn appropriate lessons from implementation of the maintenance rule. Another commented that Option B of 10 CFR 50, Appendix J has already appropriately demonstrated the favorable results from a performance-based regulation.

Response: The NRC plans to apply the guidelines to specific regulations as part of the implementation process and does not currently plan to conduct pilot projects. Based on testing, as reported in Attachment 3, the NRC believes the guidelines are sufficiently developed such that pilots are not needed.

2. What should be the relationship between any such pilot projects and those being implemented to risk-inform the regulations?

Comment: Commenters generally stated that the ongoing pilot projects related to risk-informing the regulations need not be perturbed by including consideration of the guidelines, but appropriate coordination should be maintained. Any screening of regulations should be done one time as opposed to subjecting each regulation to various screenings at different times under different processes.

Response: The NRC proposes to integrate the interfaces between performance-based and risk-informed activities so as to help ensure a more integrated approach and avoid duplication.

F. Other Comments

1. Eliminate all high-level guidelines used to evaluate opportunities for regulatory improvement (II. Guidelines to Assess Performance-Based Regulatory Change):

Comment: One commenter at the public workshop suggested that the set of guidelines to assess performance-based regulatory improvement be eliminated.

Response: The NRC continues to believe that this set of guidelines constitutes an integral part of a structure and logic to consider explicitly the values important to any regulatory improvement program. No changes were made based on this comment.

2. Inclusion of the Advisory Committee on Medical Uses of Isotopes (ACMUI):

Comment: One commenter at the public workshop suggested that ACMUI should be included among the advisory committees which would have an opportunity to review the high-level guidelines.

Response: ACMUI has been included with ACRS and ACNW as committees whose feedback will be sought before the guidelines are submitted to the Commission.

3. Inclusion of perspective from the NRC regions in the work of the Performance-Based Regulations Working Group (PBRWG):

   Comment: One commenter at the public workshop suggested that a representative from the NRC regional offices should be included in the PBRWG, which will play an instrumental role in developing and applying the guidelines.

   Response: Regional representation has been added to the PBRWG.

4. Inspection and enforcement considerations:

   Comment: Comments from within and outside the NRC expressed the need for inspection and enforcement aspects to be front-end considerations. A commenter also suggested that performance above a threshold should result in reduced NRC scrutiny, as long as future departures from good performance would be detectable. Similarly, another commenter supported the notion that past performance could be used to determine the level of flexibility, thereby rewarding or penalizing licensees based on performance history.

   Response: An amplifying guideline has been added under the guideline "The performance-based approach can be incorporated into the regulatory framework" to address this comment.

5. Consideration of a significantly different regulatory paradigm:

   Comment: One commenter offered suggestions to significantly modify the regulatory framework so that any changes undertaken by the NRC would have as a pre-requisite an improvement in the level of safety.

   Response: The NRC notes that current NRC procedures fully allow for identification and implementation of safety enhancements subject to the Backfit Rule. The proposals presented would have wide ranging impacts, and consideration of performance-based initiatives would be only tangentially related to most of them. No specific changes to the guidelines were made in consideration of these comments.

Process and Case Studies Applying High-Level Guidelines

The purpose of this attachment is to present case studies in which the high-level guidelines are applied to specific regulatory provisions. The guidelines to assess viability are emphasized because they represent what is distinctive regarding identifying and assessing performance-based activities. The guidelines were applied to two areas. The first was based on a postulated set of regulatory requirements which the staff hypothesized may be identified as performance-based candidates. The second was a retrospective evaluation of a regulation recently promulgated to assess whether the changes could be seen as having made the existing regulation more performance-based.

Process, Concepts and Definitions

The high-level guidelines to assess viability center on selection or formulation of performance parameters and associated performance criteria. Application of these guidelines depend on certain definitions, which are developed below.

<u>Kinds of "Performance"</u>

In formulating a concept for performance, the staff has drawn on ideas used in the Revised Reactor Oversight Process, in which "performance" refers to those activities in design, procurement, construction, maintenance, and operation that support achievement of the objectives of the cornerstones of safety in the Reactor Oversight Process. In an analogous manner, other applications would entail identification of key aspects of performance and focus on activities which are important to safety.

Risk-significant performance changes generally affect system characteristics such as frequency of events and reliability, availability, or capability of systems, structures, and components (SSCs). Here, "capability" refers to the physical capacity of the system to accomplish a given function, such as "deliver required flow at a given pressure," "successfully bear a given load," or "effectively filter air taken into a breathing apparatus." Availability refers to the fraction of time that the SSC is capable of performing its function. Reliability refers to the probability that a given SSC will function on demand and during the required mission time, given that it was available.

Many kinds of performance affect the system characteristics including such factors as human performance, and the condition in which equipment is left after preventive or corrective maintenance (recognizing that the conduct of testing and maintenance itself affects availability). Ultimately, licensee corrective action programs also affect reliability and availability. Even spare parts management can affect availability.

<u>Characteristics of Functional Safety Requirements</u>

A complete functional safety requirement includes the following:

(1) A definition of the safety mission to be carried out.

This entails at least an implicit specification of the physical challenge that needs to be met. Meeting the challenge will require a level of performance characterized in terms of one or more physical parameters such as flowrate at a particular pressure, or heat removal rate. The system performance specification may be made implicitly, as when a functional outcome is mandated, conditional on a specific challenge (such as maximum peak clad temperature following a specific LOCA, or "no containment failure due to hydrogen combustion" following major core damage).

(2) An indication of the required degree of assurance (functional reliability) that the mission will be carried out successfully.

Assurance of successful performance has previously been approached using concepts such as redundancy (single-failure proof design), special treatment requirements (in procurement, installation, and surveillance), and limiting conditions of operation (so that individual trains or channels of the system cannot be out of service longer than allowed outage times). Surveillance testing or inspection may be mandated at specified intervals so that the probability of undetected faults is limited. System reliability can be promoted by requirements on redundancy, QA, surveillance testing, and allowed outage times.

Implementation Phases of Functional Safety Requirements

There are two distinct kinds of activities involved in implementation of functional safety requirements involving performance parameters. The first kind of activity is associated with design and construction (includes design, procurement, installation and gaining assurance that system design is capable of achieving the desired reliability). The second kind of activity is operational and aimed at maintaining the required reliability and availability. It includes such things as surveillance testing, preventive maintenance, corrective maintenance, and corrective action programs. In the regulatory sphere the first kind of activity is generally associated with licensing. Later plant modifications may also be included. The first kind of activity includes formulation, initial achievement, and subsequent modification of a safety case; the second kind of activity is aimed at keeping the current safety case valid.

Hierarchy of Regulatory Framework

Current regulatory requirements are formulated at several distinct levels which are termed as the hierarchical structure within the regulatory framework. Rules generally state high-level requirements, while lower-level guidance documents provide more specific guidance, including examples of acceptable ways to meet requirements. Technical Specifications and other license conditions also play a role in imposing requirements on licensees. It is found that assessment of the viability of performance-based approaches in a given area is best discussed in light of a comprehensive picture of requirements existing at all of these levels.

Rule Level

The rule states the mission, including the challenges to be addressed and the definition of successful performance. Some existing rules explicitly quantify physical success criteria, such as peak clad temperature, or percentage of metal assumed to react with water to produce hydrogen in certain scenarios.

Evaluation Guidance Level

At this level, which includes both regulatory guides and standard review plans, numerical success criteria are given if they were not stated as part of the rule. These may relate to capability requirements or reliability requirements. Guidance at this level does not have the standing of rules, but it may articulate standards that are considered to be a way to satisfy the intent of rules.

Guidance on acceptable evaluation methods is also provided, including conservative analysis assumptions that may be required in order to assure that conclusions based on the evaluations are robust.

Operational Level (Technical Specifications, Commitments, other elements of the Licensing Basis, etc.)

At this level, requirements are aimed at assuring that assumptions related to safety are upheld. Requirements may be imposed on surveillance test interval and/or test protocol. Technical Specifications may limit the amount of time that the plant is allowed to operate with certain equipment trains out of service. Consensus engineering standards cited by rules are also effectively operational level guidance.

Case Study 1: Combustible Gas Control

This case study applies the viability guidelines to a hypothetical new requirement concerning combustible gas control. The purpose of this hypothetical requirement is to control the probability of containment failure from uncontrolled burns of combustible gas which can occur under certain scenarios in certain containment designs. If the requirement satisfies the viability guidelines concerning measurable performance parameters, objective performance criteria, licensee flexibility, and safety margin, this is an indication that the requirement can be made performance-based.

The case study assumes the following:

- For plants with certain containment designs, some risk-significant scenarios lead to the burning of combustible gas at levels that can threaten containment integrity.
- A technical basis exists for identifying and quantifying risk-significant scenarios and their elements on a plant-specific basis.
- A technical basis exists for quantifying the amounts and rates of generation of combustible gases, and modeling the phenomenology of burns (including the resulting loads).
- A technical basis exists for analysis of containment response to loads caused by combustion of gas.
- A technical basis exists for establishing a needed functional reliability. This could be derived from an argument based on the Quantitative Health Objectives (QHOs), the

frequency at which this function is challenged, and the expected radiological consequences of functional failure of combustible gas control, given that it is challenged.

Formulation of a Requirement on Combustible Gas Control

For purposes of this illustration, a hypothetical requirement on combustible gas control has been formulated that would be applicable to specific classes of plants. This hypothetical requirement on combustible gas control is characterized as follows in terms of the concepts discussed above.

The Safety Issue:

The safety issue is prevention of failure of containment due to loads caused by burning of combustible gases in conjunction with other loads (e.g., steam pressurization, HPME) during risk-significant core damage scenarios that produce significant amounts of combustible gas. The emphasis on "risk-significant" core damage scenarios means that station blackout sequences need to be addressed (including the availability of power for ignition systems) and the phenomenology of core damage scenarios needs to be allowed for, including the amounts and rates of hydrogen generation and the severity of the environments that result. It is also necessary to include methodology for evaluation of containment loads resulting from burns, and specification of required margin on containment performance, if this is warranted.

Physical Definition of Success:

A possible definition of success is "Prevention of containment failure from burning of combustible gas concurrent with other containment loadings, given severe core damage with accompanying evolution of gas."

This is to be assessed using evaluation methods and assumptions mandated in specification of the safety issue (above), and depends on technology. For igniters, it will be necessary to specify physical ignition capability: surface temperature, number, and distribution.

Depending on implementation of technology selected, Technical Specifications on capability may be warranted (specification of the physical ignition capability required to be confirmed by test).

Specification of Functional Reliability Needed To Meet Requirement:

As discussed earlier, the desired functional reliability can be determined from such considerations as the QHOs, the consequences of functional failure, and the frequency of challenges to this function (the frequency of severe core damage). In the discussion that follows, it is assumed that such a determination has been carried out, and that for plants in the class subject to this requirement, the overall functional failure probability is to be maintained well below 0.1. This probability is conditional on the scenario ingredients called out previously, such as station blackout. This assumption bears on licensee flexibility and on the feasibility of detecting performance changes within a reasonable time.

As formulated, this hypothetical requirement specifies evaluation methodologies with respect to the challenge and definition of success. These evaluations could be carried out on a plant-specific basis, or for classes of plants; for purposes of the present case study, it is tacitly assumed that each plant carries out the evaluations according to the acceptable methodologies. The performance parameters thus derived will take credit for aspects of containment performance that are themselves the subject of other requirements, which may be prescriptive. The hypothetical requirement does not force a choice of technology.

Application of the Viability Guidelines

The following aspects of the overall requirement, as hypothesized, warrant consideration as areas that could be performance-based: igniter capability, functional reliability, division reliability, and division availability. (For this case study, the choice of igniter technology is presumed, although this choice might not be made in all cases.) Atmospheric mixing is a related area that could be performance-based, but it is not treated here. The following discussion applies the four viability guidelines to each potential performance-based area in turn.

Igniter Capability:

In order to succeed, the igniter function must provide sufficient physical capability (e.g., enough surface area at a sufficiently high temperature). The functional reliability associated is discussed separately.

Guideline IA: Several capability parameters exist: surface temperature, number, and distribution.

Guideline 1B: Criteria for each of these parameters can be developed based on ignition phenomenology.

Guideline IC: Within igniter technology, relatively little flexibility in achieving these parameters may exist, but choice of technology itself may be allowed.

Guideline ID: Provided that performance is actually monitored periodically, so that the failure is detected in test and not in an actual accident scenario, not meeting the criterion does not immediately cause a safety concern. This is based on the fact that the frequency of severe core damage is itself limited.

Functional Reliability:

Here, the phrase "functional reliability" refers to the probability that the ignition function will be carried out successfully, given that a need for the function arises. Since the function may be performed by a collection of SSCs, which may be designed to allow for some failures, the functional reliability depends on lower-level figures of merit such as division-level, train-level, or component-level reliability and availability.

Guideline IA: This guideline is met. At the functional level, for this case, it would be calculated from division and component level performance and availability data.

Guideline IB: This guideline is met. Functional reliability criterion is derivable as indicated above from QHO arguments, or could be formulated based on other lines of reasoning.

Guideline IC: Choice of technology is one level of flexibility. Within igniter technology, there is flexibility in system redundancy and in licensee management of division availability.

Guideline ID: Declining reliability is not an immediate safety concern. This is based on the fact that the frequency of severe core damage is itself limited.

Division Reliability:

Here, the phrase "division reliability" refers to the reliability of a functional subset of the igniter function. In fact, divisional redundancy may not be required for this function – it is possible that a single division might meet the requirement. The present discussion tacitly assumes that some redundancy would be incorporated into the design. Depending on the design, the functional reliability requirement would then be decomposed into division reliability requirements and division availability requirements.

Guideline IA: Division reliability would be calculated from component level performance data.

Guideline IB: An objective criterion can be developed based on the functional reliability criterion discussed above.

Guideline IC: There is flexibility in design and in operational practices to meet this requirement.

Guideline ID: Declining reliability is not an immediate safety concern. This is based on the fact that the frequency of severe core damage is itself limited.

Division Availability:

Here, the phrase "division availability" refers to the availability of a functional subset of the igniter function. In fact, divisional redundancy may not be required for this function - it is possible that a single division might meet the requirement. The present discussion tacitly assumes that some redundancy would be incorporated into the design.

Guideline IA: Division availability would be evaluated directly from test and maintenance records.

Guideline IB: An objective criterion would be developed, based on system redundancy, the functional reliability criterion and the division reliability criterion discussed above.

Guideline IC: Flexibility exists in licensee management of maintenance.

Guideline ID: Not meeting the availability criterion would not be an immediate safety concern. In addition to factors cited above for other parameters, the availability criterion has the property of being relatively easily observable, in that changes in performance are not masked by statistical fluctuations.

Summary

For active ignition technology, several capability parameters were identified. These satisfy some of the remaining guidelines in that they are measurable, criteria exist, and failure to meet performance criteria does not result in an immediate safety concern. However, within igniter technology, there may not be very much flexibility in meeting these criteria. Other technologies could be considered. Inquiry needed to establish the practicality or necessity of monitoring the efficacy of atmospheric mixing was not carried out.

Reliability parameters satisfy three of the four guidelines and might satisfy the fourth. Criteria can be derived, flexibility is afforded, and failure to satisfy reliability requirements is not an immediate safety concern. However, whether it is practical to confirm reliability through monitoring is a plant-specific evaluation. Viability requires that unacceptable performance cause enough failure events within a reasonable monitoring time to manifest the current (degraded) performance level. For this system, it is expected that quantitative evaluation would lead to a satisfactory finding for this guideline as well.

Therefore, the viability guidelines are substantially satisfied by several key elements of this requirement. A substantially performance-based version of this requirement would be viable. However, as noted previously, the evaluations carried out for this area will take credit for passive containment performance under severe conditions including high temperatures. Performance-basing of requirements on these less-testable aspects of containment integrity may not be viable. Moreover, this hypothetical requirement mandates evaluation of the frequency of this particular functional challenge (i.e., the frequency of severe core damage events that challenge this function). This frequency itself reflects credit for satisfaction of requirements that may not be performance-based. Nevertheless, the utility of the guidelines has been demonstrated to identify components of the regulatory framework which can be made substantially performance-based.

Case Study 2: Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas

This case study applies all three groups of guidelines to examine the recent changes to 10 CFR 20, Subpart H, Respiratory Protection and Controls to Restrict Internal Exposures. The stated goals of the revision were to revise the requirements to reflect current guidance (ANSI and OSHA) and to make the requirements for radiological protection less prescriptive while reducing unnecessary regulatory burden without reducing worker protection. A review of the changes made to the requirements indicates three generic types of changes:

1. Administrative changes that clarify the requirements,

2. Regulatory framework changes to the structure of the requirements resulting in a more logical order (e.g., moving Appendix A footnotes to the regulatory text), and

3. Regulatory changes that actually change the requirements explicitly identified in the rule and thus may impact the licensees' regulatory burden.

The purpose of this case study is to apply the three groups of guidelines to specific regulatory requirements and determine whether the revised rule can be judged to be more performance-based than the prior version of the rule. Hence, the guidelines are being applied as an assessment tool to the changes made to the rule by the recent revision, and not to the rule as a whole. The assessment was performed using a sampling approach. To assess the impact of the change to Subpart H, three of the changes to the rule were analyzed. The three changes selected were of the third type above. One change reflected an increased regulatory burden, one a reduction in regulatory burden, and one an overall neutral impact on the regulatory burden.

Application of the Viability Guidelines

The sample of three rule changes are examined below:

(i) A provision to reduce regulatory burden was contained in §20.1702(b), which added text to permit licensees to consider safety factors other than radiological factors when performing an ALARA analysis to determine whether or not respirators should be used. Applying the viability guidelines to assess this change results in the following:

Guideline I.A.: The parameters should reflect licensee performance of the ALARA program as well as consider non-radiological factors that affect worker safety. Under the original rule requirements, the non-radiological factors had to be considered, but were divorced from the radiological ALARA determination. This could have resulted in reduced worker protection from non-radiological factors while licensees sought to meet ALARA requirements. Measurable or calculable parameters would be available from performance history associated with the non-radiological and ALARA factors. When compared to the prior version of the Subpart H requirements, the revised requirement would only require identification of parameters associated with non-radiological safety factors, such as trending of occupational health and safety incidents, in addition to parameters associated with radiological factors.

Guideline I.B.: Objective criteria to assess performance of a licensee's ALARA program exist in the form of past performance. Objective criteria on performance of a licensee's ALARA program could be based on trending of worker doses.

Guideline I.C.: The prior version of the requirement allowed licensee flexibility by the definition of ALARA. The revised requirement provides another degree of freedom for the ALARA analysis by including non-radiological safety factors. Under the revised requirement, it is possible for the ALARA analysis to result in higher doses to workers but lower overall risk to the workers once non-radiological safety factors are included. By allowing slightly higher worker doses in this scenario, the NRC has provided the licensee increased flexibility. Thus, flexibility is increased with the revised requirement.

Guideline I.D.: By definition, the ALARA program operates in a dose regime that does not correspond to an immediate safety concern. Generally, the airborne concentrations of radioactive material are such that failure of performance criteria will not result in an immediate safety concern. By including non-radiological safety factors, the revised requirement should result in lower total risk. Thus, the revised requirement should generally increase the safety margin. On occasion, hazards may be such that a failure of equipment might result in a relatively small safety margin. These rare cases result in more prescriptive requirements for equipment that will be discussed in further detail in the next requirement change example.

Summary – This change expands the scope of the ALARA analysis by including non-radiological safety factors. This introduces greater flexibility by not requiring respirator use in some circumstances in which it would previously have been required. The licensee may, however, expend some extra effort in justification. The net effect may be to decrease overall licensee burden. In summary, this change satisfies the viability guidelines, making the revised rule more performance based than the prior version.

(ii) A provision that increased regulatory burden was contained in §20.1703(c)(6) which added text to require fit testing before first field use of tight-fitting, face sealing respirators and at least annual testing thereafter. The quantitative criteria for successful fit testing are also codified. The prior version of the rule only included a requirement that the licensee's respiratory protection program include written procedures for fitting. The revised rule does not alter these requirements, but includes specific requirements for fit testing frequency and quantitative criteria for test fit factors that must be achieved during testing in order to use the Appendix A APFs. These new specific requirements explicitly provide lower-level (less outcome-oriented) objective criteria for assessing fit testing. Both the prior version of the rule and the revised rule included a requirement that the licensee include surveys and bioassays, as necessary, to evaluate actual intakes in the respiratory protection program. Applying the viability guidelines to assess this change results in the following:

Guideline I.A.: The parameters that measure desired outcomes associated with this requirement, dose due to internal exposure, are not affected by this change. The revised requirement explicitly mentions lower-level parameters for monitoring performance, but these parameters do not measure outcomes and were implicit in the prior version of the rule.

Guideline I.B.: Objective criteria to assess performance of a licensee's fit testing exist. The revision simply explicitly stated some of the objective criteria for fit testing.

Guideline I.C.:The prior version of the rule allowed licensee flexibility by only specifying that a written procedure for fitting be included in the respiratory protection program. The revision adds requirements at a lower level: it increases the specificity of requirements imposed by the rule. Thus, application of the third viability guideline would indicate that the revised rule may be less performance-based.

Guideline I.D.: For performance in the area of respirator equipment fitting, sufficient safety margin may not exist when performance criteria are not met. As discussed above in the analysis of the ALARA program, hazards may be such that a failure of the respirator fitting properly may result in a relatively small safety margin. In addition, time is not available for taking corrective action due to the nature of the hazards, such as internally deposited radioactive material or non-radioactive airborne materials, and the typical frequency of surveys and bioassays. These scenarios require prescriptive requirements for fit testing. In addition, since proper fit is assumed when making dose calculations for legal records, prescriptive requirements are necessary to provide the proper assurance of accuracy. This guideline therefore corresponds to the motivation for the rule change.

Summary – This revision to the rule does not make the rule more performance-based. However, the reason for this is that sufficient safety margin and time for taking corrective action do not exist in the event the performance criteria are not met. The viability guidelines indicate that this area of the rule is not suitable for performance-based activities and support the motivation for the rule change.

(iii) A provision considered neutral relative to regulatory burden was included in the rulemaking relative to §20.1703(a)(6) [which becomes §20.1703(e) in the revised rule] such that text was added to require consideration of low temperature freezing of exhaust valves on negative pressure respirators, and removed text that specified protection against skin contamination. The only difference between the prior version of the rule and the revised rule for this particular change is the list of requirements explicitly mentioned by the rule that need to be considered when selecting respiratory protection equipment. Adding the requirement for consideration of low temperature work environments increases the analysis effort explicitly required. Removing the requirement for consideration of skin contamination requires the licensee to address skin contamination using means other than respiratory equipment. Applying the viability guidelines to assess this change results in the following:

Guideline I.A.: The parameters would be equivalent for the prior version of the rule and the revised rule.

Guideline I.B.: The objective criteria may be based on performance history.

Guideline I.C.: Although the list of requirements explicitly mentioned changes, the net affect on licensee flexibility is negligible. The level of specificity of the explicit requirements does not change. Since the objective criteria remain equivalent, the flexibility is unchanged by the change to the Subpart H requirements.

Guideline I.D.: Failure to meet the performance criteria of either the prior version of the rule or the revised rule could lead to situations that do not provide sufficient safety margin or time for taking corrective actions. For example, failure to consider low temperature work environments could result in exhalation valves on negative pressure respirators to freeze in the open position due to moisture from exhaled air when temperatures are below freezing. This situation would provide a pathway for airborne hazards, such as radioactive material, to bypass the respirator filter without the users knowledge. Thus, requirements are necessary to provide worker protection while in radioactive areas. This guideline therefore corresponds to the motivation for the rule change.

Summary – The revised rule is neither more or less performance-based than the prior version of the rule. The specific requirements changed in this example are prescriptive due to the fact that sufficient safety margin and time for taking corrective action do not exist in the event the performance criteria are not met. This example does demonstrate the validity of using the viability guidelines to assess performance-based activities and support the motivation for the rule change.

Conclusion: Application of the guidelines to the three selected changes to the rule indicates that the changes appear to comport with the guidelines. A premise in the testing of the guidelines was that the process of testing may indicate a need to change one or more of the guidelines. The guidelines worked well as they are and no changes are proposed as a result of the testing.

Application of the Guidelines to Assess Performance-Based Regulatory Change

For completeness, the changes to the requirements of Subpart H were evaluated against the remaining performance-based guidelines to verify that the changes resulted in a net regulatory benefit. For this evaluation, the composite of all the changes must be evaluated to provide the integrated consideration required, rather than evaluating each change individually. Thus, the results of the sampling approach above are extrapolated to include all changes to the rule when necessary. However, this evaluation is based primarily on the existing results contained in the staff's Statement of Considerations and the Regulatory Analysis for the amendment of Subpart H requirements.

Guideline II.A.: The following factors were noted:

- Allowing the consideration of non-radiological safety factors when performing an ALARA analysis results in an overall reduction in the worker's risk from all hazards;

- Explicitly identifying fit test criteria, intended to ensure that sufficient margin of safety (specifically, proper fit) is maintained under field and work conditions, increases assurance that respiratory equipment will perform as expected during use;

- Explicitly identifying environmental factors, such as low temperatures, for consideration in determining respiratory protection increases assurance that the proper operation of respiratory equipment will not be adversely affected during use.

Guideline II.B.: The following factors were noted:

- Identifying regulatory requirements in the amended rule text and removing guidance from the rule, such as moving some of the Appendix A footnotes to the regulatory text and deleting some that are addressed in the Regulatory Guide, clarifies the requirements and reduces confusion;

- Recognizing new devices and new technologies updates the rule to reflect current practices by licensees;

- Allowing use of single-use disposable masks when ALARA analysis indicates that respiratory protection is not necessary, provides a means for addressing respiratory protection equipment when requested by the worker.

Guideline II.C.: The following factors were noted:

- Including decontamination to reduce resuspension of radioactive material in the work place provides an effective and efficient means of controlling internal dose instead of using respirators;

- Adopting the existing guidance of ANSI, such as reduced equipment assigned protection factors (APFs) provides consistency;

- Adopting the existing requirements of OSHA, such as fit testing frequency and fit factors for positive pressure, continuous flow, and positive-demand devices, provides consistency.

Guideline II.D: The following was noted:

- Each amendment to the rule was reviewed by the staff to determine the impact on licensee burden and the conclusion was that 13 amendments reduced burden, 3 amendments increased burden, and 36 amendments had no impact on burden; with the net result being a reduction in licensee burden.

Guideline II.E: The following was noted:

The backfit analysis performed by the staff for the amendments concluded that the changes constitute not only a burden reduction, but also a substantial increase in the overall protection of public (worker) health and safety. Based on a review of public comments, public confidence is not significantly affected by the rule amendments. However, it is assumed that the substantial increase in the overall protection of worker health and safety would result in an associated increase in public confidence. The Regulatory Analysis estimated a net benefit of $1.5 million per year, including the cost to revise licensee procedures. Finally, since this is an amendment to an existing rule, the regulatory framework can inherently incorporate the approach into the existing regulatory framework. Thus, the existing Regulatory Analysis adequately addresses the regulatory improvement guidelines, demonstrating that the amendments to the rule result in a net regulatory benefit.

Application of the Guidelines to Assure Consistency with Other Regulatory Principles

The revision is inherently consistent with other regulatory principles. However, use of the guideline will support the assertion that the guideline is valid for evaluating future performance-based activities. The revised rule is consistent with 1992 American National Standards Institute (ANSI) guidance for respiratory protection and respiratory protection regulations published by Occupational Safety and Health Administration (OSHA). The findings of the environmental assessment analysis state that the revised rule is expected to result in a decrease in the use of respiratory protection and an increase in engineering and other controls to reduce airborne contaminants while maintaining total occupational dose as low as reasonably achievable. Thus, subject to the limitations of the sampling approach used, the revision to the rule is consistent with other regulatory principles.

Wed 8/30
@ 9:30

(13)

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

August 29, 2000

MEMORANDUM TO: ACRS Members

FROM: Noel Dudley, Senior Staff Engineer
Noel Dudley

SUBJECT: ACRS REVIEW PLANS FOR LICENSE RENEWAL GUIDANCE DOCUMENTS

During the June 7-9 and July 12-14, 2000 ACRS meetings the Committee discussed and approved a plan, assignments, and guidance proposed by Dr. Bonaca for reviewing license renewal guidance documents. The final plan, assignments, and guidance are attached. I plan to provide each member with a copy of the proposed Standard Review Plan, Regulatory Guide, and associated NEI 95-10. Since the Generic Aging Lessons Learned (GALL) report is over 2,000 pages, individual members will receive only those sections of the Report assigned to them for review. A complete copy of the GALL Report will be provided upon request.

Attachments:
1. License Renewal: Plan for Reviewing Guidance Documents
2. ACRS Review Assignments for License Renewal Guidance Documents
3. License Renewal: Guidance for ACRS Review of Guidance Documents

cc.: J. Larkins
H. Larson
S. Duraiswamy
ACRS Fellows and Staff

Attachment 1

LICENSE RENEWAL
PLAN FOR REVIEWING GUIDANCE DOCUMENTS

The staff plans to brief the Plant License Renewal Subcommittee in October 2000 concerning drafts of the Standard Review Plan, Generic Aging Lessons Learned II Report, and Regulatory Guide related to preparation and review of license renewal applications. The Nuclear Energy Institute (NEI) has revised its NEI 95-10 Report , which provides guidance to licensees concerning implementation of the requirements for preparing a license renewal application. The staff plans to review and endorse NEI 95-10 in a regulatory guide.

The staff has held meetings with NEI and the industry concerning these documents and plans to issue draft documents for public comment in August 2000. The staff plans to hold a public workshop in September and brief the Committee at the November 2000 ACRS meeting.

All ACRS members should participate in the review of these generic guidance documents, since they may become members of the Plant License Renewal Subcommittee in the future. To ensure the Committee members have sufficient time to conduct a thorough and integrated review of these document, the following course of action is recommended:

- assign Members primary responsibilities for reviewing specific portions of the documents,
- provide pre-draft documents to the members in May 2000,
- discuss Committee approach for reviewing generic documents at the June ACRS meeting,
- provide draft public comment generic documents to the members in August 2000,
- schedule a half an hour session at the September ACRS meeting to discuss reviewing the documents (NRR will provide an overview),
- members attend the September NRC workshop,
- schedule a half an hour session at the October ACRS meeting to discussed members' issues and concerns,
- Plant License Renewal Subcommittee meeting in October to review generic documents,
- review and comment on the documents at the November 2000 ACRS meeting, and
- review proposed final documents at the March 2001 ACRS meeting.

ACRS REVIEW ASSIGNMENTS FOR LICENSE RENEWAL GUIDANCE DOCUMENTS

Updated August 28, 2000

REVIEW ITEMS	Assigned Member	Standard Review Plan Sections			GALL Report Chapters	NEI 95-10 Chapters
Introduction or Administrative Information	MVB ALL	1.0			I	1.0
Scoping and Screening Methodology	MVB ALL	2.1				3.0
Plant	JDS DAP	2.2			IV	4.0
Reactor Coolant System	RLS WJS	2.3	3.2	4.2	V	6.0
Engineered Safety Features	TSK GBW	2.3	3.3	4.2	VII	6.0
Auxiliary Systems	JDS REU	2.3	3.4		VIII	6.0
Steam and Power Conversion	JDS REU	2.3	3.5		III	6.0
Structures	JDS DAP	2.4	3.6		VI	6.0
Electrical and I&C	REU GA	2.5	3.7	4.4		6.0
Time-Limiting Aging Analyses	REU DAP			4.1 4.8		5.0
Reactor Vessel	DAP RLS			4.2	IV	6.0
Containment	TSK GBW			4.6	II	6.0

Attachment 3

LICENSE RENEWAL
GUIDANCE FOR ACRS REVIEW OF GENERIC DOCUMENT

The proposed Standard Review Plan (SRP) for license renewal provides guidance on an acceptable method for applying the scoping and screening criteria to identify the long lived passive structures and components. The SRP provides guidance on how to reference the Generic Aging Lessons Learned (GALL II) report and to identify the aging effects and the acceptable aging management programs or activities. Items to consider during review of the license renewal generic documents and the proposed licenses renewal process include:

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

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

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

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

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

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

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

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
AUGUST 30, 2000

LICENSE RENEWAL GENERIC ACTIVITIES

Sam Lee - Senior Material Engineer RLSB/NRR

License Renewal Generic Activities

Agenda

- Background, Overview and Schedule
- Generic Aging Lessons Learned (GALL)
- Standard Review Plan
- Regulatory Guide and NEI 95-10
- Solicitation of Comments

Background

- Guidance provided by SRM for SECY 99-148
  - Document basis for acceptance of existing programs
  - Focus on areas where existing programs should be augmented
  - Develop documents with stakeholder participation
  - Brief Commission on public comments
  - Commission approval
  - Recommendation on rulemaking after additional review experience

Overview

- GALL report and SRP intended to work together
- Draft Regulatory Guide (DG-1104) proposes to endorse NEI 95-10
- Invite stakeholders comments
  - Workshop held on December 6, 1999
  - 12 public meetings held from March-July 2000
  - Workshop scheduled for September 25, 2000
- Documents have been integrated to the extent practicable

Schedule

Item	Date	Actual
Issue draft GALL, SRP, and RG/NEI 95-10 for public comment	8/00	8/31/00
Public meeting and workshop to gather public comments	9/00	9/25/00
NEI revise NEI 95-10	10/00	
ACRS License Renewal Subcommittee Meeting	10/00	
ACRS Full Committee Meeting	11/00	
Commission briefing on public comments on draft GALL, SRP, and RG/NEI 95-10	11/00	11/27/00
ACRS meeting on GALL, SRP, and RG/NEI 95-10	2/01	
Commission approval of GALL and SRP	3/01	
NEI comment on need for rulemaking	4/01	
Public meeting to discuss need for rulemaking	5/01	
Staff recommendation to Commission on rulemaking	7/01	

Generic Aging Lessons Learned Report

- Build on previous GALL report (NUREG/CR-6490)
- Review aging effects
- Identify relevant existing programs
- Evaluate program attributes to manage aging effects

Generic Aging Lessons Learned Report

Table of Contents for Volume 1 (Summary)

Generic Aging Lessons Learned Report

Table of Contents for Volume 2 (Tabulation of Results)

Standard Review Plan

- Reference GALL report for crediting existing programs
- Incorporate lessons learned and resolution of license renewal issues
- Compatible with standard format of license renewal application

Standard Review Plan

Table of Contents

Regulatory Guide for License Renewal

- DG-1047 issued 8/96
  - endorsed Nuclear Energy Institute (NEI) 95-10, Rev 0
- DG-1104 to be issued 8/00
  - proposes to endorse NEI 95-10 Revision 2

NEI 95-10 Revision 2

Table of Contents

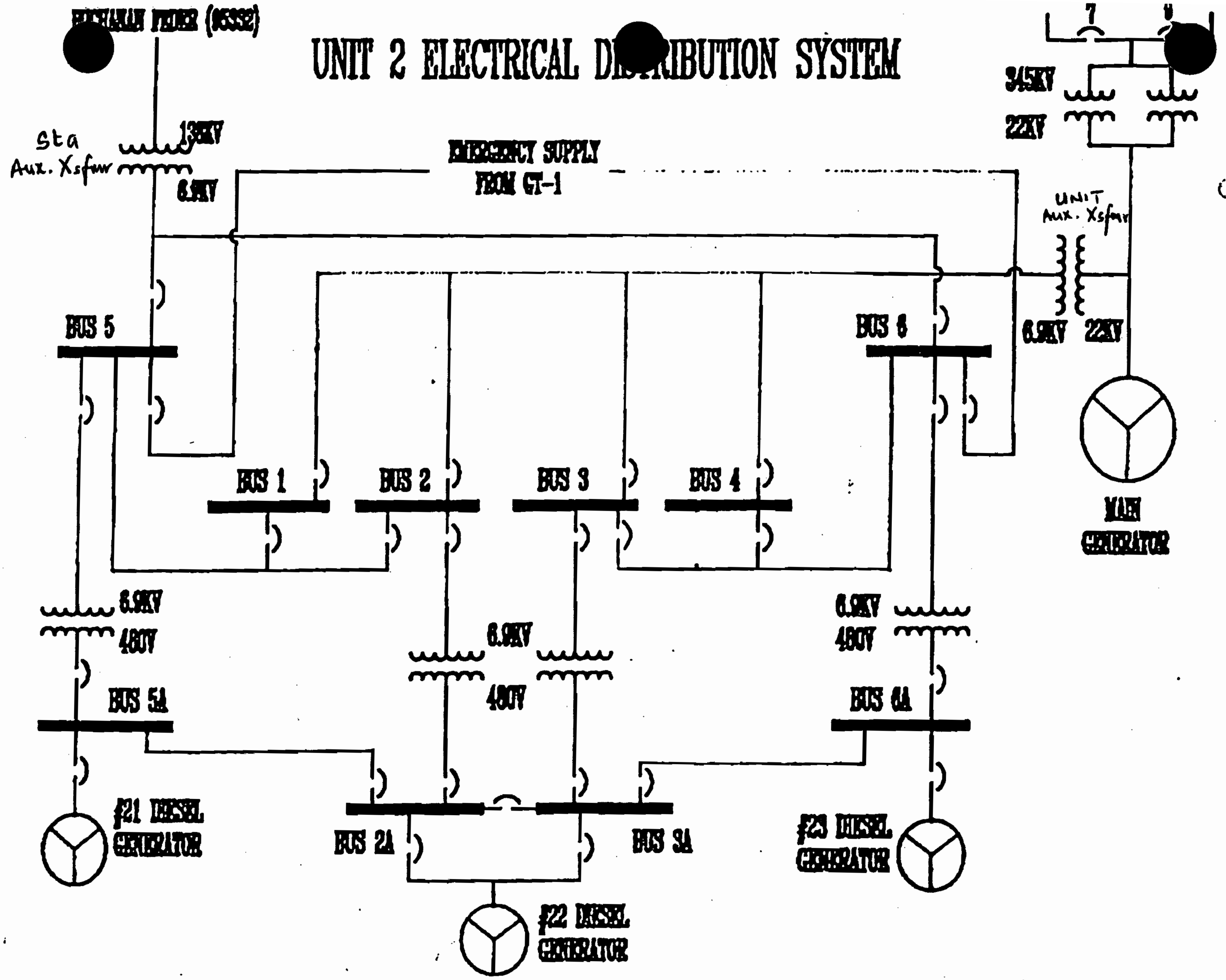
Solicitation of Comments

- Does the draft GALL report provide sufficient credit for existing programs?
- Does the draft GALL report provide too much credit without sufficient technical basis?
- How should the GALL report reference editions of national codes and standards that are not subject to the Commission's approval process?
- Should the applicant be required to justify the omission of any aging effects identified in the GALL report that the applicant determined not to be applicable?

STANDARD REVIEW PLAN FOR LICENSE RENEWAL
TABLE OF CONTENTS

TABLE OF CONTENTS (CONTINUED)

UNIT 2 ELECTRICAL DISTRIBUTION SYSTEM



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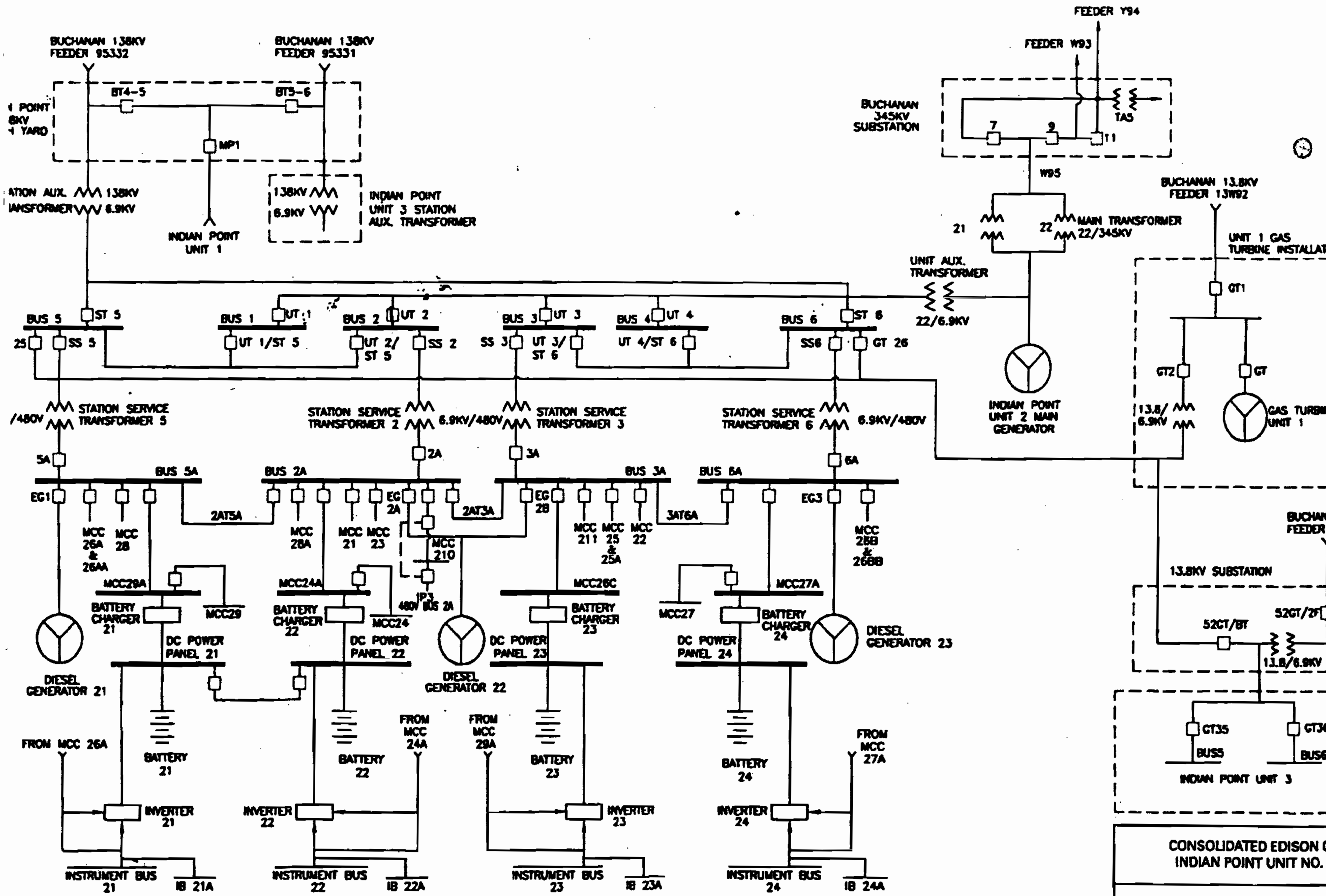
TOTAL P.15



CONSOLIDATED EDISON C
INDIAN POINT UNIT NO. 2

Figure 8.2-2
Electrical Power System
Diagram

ACRS BRIEFING
AUGUST 30, 2000
INDIAN POINT 2 EVENTS

EVENT 1: REACTOR TRIP AND
PARTIAL LOSS OF VITAL POWER
AUGUST 31, 1999

EVENT 2: STEAM GENERATOR TUBE FAILURE
FEBRUARY 15, 2000

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INDIAN POINT 2 EVENTS

PRESENTERS

- Opening Remarks: Ledyard B. Marsh
  - Chief, Events Assessment, Generic Communications and Non-Power Reactors Branch, NRR
- Introduction: Eric J. Benner, NRR
- 8/31/1999 Event: Jimi T. Yerokun, Region I
- 2/15/2000 Event: Raymond K. Lorson, RI
- Risk Insights: James M. Trapp, RI
- Closing Remarks: Brian E. Holian
  - Deputy Director, Division of Reactor Safety, Region I

INDIAN POINT 2 EVENTS

INTRODUCTION

- Reactor Trip and Partial Loss of Vital Power
  - August 31, 1999
- Steam Generator Tube Failure
  - February 15, 2000
- Aspects of Events to be Discussed:
  - Sequence of events
  - Licensee response
  - Safety Significance
  - Root cause areas
  - Risk insights

Reactor Trip and Partial Loss of Vital Power

- Initiator
  - Reactor Trip
    - Channel 3, OTDT in "Trip" for Maintenance
    - Spurious Actuation of Channel 4, OTDT
- Complications
  - (1) Offsite power lost to all vital 480 volts buses
  - (2) Essential power (EDG) lost to 480 volt bus 6A
- Result: Loss of one 125 VAC Instrument bus
  - Loss of >75% CR Annunciators
    - Declaration of Unusual Event

SEQUENCE OF EVENTS

- Reactor Trip - Aug 31, 1999, 2:31 P.M.
  - Four 6.9 kV Buses Transfer From Unit to Station Auxiliary Transformer - as designed
- Offsite Power Lost to Vital 480 Volt Buses
  - (2A, 3A, 5A and 6A)
  - EDGs Started
- EDG 23 Output Breaker to Bus 6A Opens
  - Battery Charger 24 De-energized

Sequence of Events (continued)

- Battery 24 Depleted (~ 7.5 hours)
  - Loss of 125 VAC Instrument Bus 24
  - Loss of > 75% CR Annunciators
- Unusual Event Declared (8/31, 9:55 P.M.)
- Emergency Power Restored To Bus 6A
- Unusual Event Terminated (9/1, 3:30 A.M.)
- Offsite Power Restored to Bus 6A

SAFETY SIGNIFICANCE

- Loss of Bus 6A
- Loss of Battery 24
- Increased Burden to Operators

ROOT CAUSE AREAS

- Configuration Control
- Management Oversight
- Technical Support
- Corrective Actions

CONFIGURATIONAL CONTROL

- Station Aux. Transformer Load Tap Changer
  - Control Room Switch Not Maintained in "AUTO"
- Vital Bus Degraded Voltage Relay Setting
  - Reset Set Point Not Verified
- EDG 23 Breaker Over-Current Trip Setting
  - Not Properly Set (3200 Vs. 6000 amps)

MANAGEMENT OVERSIGHT, TECHNICAL SUPPORT AND CORRECTIVE ACTIONS

- Weak Response During The Event
  - Focus on Shutdown Work Plans
  - Coordination/Use of Resources

- Weak Technical Support Before The Event
  - Degraded Voltage Relay Setting
  - Procedures - EP, 480 Volt Bus Recovery

- Inadequate Corrective Actions
  - Prior RPS OTDT Anomalies
  - Repair of Load Tap Changer

RISK SIGNIFICANCE

- CCDP ~ 2E-4

- Dominant Sequence
  - Loss of one MDAFW Pump + Loss of TDAFW Pump + Failure to Recover Feedwater

- Key Assumptions
  - No Credit for 480 Volt Bus Recovery
  - Bleed and Feed Success needs 2 of 2 PORVs

INDIAN POINT UNIT 2
Steam Generator Tube Failure
February 15, 2000

- Sequence of Events
- Safety Significance
- Root Cause Areas
- Risk Significance

EVENT DESCRIPTION

- Initiator: PWSCC of the R2C5 tube of the #24 SG; initial primary to secondary leak rate of approximately 150 gpm.

- Complications: Several operator, procedural and equipment problems delayed establishing cold, shutdown conditions.

- Results:
  - The plant remained in an "Alert" Status ~24 hours
  - Minor radiological release.

SEQUENCE OF EVENTS

<u>February 15, 2000</u>

7:17 p.m.	--	Operators Identified Increased SG Leak
7:29 p.m.	--	Declared Alert
7:30 p.m.	--	Tripped Reactor
7:41 p.m.	--	State/County Officials Notified
8:31 p.m.	--	Isolated Affected SG
9:02 p.m.	--	Operators Initiated Plant Cooldown
9:04 p.m.	--	Manually Initiated Safety Injection
11:38 p.m.	--	Tube Leak Stopped

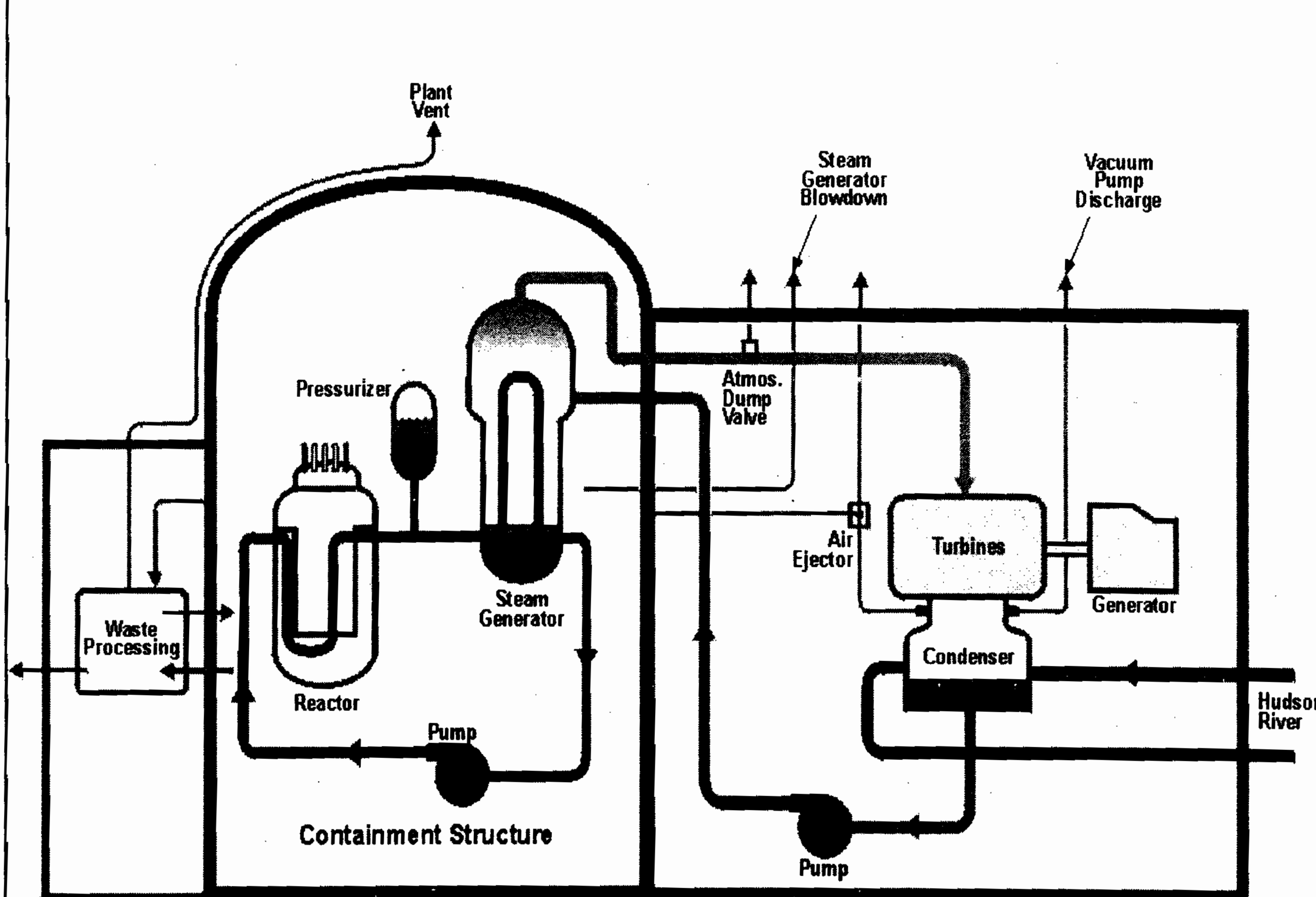
<u>February 16, 2000</u>

12:39 p.m.	--	Shutdown Cooling System
4:57 p.m.	--	Achieved Cold Shutdown
6:50 p.m.	--	Terminated Alert

SAFETY SIGNIFICANCE

- Initial Operator Response Prompt/Appropriate
- Licensee Successful in Achieving Cold Shutdown
- Several Operator Performance/Procedural Issues, and Equipment Issues Identified Which Delayed Achieving Cold Shutdown Conditions
- Several Emergency Response Problems
- No Measurable Offsite Radiological Release Impact (consistent with calculated results)
- No Impact on Public Health and Safety

INDIAN POINT 2



ROOT CAUSE AREAS

- Operator Performance
- Procedural Adequacy
- Equipment Performance
- Emergency Response

OPERATOR PERFORMANCE

- Initial Response Prompt and Appropriate; Procedure Adherence Good Overall

- Some Deficiencies in the Plant Cooldown Phase

  - Initial Cooldown Excessive (led to SI)
  - Operator Recognition of Plant Configuration (CCW Valve Configuration, Auxiliary Spray)

PROCEDURE QUALITY

- Procedures (AOPs/EOPs) to Guide Initial Response were Good

- Several Procedural Deficiencies Challenged Operators During the Plant Cooldown Phase

  - Delayed Placing Shutdown Cooling In-Service
  - System Configuration (CCW Valves, Aux Spray)
  - Shutdown Conditions (RCS Temperature)

EQUIPMENT PERFORMANCE

- Event Mitigation Systems Worked Properly
  - Reactor Protection System
  - Auxiliary Feedwater System
  - Safety Injection System
- Some Pre-existing Equipment Problems Challenged Operators
  - Automatic Condenser Vacuum Control Valve
  - Condenser Mechanical Vacuum Pump
  - Containment Valve Seal Water System Design Problem
  - Pressurizer Power Operated Relief Valve Design Problem

EMERGENCY RESPONSE

- Emergency Response Protected Health and Safety of Public
- Event Classified Properly/Good Critique of Emergency Response

- Emergency Plan/Implementing Procedure Problems
  - Augmented Emergency Response Facility Staffing Not Timely
  - Accountability Problems
  - Emergency Response Data System (ERDS) not Operable for Several Hours (Pre-Existing Problem)
  - Problems in Implementation of the Media Response Plan
  - Emergency Response Facility Equipment Problems
  - Technical Support Timeliness and Quality Issues

- Supplemental EP Inspection

RISK SIGNIFICANCE

Actual Event Risk:

- Initial estimated CCDP for a SGTR ~ 1E-4 GEM/SPAR & ~7.7E-5 based IPE
- Revised CCDP based on actual leak rate was ~ 2.2E-6

Key Assumptions:

- Actual SGT failure leak rate ~ 100gpm - HRA revised accordingly
- Charging pumps available for HP makeup

SDP Conditional Risk Assessment:

- Delta-CDF is used to determine risk significance of inspection findings
- Deficiencies with the 1997 SGT inspection program have a high delta-CDF and are risk significant

Key Assumptions:

- SGT failure IE frequency ~ 1/RY
- ½ tube failures result in ruptures

SUMMARY

Supplemental Inspections/Actions

August 1999 Event
Emergency Preparedness
Steam Generator Tube Failure Root Cause
Issuance of Information Notice 2000-09

Agency Focus (5/23/00)

Communication and Coordination
Engineering Support
Configuration Management /Control
Equipment Reliability/Large Backlog
Operator Knowledge, Station Training, Procedures
Emergency Preparedness

Public Meeting

September 11, 2000 - On Site

ACRS MEETING HANDOUT

(18)

Meeting No.	Agenda Item	Handout No:
475th	17	17.2

Title **FUTURE ACRS ACTIVITIES**

Authors **JOHN T. LARKINS**

List of Documents Attached

FUTURE ACRS ACTIVITIES - 476TH ACRS MEETING - 10/5 - 7, 2000

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From Staff Person
JOHN T. LARKINS

FINAL DRAFT: 8/24/00

II. ITEMS REQUIRING COMMITTEE ACTION

1. Differing Professional Opinion on Steam Generator Tube Integrity (Open) (DAP/SD) ESTIMATED TIME: 3 hours

Purpose: Determine a Course of Action

Review requested by the Executive Director for Operations [F. Miraglia, OEDO]. The Executive Director for Operations requested that the ACRS act as the panel to review a Differing Professional Opinion concerning steam generator tube integrity issues.

An ad hoc Subcommittee consisting of D. Power (Chairman), M. Bonaca, T. Kress, M. Shack, and J. Sieber has been proposed to review this matter. A subcommittee meeting will be scheduled for the week of November 13-17, 2000. Subject to satisfactory completion of the subcommittee review, this matter will be scheduled for the December ACRS meeting.

The Planning and Procedures Subcommittee will recommend a course of action.

2. Proposed Revision to the Revised Reactor Oversight Process (Open) (JDS/MWW) ESTIMATED TIME: 1 ½ HRS.

Purpose: Determine a Course of Action

Review requested by the Commission [M. Johnson, NRR]. In a Staff Requirements Memorandum dated April 5, 2000, the Commission requested the ACRS to review the use of performance indicators (PIs) in the revised reactor oversight process (RROP) to ensure that the PIs provide meaningful insights into aspects of plant operation that are important to safety. The Commission also requested the ACRS to review the initial implementation of the significance determination process (SDP) and assess the technical adequacy of the SDP to contribute to the RROP.

The staff is in the process of revising the performance indicators. They are developing metrics for the assessment of the of the RROP

The Committee last reviewed the RROP in March 2000 and provided a report to the Commission dated March 15, 2000. In that report, the Committee supported the staff's proposed initial implementation of the RROP but offered comments and recommendations on the choices of PIs and associated thresholds, completeness of the SDP, and further development needed for full and effective implementation. In accordance with the Staff Requirements Memorandum dated

April 5, 2000, the Committee plans to continue its review of the results of the use of performance indicators and the SDP subsequent to initial implementation of the RROP. The staff has suggested that an interim briefing during the November 2000 ACRS meeting might be appropriate. However, results of the assessment and the revision of the performance indicators will not be available for ACRS review until March-April 2001.

The Committee needs to decide if it wishes an information briefing/discussion in November, rather than wait until the first of the year to hear the results of or efforts to respond to the SRM. As a result of information presented during the visits to Davis Besse and Region III, the Committee might want to consider an interim information briefing to the subcommittee before having a full Committee briefing. A subcommittee briefing would provide an opportunity to discuss some of the findings that resulted from the visits in more detail.

The Subcommittee recommends that an information briefing by the staff be scheduled for the October ACRS briefing.

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

   Purpose: Determine a Course of Action

   Review requested by the NRC staff [Michael Jamgochian, NRR]. The staff briefed the Committee on its reevaluation of power reactor physical protection regulations and its position on a definition of radiological sabotage at the May 2000 ACRS meeting. The staff is preparing a proposed revision to 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage." The staff plans to provide the ACRS with a copy of the proposed revision by August 31, 2000.

   Dr. Kress has agreed to recommend a course of action after receiving the document.

4. RETRAN-3D Transient Analysis Code (Open) (GBW/PAB) ESTIMATED TIME: 1 hour

   Purpose: Determine a Course of Action

   Review requested by the NRC staff [R. Caruso, NRR]. As part of its Thermal-Hydraulic (T/H) Code Review Action Plan, NRR initiated review of the EPRI RETRAN-3D thermal-hydraulic transient analysis code. The code is designed for analysis of FSAR Chapter 15 transients (excluding Appendix K LOCA analysis), and plant events. The T/H Phenomena Subcommittee began its review of this code during its December 16-17, 1998 meeting.

NRR had developed a detailed schedule for reviewing the RETRAN-3D code. In accordance with this review schedule, the T/H Phenomena Subcommittee met on March 23, 1999. A Subcommittee report was provided to the Committee during its April 1999 meeting.

Dr. Wallis conducted a detailed review of portions of the RETRAN code documentation. He has identified several issues of a significant nature with the models and correlations used in the "3D" version of the code. NRR has also identified a number of significant issues regarding the code modeling. In addition, EPRI was required to modify its "five equation" flow model to correct known errors. A meeting was held on June 29, 1999 between NRR and EPRI to address these matters. The outcome of the meeting gave indication that a significant amount of work remains before completing the review of this code.

Dr. Wallis provided a report to the Committee during the July 1999 ACRS meeting regarding his concerns. The Committee considered a draft letter to the EDO on this matter, but the letter was tabled. During the September meeting, the Committee discussed the direction to be taken by the ACRS regarding future review of the RETRAN-3D code. It was agreed that the Committee would defer further action on this matter, pending receipt of the staff's review document.

Representatives of NRR and EPRI discussed the status of the RETRAN review during the March 15, 2000 T/H Phenomena Subcommittee meeting. Dr. Wallis reported the results of the subcommittee meeting to the ACRS during its April meeting. He said that the subcommittee plans no future action on this matter, subject to further action from the NRR staff. NRR has recently provided the ACRS with a copy of the draft SER. However, two issues need to be resolved prior to its final issuance: (1) EPRI must formally respond to the list of conditions specified by the staff in the SER for use of the code, and, (2) EPRI and NRC are in dispute relative to the need for EPRI to pay for the staff's review.

The Planning and Procedures Subcommittee recommends that Dr. Wallis propose a course of action after reviewing the draft SER.

5. SECY-00-0145,"Integrated Rulemaking Plan for Nuclear Power Plant Decommissioning" (Open) (TSK/MME) ESTIMATED TIME: 1 ½ hours

Purpose: Decide on a Course of Action

Review requested by the ACRS. The subject SECY, "Integrated Rulemaking Plan for Nuclear Power Plant Decommissioning" issued on June 28, 2000, requests Commission approval to proceed with developing an integrated rulemaking for nuclear power plant decommissioning. The regulatory areas addressed by this rulemaking plan are emergency planning, insurance, safeguards, staffing and training, and backfit.

The staff briefed the ACRS in April 2000 regarding the draft technical study on spent fuel pool accident risk at decommissioning nuclear power plants. The ACRS issued its report on April 13, 2000 regarding this issue and the first recommendation was "The integrated rulemaking on decommissioning should be put on hold **until** the staff provides technical justification for the proposed acceptance criterion for fuel uncovery frequency. ....." The staff, however, on page 3 of SECY-00-0145 (second paragraph) states that "The staff believes that the ACRS comments **will not** impact the overall conclusions of the staff's risk study."

SECY-00-0145 describes sample regulatory languages for emergency planning, insurance, security, operator staffing and training, and applying the backfit rule. The staff also did not approve NEI's request for adapting 10 CFR Part 50 to decommissioning plants. The subject SECY provides two options on this issue, namely;

- **Option 1**, approval of this rulemaking could be placed on hold until the staff has provided the Commission a more comprehensive assessment of decommissioning regulatory improvements, due to the Commission on September 15, 2000.

- **Option 2,** approve the initiation of the integrated rulemaking plan.

The staff indicated its preference for Option 1. However, the staff's reason seems to be the absence of any anticipated nuclear power plant decommissionings in the near future, rather than the importance of the ACRS comments and the inadequacy of the technical study.

Dr. Kress plans to provide his views on the need for the Committee to review this matter following the staff's presentation on the revised technical study of spent fuel pool accident risk at decommissioning plants during the November 2000 ACRS meeting.

ANTICIPATED WORKLOAD
AUGUST 29-SEPTEMBER 1, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Markley	Risk-Informing 10 CFR Part 50: Option 2 guidelines for STP exemption request, comments on ANPR, and Option 3 framework document for evaluating technical requirements (10 CFR 50.44)	Report		P&P 8/28 (P.M.)
		Markley	Assessment of the Quality of PRAs	Report		
Bonaca	Seale	Dudley	Overview of License Renewal Guidance Documents - **Information Briefing**	--	--	P&P 8/28 (P.M.)
Kress	--	Duraiswamy	AP-1000 Pre-application Review (Phase 1)	Report	--	
Powers	All Members	Larkins/Duraiswamy, et.al	Preparation for meeting with the Commissioners	--	P&P 8/28 (P.M.)	
	--	El-Zeftawy	Research Report to the Commission	--		

ANTICIPATED WORKLOAD
AUGUST 29-SEPTEMBER 1, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Seale	--	Weston	Proposed Final Regulatory Guide (DG-1093) on Design-Bases Information	Report	--	
		Weston	Causes and Significance of Design-Basis Issues- Information Briefing	--		
		Weston	Operating Events at Indian Point Unit 2-Information Briefing	--		
Sieber	--	Dudley	Performance -Based Regulatory Initiatives	Report		
Wallis	--	Boehnert	Subcommittee report on SRELAP-5 Code			

ANTICIPATED WORKLOAD
October 5-7, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Markley Markley	Proposed ANS Standard for PRA Quality (Phase 2)- **Dr. Apostolakis to develop assignments for members after receiving the draft ANS Standard.** **Staff views on ASME Std. For PRA** Quality	Report	RPRA 10/4 (P.M.)	P&P 10/4 (A.M.)
Kress	--	Dudley	Proposed Change to CFR 73.55, Physical Security Requirements	Report		M+M 9/21
Powers	All Members	Larkins El-Zeftawy	Meeting with the Commission Research Report to the Commission	--	P&P 10/4 (A.M.)	M+M 9/21
Shack	Wallis	Dudley	PTS Technical Basis Revaluation Project	Report	M+M 9/21	
Sieber	Leitch	Weston	Proposed Revision to the Reactor Oversight Process	Report		PS 10/31

ANTICIPATED WORKLOAD
October 5-7, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Uhrig	--	Singh	GSI-168:EQ of Electrical Equipment	Report	PS 10/31	RPRA 10/4(P.M.)
Wallis		Boehnert	Draft SER on EPRI RETRAN-3D Code **(Tentative)**			

ANTICIPATED WORKLOAD
November 2-4, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	Leitch	Markley Markley	Risk-Informed Regulation Implementation Plan Proposed final modifications to SRP Chapter 19 and R.G. 1.74 re. Use of risk-informed decisionmaking in license amendment reviews	Report Report		P&P 10/31 (P.M.) PLR 10/19-20 RES 11/1
Bonaca	Seale	Dudley	License Renewal Documents: SRP, GALL II, and Regulatory Guide	Report	PLR 10/19-20	P&P 10/31 (P.M.) RES 11/1
Kress	--	El-Zeftawy	Spent Fuel Pool Accident Risk at Decommissioning Plants	Report	SAM 11/15	PLR 10/19-20 RES 11/1
Powers	All Members	El-Zeftawy	Research Report to the Commission	Report	P&P 10/31 (P.M.) RES 11/1	
Seale		Weston	Regulatory Effectiveness of ATWS Rule	Report (Tentative)		PLR 10/19-20 RES 11/1

ANTICIPATED WORKLOAD
November 2-4, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Sieber	Powers --	Singh	Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues	Report	10/23-24	RES 11/1 PS 10/31
Uhrig	--	Duraiswamy	Proposed Update to 10 CFR Part 52	--	PS 10/31	RES 11/1 PLR 10/19-20
	--	Singh	ABB/CE and Siemens Digital I&C Applications (Subcte. Report) **The P&P Subcommittee recommends that the Committee consider issuing a "generic" report tothe Commission, outlining issues of concern in the area of digital I&C**	--		

G Drive:PP (19)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No:
475th	17	17.1

Title MINUTES OF PLANNING & PROCEDURES SUBCOMMITTEE MEETING - AUGUST 28, 2000

Authors JOHN T. LARKINS

List of Documents Attached

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From Staff Person
JOHN T. LARKINS

MINUTES OF THE PLANNING AND PROCEDURES SUBCOMMITTEE MEETING MONDAY, AUGUST 28, 2000

The ACRS Subcommittee on Planning and Procedures held a meeting on August 28, 2000, in Room 2 B1, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 1:00 p.m. and adjourned at 3:45 p.m.

ATTENDEES

D. A. Powers, Chairman
G. Apostolakis
M. Bonaca

ACRS STAFF

J. T. Larkins
H. Larson
R. P. Savio
S. Duraiswamy
C. Harris
S. Meador
M. El-Zeftawy
U. Shoop

EDO STAFF

G. Millman

DISCUSSION

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

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

RECOMMENDATION

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

2) Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through November 2000 is included in a separate handout. The objectives are to:

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

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

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate. The Committee needs to consider the Subcommittee's recommendations on items listed in Section II of the Future Activities.

3) Differing Professional Opinion (DPO) Issues Associated with Steam Generator Tube Integrity

In a memorandum dated July 20, 2000, to the ACRS Executive Director (p. 1), Dr. Travers, Executive Director for Operations (EDO), requested that the ACRS assist in the process to review a DPO on steam generator tube integrity issues. Specifically, the EDO requested that the ACRS function as the equivalent of an ad hoc panel, under the NRC Management Directive 10.159 to review the DPO.

Subsequent to the EDO memorandum, the DPO author requested a meeting with the ACRS Executive Director. On July 24, 2000, Dr. Larkins and Mr. Duraiswamy met with the DPO author to discuss the EDO's request to the ACRS, previous ACRS comments on Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes," and other related matters. During that meeting, the DPO author stated that he did not have any objection to the ACRS reviewing the DPO issues as requested by the EDO, but has some concerns that he would like to bring to the attention of the EDO. In a memorandum dated July 28, 2000, the DPO author provided his concerns to the EDO (pp. 2-3). The EDO responded to the DPO author on August 4, 2000 (p. 4) stating that: "In selecting the ACRS as the ad hoc panel, I considered its previous involvement in and knowledge of the technical issues." Dr. Larkins also sent a memorandum to the DPO author on August 14, 2000 (pp. 5-6) documenting the items discussed with the DPO author on July 24, 2000. The EDO plans to provide consultants (Dr. Catton, Thermal-Hydraulic Issues; Dr. Richer, NIST, IGSCC; and Mr. Higgins, BNL, Human Performance) to the ACRS to provide technical support in reviewing the DPO issues.

RECOMMENDATION

The Subcommittee recommends that the Committee:

- Review the technical merits of the DPO issues and authorize the ACRS Chairman to send the attached response to the EDO (pp. 7-11)

- Establish an ad hoc subcommittee (Chairman D. Powers, Members: M. Bonaca, T. Kress, W. Shack, and J. Sieber)
- Review and report on the DPO issues at the November/December 2000 meeting, subject to satisfactory completion of the ad hoc Subcommittee's review during October 2000. A meeting of the ad hoc Subcommittee is tentatively scheduled for October 10-13, 2000.

Dr. Powers will discuss the review process and specific assignments to the members. Dr. Apostolakis suggests that the Committee consider using Dr. Ballinger, MIT, as a consultant to provide technical support to the Committee in reviewing metallurgical issues associated with the DPO.

4). <u>ASLB Decision on Shearon Harris</u>

The ACRS reports on spent fuel pool fires at decommissioning plants and the report on generic safety issue for spent fuel pools for operating plants have been referenced in the ASLB petition on Shearon Harris' amendment to its operating license to modify its spent fuel pool (pp. 12-32). As a result of interveners referencing the ACRS reports in their case to support the need for NRC staff to prepare an environmental impact statement, the ACRS members, staff, or consultants could be subject to discovery in these proceedings, which may require ACRS members, staff, or consultants to provide testimony or written material for these hearings.

The Board of Commissioners of Orange County (BCOC), North Carolina, is seeking admission of four late-filed environmental contentions (ECs) in the matter of Carolina Power & Light Co. (Shearon Harris Nuclear Power Plant). The Atomic Safety and Licensing Board (ASLB) on August 7, 2000, ordered that one contention (EC-6) to be admitted for litigation; and rejected three contentions (EC-7, EC-8, EC-9) as inadmissible for litigation.

The ASLB in its ruling ordered the parties to conduct discovery beginning on August 21, 2000, and ending on October 20, 2000. The ASLB also notes (note #4) that any attempt to obtain discovery materials from the ACRS is subject to the exceptional circumstances of 10 CFR 2.720 (h).

<u>RECOMMENDATION</u>

The Subcommittee recommends that if the members receive any requests for testimony or material related to this ASLB hearing on Shearon Harris, they should notify the ACRS Chairman and Executive Director. The ASLB will decide on what material should be included in the discovery process.

5) <u>Power Uprate Issues</u>

Mr. Boehnert has summarized the list of issues associated with power uprates along with an anticipated schedule for ACRS review of power uprate applications (pp. 33-68)

Also, as instructed by the Committee, Dr. Cronenberg, ACRS Senior Fellow, has developed a list of central issues associated with power uprates. This list was distributed to the members during the July 2000 ACRS meeting for review and comment. So far, no comments have been received.

RECOMMENDATION

The Subcommittee recommends that the members review the proposed list of issues and provide comments during the September ACRS meeting. The Committee needs to decide how it wants to disposition Dr. Cronenberg's issues and what strategy should be pursued in reviewing the uprate applications.

6) Technical Exchange Meeting with RSK

During the July 2000 ACRS meeting, the Committee has selected November 6-10, 2000, for a technical exchange meeting with RSK. The RSK has agreed to these dates for this meeting. ACRS members Apostolakis, Bonaca, Kress, Sieber, and Wallis plan to attend this meeting. Current plans would include travel on Sunday, November 5 to Germany and travel to Erlangen for a visit and discussion with Siemens and GRS consultants on digital I&C systems. Subsequently, we would travel to Munich, Garching, for a meeting with members of the RSK and GRS and BMU to discuss I&C issues, use of PRA in the regulatory process, future research needs for reactor safety, and other generic safety issues of interest to either Committee. Subsequently, we can return home on Thursday, November 9. Additionally, the RSK members have suggested a visit to the nuclear power plant at Neckarwesthein, near Studgard. However, the Committee members indicated in the past that they were not particularly interested in visiting a reactor site in Germany.

RECOMMENDATION

The Subcommittee recommends that Dr. Uhrig propose an agenda for this meeting and that the Committee discuss and finalize the agenda during the September meeting

7) American Nuclear Society 2000 Utility Working Conference

Mr. Noel Dudley, ACRS staff, attended the ANS 2000 Utility Working Conference held at the Amelia Island Plantation, Florida, on August 6-10, 2000. The primary focus of the conference was on managing the business of nuclear power. A summary report prepared by Mr. Dudley is attached (pp. 69-76).

RECOMMENDATION

The Subcommittee recommends that the ACRS staff brief the Committee members with regard to what they should and should not do in filling out the Compensation Report Form and that the Mr. Dudley provide a brief presentation to the full Committee during the September meeting. Also, the Chairman or Vice Chairman of the Committee should attend this meeting in the future

8) New ACRS/ACNW Compensation Report Form

The ACRS/ACNW Member Compensation Report (blue sheet) has been revised to capture data on how much time members spend on the review of technical topics (e.g., license renewal, AP 1000, etc.). Members will be provided with copies of the revised report and are requested to begin using them for all compensation claims submitted after this meeting.

RECOMMENDATION

The Subcommittee recommends that members address any questions or comments they have concerning the revised Compensation Report to Carol Harris.

9) Member Issues

10.1) License Renewal White Paper

The Subcommittee discussed a paper prepared by Dr. Bonaca (pp. 77-79) on Potential Synergistic Effects of Industry Initiatives to Extend Plant Life, Increase Production, and Reduce Regulatory Burden.

RECOMMENDATION

The Subcommittee recommends that the issues raised by Dr. Bonaca be included in the Research report to the Commission.

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

July 20, 2000

MEMORANDUM TO: John Larkins
Executive Director
Advisory Committee on Reactor Safeguards

FROM: William D. Travers
Executive Director for Operations

SUBJECT: DIFFERING PROFESSIONAL OPINION ON STEAM GENERATOR TUBE INTEGRITY ISSUES

The purpose of this memorandum is to request that the Advisory Committee on Reactor Safeguards (ACRS) assist in the process to review a Differing Professional Opinion (DPO) on Steam Generator Tube Integrity Issues. Specifically, I am requesting that the ACRS function as the equivalent of an ad hoc panel, under Management Directive (MD) 10.159, to review the DPO.

The issues raised in the DPO are reflected in the Staff Consideration Document dated November 1, 1999, and the DPO Reply Document dated December 16, 1999 (and attachments). Consideration of this differing professional opinion (DPO) has been proceeding according to a memorandum dated December 29, 1998, included as an attachment, which established a three-step approach. Step (1) publication of specific documents for public comment, and Step (2) preparation of a final staff position, have been completed. The author of the DPO, has completed his part of Step (3) by reviewing the staff's final position and providing a response in which he identifies areas which he believes are still unresolved. The appointment of an ad hoc panel to address the remaining issues completes Step (3). We have attempted to establish an ad hoc panel comprised of members of the NRC staff who are suitable for the task and acceptable to the DPO author. However, these attempts have been unsuccessful. In light of the broad expertise and independence of the ACRS, I am requesting that for this particular DPO, the ACRS function as the equivalent of an ad hoc panel described in MD 10.159.

This DPO deals with complex technical issues. After completing the review, I request that the ACRS provide me a summary report that documents its conclusions and any recommendations relative to the pertinent technical issues.

Since 1991, an extensive record of documentation has been developed on the underlying technical issues. These documents would be provided to the ACRS to assist in the review. To facilitate transferring the collected documentation and information regarding the DPO, please contact my staff to establish a mutually agreeable time to meet.

Thank you for your assistance in reviewing this important matter.

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

July 28, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Dr. Joram Hopenfeld
Engineering Research Applications Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research

SUBJECT: DIFFERING PROFESSIONAL OPINION ON STEAM GENERATOR TUBE INTEGRITY ISSUES

Your July 20, 2000, memorandum informed me that you have requested the Advisory Committee on Reactor Safeguards (ACRS) to function as the DPO ad- hoc panel. The following are my concerns about your request:

1. Since ACRS members can not devote more than 50% of their time to NRC related business they may not be able to pack into their already busy schedule the amount of time that the DPO review requires. Over the past 10 years, both the industry and the NRC have generated numerous documents to promote the practice of leaving defective steam generators in service. These documents, which deal with highly complex technical subjects, contain inaccurate, inconsistent and misleading information, as well as important assumptions which are poorly stated. Additionally, certain subjects have implied assumptions which are not stated. I estimated that it would take about two months for each panel member to properly and adequately review the necessary documents to make a valid technical determination of the facts.

2. The ACRS has participated in the agency positions which are at issue. In 1994 (Attachment 1), the ACRS agreed to the implementation of GL-95-05 "Voltage Based Repair Criteria For Westinghouse Steam Generator Tubes" and disagreed with the DPO claim that the GL represents a serious safety risk to the public. In 1997, in a disagreement with the DPO position, the ACRS concluded that GL-98-XX "Steam Generator Tube Integrity" may be released for public comments (Attachment 2).

The DPO position has always been that the Voltage Based Criteria as prescribed in GL-95-05 should not be accepted by the NRC as a substitute for the 40% plugging rule. The ACRS has previously rejected the position of the DPO. There were adequate technical information and uncertainties at the time of the ACRS's decision concerning voltage based repair criteria to have rejected that, but the ACRS did not do so.

It should be noted, however, that ACRS approval of GL 95-05 was conditioned on that GL-95-05 was an interim measure and the understanding that the tube support plate provided structural constraints. Both of these stipulations have been shown to be incorrect. The ACRS also expressed reservations regarding how radiological releases were calculated, the lack of adequate data base for leakage calculations, and the lack of adequate techniques for detection and characterization of degradation. There has been no significant progress in these fields since GL-95-05 was released.

cc: Chairman Meserve
John Larkins

Accession Number: ML003735901

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

August 4, 20000

MEMORANDUM TO: Joram Hopenfeld
Engineering Research Applications Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research

FROM: William D. Travers
Executive Director for Operations

SUBJECT: DIFFERING PROFESSIONAL OPINION (DPO) ON STEAM GENERATOR TUBE INTEGRITY ISSUES

Your letter dated July 28, 2000, expresses two concerns with selection of the Advisory Committee on Reactor Safeguards (ACRS) as the ad hoc panel for your DPO. The concerns refer to the timeliness of the ACRS review and previous involvement of the ACRS in issues related to the DPO.

I considered timeliness and objectivity before requesting that the ACRS function as the equivalent of an ad hoc DPO panel. On timeliness, the ACRS will develop a schedule that integrates its review of the DPO with those of its other priorities. This will provide for review by the ACRS of the DPO consistent with its other duties. In selecting the ACRS as the ad hoc panel, I considered its previous involvement in and knowledge of the technical issues that concern the DPO. On balance, I believe the ACRS will provide an informed and objective evaluation of the technical issues.

cc: ✓J. Larkins, ACRS

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SEPARATE HANDOUT - ITEM 10

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

August 14, 2000

MEMORANDUM FOR: Joram Hopenfeld
Division of Engineering Technology
Office of Nuclear Regulatory Research

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

SUBJECT: MEETING ON JULY 24, 2000, TO DISCUSS THE JULY 20, 2000 MEMORANDUM FROM THE EDO ON THE PROCESS FOR REVIEWING THE DPO ISSUES

On July 20, 2000, the Executive Director for Operations (EDO) sent a memorandum to me requesting that the ACRS function as the equivalent of an ad hoc panel, under Management Directive (MD) 10.159, to review the differing professional opinion (DPO) on steam generator tube integrity issues and provide a summary report documenting its conclusions and any recommendations. Subsequent to the issuance of this memorandum, at your request, Mr. Sam Duraiswamy of my staff and I met with you in my office on July 24, 2000 and discussed the EDO's request to the ACRS, previous ACRS position on Generic Letter 95-05, expertise needed on the Committee to review the technical merits of the DPO issues, and other related matters.

During that meeting, you stated explicitly that you did not have any objection to the ACRS reviewing the DPO issues as requested by the EDO, but you have some concerns you would like to bring to the attention of the EDO. Also, you agreed to provide a list of subissues along with a clear definition of each of the issues and identify the expertise needed to review each DPO issue. On July 25, 2000, you provided a list of expertise needed to review the main DPO issues, and instead of providing a list of subissues, you provided a list of questions under each issue without explaining the rationale behind preparing these questions.

In your memorandum of July 28, 2000, to the EDO, you outlined your concerns about the EDO's request that the ACRS function as the equivalent of an ad hoc panel, under MD 10.159, to review the DPO issues. Your memorandum did not mention the fact that you already informed Mr. Duraiswamy and me that you had no objection to the ACRS reviewing the DPO issues and providing its conclusions and recommendations to the EDO.

To alleviate your concerns about previous ACRS positions on the DPO issues and related matters, the Committee will revisit its previous comments and recommendations and will attempt to minimize the influence of previous decisions in formulating its conclusions and

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recommendations to the EDO. The Committee's recommendations will be unbiased, independent, and will be based on its review of the technical merits of the DPO issues.

cc: EDO
D. Powers, ACRS Chairman
G. A. Apostolakis, ACRS Member
M. V. Bonaca, ACRS Member
S. Duraiswamy, ACRS

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Rowe:DPO

MEMORANDUM FOR: William D. Travers
Executive Director for Operations

FROM: D. A. Powers, Chairman, ACRS

SUBJECT: DIFFERING PROFESSIONAL OPINION ON STEAM GENERATOR TUBE INTEGRITY ISSUES

In a memorandum dated July 20, 2000, to the ACRS/ACNW Executive Director, you requested ACRS assistance in the technical resolution of a Differing Professional Opinion associated with steam generator tube rupture events. Specifically, you requested that the ACRS function as the equivalent of an ad hoc panel, under Management Directive 10.159, to review the differing professional opinion (DPO) on steam generator tube integrity issues, and provide you with a summary report documenting the conclusions and any recommendations relative to the pertinent technical issues. The ACRS has agreed to your request and will in the next few weeks establish a schedule for reviewing the technical issues associated with the DPO.

SCOPE OF ACRS REVIEW

In addition to accepting your request, this memorandum attempts to clarify the scope of the ACRS review. We understand that the scope of the ACRS review is to assess the technical merits of the DPO issues and provide its recommendations for your use in resolving the DPO. We assume that the main DPO issues, noted below, are accurately defined in the "Differing Professional Opinion Consideration Document," which is attached to your memorandum, dated November 1, 1999 to Dr. Hopenfeld.

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- NDE Issue
- MSLB Issue
- Risk Increase Issue
- Iodine Spiking Issue
- Severe Accident Issues

Although ACRS will focus on the issues in the DPO Consideration Document, dated November 1, 1999, and the DPO Authors Response to the EDO, dated January 5, 2000, there may be ancillary issues that the Committee may need to consider as part of its review. In performing this task, the Committee plans to review the referenced documents ~~unless instructed otherwise.~~ as well as other relevant documents.

During a meeting between the ACRS Executive Director and a member of his staff on July 24, 2000, Mr. Hopenfeld agreed to have the ACRS serve as an ad hoc panel for reviewing the technical issues of his DPO except he expressed some concerns about previous ACRS decisions as noted in his recent memorandum to you on July 28, 2000. We understand Dr. Hopenfeld's concerns about previous ACRS positions on these issues and we will attempt to minimize the influence of previous decisions in our review. To the extent practicable, the Committee will revisit its previous comments and recommendations on this matter included in the reports and letters listed below.

- ACRS report dated September 12, 1994, from T. S. Kress, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Proposed Generic Letter 94-xx, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes."

- ACRS letter dated May 15, 1995, from T. S. Kress, ACRS Chairman, to James M. Taylor, EDO, Subject: Proposed Final Generic Letter 95-xx, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes."
- ACRS letter dated November 20, 1996, from T.S. Kress, ACRS Chairman, to James M. Taylor, EDO, Subject: Proposed Rule on Steam Generator Integrity.
- ACRS letter dated October 10, 1997, from R. L. Seale, ACRS Chairman, to L. Joseph Callan, EDO, Subject: Resolution of the Differing Professional Opinion Related to Steam Generator Tube Integrity.

PROPOSED REVIEW PROCESS

Currently, the Committee plans to establish an Ad Hoc Subcommittee to review the technical merits of the DPO issues. The Subcommittee will function under the provisions of the Federal Advisory Committee Act (FACA). The Subcommittee and the full Committee will use the consultants, you have agreed to provide, to obtain technical support in reviewing certain DPO issues. After an initial meeting, the Subcommittee will decide on the scope and need for additional meetings. At the conclusion of the subcommittee's review, the full Committee will discuss this matter and provide you with a letter, documenting its independent views on the DPO issues.

References:

1. Memorandum dated November 1, 1999, from William D. Travers, EDO, to Joram Hopenfeld, RES, Subject: Differing Professional Opinion on Steam Generator Tube Integrity Issues, with attachments:

a. Differing Professional Opinion Consideration Document

b. Public comments on Draft Regulatory Guide, DG-1074, "Steam Generator Tube Integrity."

2. Memorandum dated December 16, 1999, from Joram Hopenfeld, RES, to William D. Travers, EDO, Subject: Differing Professional Opinion on Steam Generator Tube Integrity Issues (Response to the November 1, 1999 memorandum from the EDO) with attachments:

a. Letter dated September 12, 1994, from T. S. Kress, Chairman, ACRS, to I. Selin, Chairman, NRC, "Proposed Generic Letter 94-xx, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes."

b. Letter dated October 21, 1997, from R. L. Seale, Chairman, ACRS, to S. A. Jackson, Chairman, NRC, "Summary Report - Four Hundred Fortieth Meeting of the Advisory Committee on Reactor Safeguards."

c. J. Hopenfeld Comments on the Thermal Hydraulic Analysis in NUREG-1570, ACRS Materials and Metallurgy Subcommittee & Severe Accidents Subcommittee, March 5, 1997.

d. Memoranda dated December 23, 1991 and March 27, 1992, regarding Differing Professional View

e. Memorandum dated September 11, 1992, from J. Hopenfeld to E. Beckjord, "Addendum to March 27, 1992, Memo Regarding Degraded Steam Generator Tubes," September 11, 1992.

f. Memorandum dated September 28, 1999, from J. Hopenfeld to W. D. Travers, "DPO Panel Review of Steam Generator Integrity."

g. J. Hopenfeld, "Differing Professional Opinion Regarding NRC Approach to Steam Generator Aging," September 25, 1998.

h. Memorandum dated May 20, 1998 from J. Hopenfeld to J. T. Larkins, "New Information Relative to Steam Generator Behavior During Severe Accidents," May 20, 1998.

i. Memorandum dated July 13, 1994, from J. Hopenfeld to J. M. Taylor, "Differing Professional Opinion Regarding Voltage-Based Interim Repair Criteria for Steam Generator Tubes."

3. Memorandum dated April 5, 2000, from Joram Hopenfeld to William Travers, EDO, Subject: Supplement to My DPO Regarding Multiple Steam Generator Leakage (Originally filed as a DPV in December 1991 and filed as a DPO in July 1994).

4. Memorandum dated May 17, 2000, from Jack R. Strosnider, to James T. Wiggins, Subject: Issues Presented in Supplement to Differing Professional Opinion Regarding Steam Generator Tube Integrity.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

LBP-00-19

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

G. Paul Bollwerk, III, Chairman
Frederick J. Shon
Dr. Peter S. Lam

In the Matter of	Docket No. 50-400-LA
CAROLINA POWER & LIGHT COMPANY	ASLBP No. 99-762-02-LA
(Shearon Harris Nuclear Power Plant)	August 7, 2000

MEMORANDUM AND ORDER
(Ruling on Late-Filed Environmental Contentions)

Pending before the Licensing Board is the motion of intervenor Board of Commissioners of Orange County, North Carolina, (BCOC) seeking admission of four late-filed contentions. Each of these issue statements concerns the purported need for the NRC staff to prepare an environmental impact statement (EIS) regarding the pending request of applicant Carolina Power & Light Company (CP&L) for an amendment to its operating license for its Shearon Harris Nuclear Power Plant (Harris) to permit the addition of rack modules to spent fuel pools (SFPs) C and D and to place those pools in service. Although both CP&L and the staff declare that a balancing of the five late-filing elements of 10 C.F.R. § 2.714(a) weighs in favor of admitting the contentions, they nonetheless assert that the contentions should be rejected as lacking adequate basis and specificity as required by section 2.714(b), (d).

For the reasons set forth below, we find that (1) the section 2.714(a) balancing process supports admission of the contentions notwithstanding their "lateness"; and (2) one of the environmental contentions, which we redesignate as Environmental Contention (EC)-6, should

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be admitted, subject to the limitations described herein. Additionally, we establish a schedule for the further litigation of contention EC-6.

I. BACKGROUND

The question of the admission for litigation of the general subject matter of the four late-filed contentions now before the Board first arose in the context of BCOC's initial, timely-filed contentions. In its April 5, 1999 supplement to its February 1999 hearing petition, BCOC proffered five issue statements, which were designated EC-1 through EC-5, challenging CP&L and staff compliance with the requirements of the National Environmental Policy Act of 1969 (NEPA) relative to the applicant's SFP expansion amendment. Among other things, those contentions asserted that the proposed license amendment was not exempt from NEPA's requirements under 10 C.F.R. § 51.22; that an EIS was required that addressed amendment effects on Harris accident probability and consequences and alternative costs and benefits, including severe accident mitigation design alternatives (SAMDAs) and dry cask storage; that the EIS needed to address storage of spent fuel from CP&L's Brunswick and Robinson plants; that an environmental assessment must be conducted; and that a discretionary EIS is required under 10 C.F.R. §§ 51.20(b)(14), 51.22(b). As we described in our July 1999 memorandum and order ruling on the admissibility of those five contentions, as a result of a superseding staff determination to prepare an environmental assessment (EA) relating to the proposed CP&L license amendment, we concluded BCOC's concerns were premature and dismissed those contentions, albeit without prejudice to their being raised at a later juncture, as appropriate. <u>See</u> LBP-99-25, 50 NRC 25, 38-39 (1999).

In that same issuance, we admitted two of BCOC's technical contentions that thereafter were subject to litigation in accordance with the provisions of 10 C.F.R. Part 2, Subpart K. While the parties were preparing for 10 C.F.R. § 2.1113 oral presentations to the Board on the

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issue of whether there were disputed material facts that warranted further exploration in an evidentiary hearing relative to the admitted BCOC technical contentions, the staff provided the Board and the other parties with a Board Notification indicating that on December 15, 1999, it had issued an EA regarding the CP&L amendment request. See Letter from Richard J. Laufer, Project Manager, NRC Office of Nuclear Reactor Regulation to Licensing Board and Parties (Jan. 10, 2000). In its EA, which was published in the Federal Register on December 21, 1999, the staff concluded that an EIS was unnecessary relative to the CP&L spent fuel pool expansion request because it did not involve a proposed action that would have a significant effect on the quality of the human environment. See 64 Fed. Reg. 71,514, 71,516 (1999).

Relative to this EA, on January 31, 2000, BCOC filed the request for admission of four late-filed NEPA-related contentions that is now pending with the Board. In these contentions, which are numbered EC-1 through EC-4, BCOC challenges the staff's EA, asserting that (1) an EIS must be prepared because the proposed Harris SFP expansion would create accident risks substantially in excess of those the staff identified in the EA or previously evaluated in the Harris operating license EIS that would significantly affect the quality of the human environment; (2) the EIS that must be prepared must evaluate the significant cumulative environmental risk posed by the operation of pools A, B, C, and D that was not acknowledged in the EA; (3) the EIS that must be prepared must include within its scope an analysis of the impacts of storage of spent fuel from the Brunswick and Robinson nuclear power plants; and (4) a discretionary EIS is needed. BCOC further asserts that a balancing of the five late-filing elements of 10 C.F.R. § 2.714(a)(1) supports a finding that the timing of its filing should not be a bar to their admission. Additionally, BCOC provides information regarding the grounds for each contention that it declares is sufficient to provide the requisite specificity and basis in accordance with the substantive contention admission standards in section 2.714(b), (d). See

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[BCOC] Request for Admission of Late-Filed Environmental Contentions (Jan. 31, 2000) at 23-27 [hereinafter BCOC Contentions Request].

On March 3, 2000, CP&L and the staff filed responses to the BCOC late-filed request. Both assert that section 2.714(a) late-filing factors three and five -- developing a sound record and broadening or delaying the proceeding -- do not support late-filed admission. In particular, both suggest relative to factor three that BCOC supporting affiant Dr. Gordon Thompson lacks the requisite education, qualifications, and experience to assist the Board in developing a sound record. Neither, however, contests that BCOC has established that the paramount "good cause" factor, along with factors two and four -- availability of other means or parties to protect BCOC's interests -- all weigh in favor of admitting the contentions, thereby tipping the overall balance in favor of a finding that late-filing does not bar admission of the contentions. See [CP&L] Response to BCOC's Late-Filed Environmental Contentions (Mar. 3, 2000) at 1-2 [hereinafter CP&L Contentions Response]; NRC Staff Response to [BCOC] Request for Admission of Late-Filed Environmental Contentions (Mar. 3, 2000) at 1-4 [hereinafter Staff Contentions Response].

What CP&L and the staff do dispute is BCOC's claim that the contentions fulfill the pleading requirements of section 2.714, asserting for various reasons that each of the contentions lacks the requisite specificity and basis. See CP&L Contention Response at 7-29; Staff Contention Response at 7-29. In a March 13, 2000 reply to the CP&L and staff responses, BCOC challenges their claims regarding the adequacy of Dr. Thompson's qualifications relative to late-filing factor three as well as their assertions concerning the adequacy of the four contentions. See [BCOC] Reply to [CP&L's] and Staff's Oppositions to Request for Admission of Late-Filed Environmental Contentions (Mar. 13, 2000) at 1-22 [hereinafter BCOC Contentions Reply].

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Subsequently, it came to the Board's attention that there was outstanding on the public record a recent draft staff technical study concerning spent fuel pool accident risks, see 65 Fed. Reg. 8752 (2000) (soliciting public comment on draft report), which was one of the matters that was of concern to BCOC in the context of its contention denominated as EC-1, Environmental Impact Statement Required. Although recognizing that this staff report dealt with spent fuel pool accident risks associated with facility decommissioning activities, the Board provided the parties with an opportunity to provide their views, and respond to the views of the other parties, on the relevance, if any, of this study to the issues before the Board. See Memorandum and Order (Requesting Additional Information) (Mar. 21, 2000) at 1-2 (unpublished). Thereafter, all three of the parties filed comments regarding the draft staff report. BCOC asserted that although the study's limited scope -- i.e., decommissioning -- restricted its relevance, the staff's technical analysis still was pertinent in that it (1) further illustrates how the staff has underestimated the risks of SFP accidents because that study does not include an assessment of the phenomena associated with partial exposure of fuel assemblies, a subject that is at the center of Dr. Thompson's concerns about the SFP accident risks; (2) fails to consider the effect of fuel age on potential for propagation of exothermic reactions; (3) does not discuss criticality accident risk from the placement of low-burnup fuel in a pool in which there is reliance on burnup credit to prevent criticality; and (4) lacks sufficient information regarding zirconium fire propagation. See [BCOC] Response to Board's Information Request (Mar. 29, 2000) at 2-10; see also [BCOC] Reply to [CP&L's] and Staff's Responses to Board's Information Request (Apr. 5, 2000) at 2-7. Both CP&L and the staff, on the other hand, found the draft report basically irrelevant to the admission of the contention because it concerns a decommissioned reactor rather than an operating reactor like Harris, although each found points in the draft report, such as the availability and timing of pool water makeup, that supported its position that BCOC contention EC-1 was not admissible. See [CP&L] Response to Board's Request

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Regarding Relevance of Staff's Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Plants (Mar. 29, 2000) at 2-6; NRC Staff Response to the Atomic Safety and Licensing Board's Request for Additional Information (Mar. 29, 2000) at 2-5; see also CP&L Reply to Parties' Responses Regarding Relevance of Staff's Draft Decommissioning Study (Apr. 5, 2000) at 2-3; NRC Staff's Reply to [BCOC] Response to the Board's Request for Additional Information (Apr. 5, 2000) at 2-5.

Thereafter, by order dated May 5, 2000, the Board again requested information from the parties in connection with the draft staff report, prompted by an April 13, 2000 public record letter from Advisory Committee on Reactor Safeguards (ACRS) Chairman Dana A. Powers to NRC Chairman Richard A. Meserve providing ACRS views on the draft staff report, including concerns about the potential for exothermic reactions in the event a pool is drained and the resulting release of ruthenium, as a source term element. See Licensing Board Memorandum and Order (Requesting Additional Information) (May 5, 2000) at 1-2 (unpublished). In its May 15, 2000 response, BCOC found this letter reinforced its contention EC-1 claim that spent fuel pool accident risks are greater than the staff assumes because the staff does not understand the potential for SFP exothermic reactions. See [BCOC] Response to May 5, 2000, Memorandum and Order (Requesting Additional Information) (May 15, 2000) at 1-4. In their May 15 responses, CP&L and the staff maintained that, like the staff draft report, the ACRS letter is irrelevant because it deals with a decommissioned facility, not an operating reactor like Harris. See [CP&L] Response to Board's Request Regarding Relevance of ACRS Letter Addressing NRC Staff Draft Decommissioning Study (May 15, 2000) at 1-3; NRC Staff Response to the Atomic Safety and Licensing Board's Second Request for Additional Information (May 15, 2000) at 2-3; see also [CP&L] Reply to Parties' Responses Regarding Relevance of ACRS Letter Addressing NRC Staff Draft Decommissioning Report (May 22,

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2000) at 2-5; NRC Staff Reply to [BCOC] Response to May 5, 2000, Memorandum and Order (Requesting Additional Information) (May 22, 2000) at 1-2.

Finally, in response to a July 12, 2000 BCOC motion, on July 13, 2000, the Board granted leave for the parties to comment on a June 20, 2000 letter from ACRS Chairman Powers to NRC Chairman Meserve concerning the proposed resolution of outstanding Generic Safety Issue (GSI)-173A, regarding an action plan for resolving issues relating to operating reactor SFPs. See Licensing Board Memorandum and Order (Granting Motion for Leave to Comment) (July 13, 2000) at 1-2 (unpublished). BCOC took the position that, as with the ACRS comments on the staff decommissioning study, this letter was relevant to its accident risk contention, particularly as it concerns SFP radiological inventories and release characteristics. See [BCOC] Comments on Relevance of June 20, 2000, ACRS Letter with Respect to Pending Environmental Contentions (July 20, 2000) at 3-4. In their comments on the ACRS letter, both CP&L and the staff asserted that this ACRS letter had no relevance to the BCOC contentions because, as with the previous ACRS letter, it does not concern that specific beyond-design-basis reactor accident scenario that is the underpinning for the BCOC accident risk contention. See [CP&L] Comments on Relevance of June ACRS Letter to Pending Environmental Contentions (July 20, 2000) at 3-8; NRC Staff Comments on [ACRS] Letter of June 20, 2000 (July 20, 2000) at 2-3; see also [CP&L] Reply to Parties' Comments on Relevance of June ACRS Letter to Pending Environmental Contentions (July 27, 2000) at 2-4.

II. ANALYSIS

All the parties recognize that the five late-filing factors set forth in 10 C.F.R. § 2.714(a)(1) are applicable to BCOC's four pending environmental contentions. And, relative to such late-filed contentions, it is well-established that the burden rests with the petitioner, here BCOC, to address affirmatively all five factors and demonstrate that, on balance, they warrant

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excusing the lateness of the filing. Moreover, even if a late-filed contention fulfills the section 2.714(a)(1) requirements, it must still satisfy the admissibility standards set forth in section 2.714(b)(2)(i)-(iii), (d)(2), in order to receive merits consideration. See, e.g., Private Fuel Storage, L.L.C. (Independent Spent Fuel Storage Installation), LBP-99-43, 50 NRC 306, 312 (1999) (citing cases), petition for interlocutory review denied, CLI-00-2, 51 NRC 77 (2000).

A. Application of 10 C.F.R. § 2.714(a)(1) Late-Filing Criteria

It is, of course, also well-established that the first factor -- whether there is "good cause" for the failure to file on time -- is the most important component in the late-filed balancing equation. The BCOC environmental contentions now at issue were not filed until some nine months after contentions were due in this proceeding. Nonetheless, section 2.714(b)(2)(iii) recognizes that a petitioner can file amended or new contentions "if there are data or conclusions in the NRC draft or final environmental impact statement, environmental assessment, or any supplements related thereto, that differ significantly from the data or conclusions in the applicant's [environmental report]." Here, the crux of BCOC's concerns, as expressed in its January 2000 contentions, is that the staff erred in its December 1999 EA in concluding that no EIS is needed. As both CP&L and the staff acknowledge, there is good cause for such a "late-filed" challenge, assuming the contentions involved are filed within a reasonable time after BCOC became, or should have become, aware of the staff EA.

In this instance, BCOC's late-filed contentions pleading was submitted some forty-five days after the EA was first provided to BCOC counsel by fax from the staff. BCOC declares that this period for filing was reasonable given that BCOC counsel (1) until January 4, 2000, was involved in preparing its 10 C.F.R. § 2.1113 written presentation regarding the two admitted technical contentions; (2) between January 8 and January 17, was on previously scheduled, ten-day overseas non-vacation trip; (3) between January 17 and January 21, was involved in preparing for and participating in the oral argument regarding that filing, which was

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held during an all-day session on January 21, 2000; and (4) between January 24 and January 31, was working on two other cases, and was out of her Washington, D.C. office on one day and was unable to reach her client on two days because of inclement weather. See BCOC Contentions Request at 23-25. Neither CP&L nor the staff dispute that, under the circumstances, the "good cause" element of the section 2.714(a)(1) test has been fulfilled such that this factor favors admitting the contentions. We agree, and thus place this central factor on the "acceptance" side of the balance.

Relative to the other four factors, we also agree with the parties that factors two and four -- availability of other means to protect petitioner's interests and extent of representation of petitioner's interests by other parties -- weigh in BCOC's favor. As to factors three and five, which among the four non-good cause elements are given more weight in the balancing process, see Commonwealth Edison Co. (Braidwood Nuclear Power Station, Units 1 and 2), CLI-86-8, 23 NRC 241, 244-45 (1986), both are problematic in terms of their impact on the balance. Given our May 2000 ruling in favor of CP&L on the two technical issues we admitted for merits consideration, see LBP-00-12, 51 NRC 247, petition for review denied as interlocutory, CLI-00-11, 51 NRC __ (June 20, 2000), the admission of any of these environmental contentions undoubtedly will broaden the issues and delay the proceeding. Moreover, relative to element three -- assistance in developing a sound record -- our observation in our May 2000 decision that Dr. Thompson's expertise on reactor technical issues appeared to be "largely policy-oriented rather than operational" does not render this a compelling element on BCOC's side of the balance. Nonetheless, in the circumstances here, these two negative elements are not sufficient to overcome the combined weight of factors one, two, and four as supporting a finding that the late-filing of these contentions does not bar their admission.

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B. Application of 10 C.F.R. § 2.714(b), (d) Admissibility Criteria

In determining whether the four BCOC environmental contentions are admissible in accordance with the standards set forth in section 2.714(b) and (d), we note initially that we previously dismissed contentions denominated as EC-1 through EC-5 in our July 1999 ruling on BCOC's standing and the admissibility of its timely filed contentions. Three of the four BCOC late-filed contentions essentially track these issues, albeit with different numbers in two instances.[1] For the sake of clarity, in considering these four late-filed contentions we have renumbered them to continue the numbering sequence begun with the already-rejected environmental contentions. And below, we discuss the admissibility of each, beginning with renumbered contention EC-6.

1. CONTENTION EC-[6]: Environmental Impact Statement Required

> In the Environmental Assessment ("EA") for CP&L's December 23, 1998, license amendment application, the NRC Staff concludes that the proposed expansion of spent fuel storage capacity at the Shearon Harris nuclear power plant will not have a significant effect on the quality of the human environment. Environmental Assessment and Finding of No Significant Impact Related to Expanding the Spent Fuel Pool Stage Capacity at the Shearon Harris Nuclear Power Plant (TAC No. MA4432) at 10 (December 15, 2000). Therefore, the Staff has decided not to prepare an Environmental Impact Statement ("EIS") for the proposed license amendment. The Staff's decision not to prepare an EIS violates the National Environmental Policy Act ("NEPA") and NRC's implementing regulations, because the Finding of No Significant Impact ("FONSI") is erroneous and arbitrary and capricious. In fact, the proposed expansion of spent fuel pool storage capacity at Harris would create accident risks that are significantly in excess of the risks identified in the EA, and significantly in excess of accident risks previously evaluated by the NRC Staff in the EIS for the Harris operating license. These accident risks would significantly affect the quality of the human environment, and therefore must be addressed in an EIS.

[1] Originally-filed contention EC-2 corresponds to late-filed contention EC-1 and the previously submitted contention EC-5 corresponds to late-filed contention EC-4.

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> There are two respects in which the proposed license amendment would significantly increase the risk of an accident at Harris:
>
> (1) CP&L proposes several substantial changes in the physical characteristics and mode of operation of the Harris plant. The effects of these changes on the accident risk posed by the Harris plant have not been accounted for in the Staff's EA. The changes would significantly increase, above present levels, the probability and consequences of potential accidents at the Harris plant.
>
> (2) During the period since the publication in 1979 of NUREG-0575, the NRC's Generic Environmental Impact Statement ("GEIS") on spent fuel storage[1], new information has become available regarding the risks of storing spent fuel in pools. This information shows that the proposed license amendment would significantly increase the probability and consequences of potential accidents at the Harris plant, above the levels indicated in the GEIS, the 1983 EIS for the Harris operating license, and the EA. The new information is not addressed in the EA or the 1983 EIS for the Harris operating license.
>
> Accordingly, the Staff must prepare an EIS that fully considers the environmental impacts of the proposed license amendment, including its effects on the probability and consequences of accidents at the Harris plant. As required by NEPA and Commission policy, the EIS should also examine the costs and benefits of the proposed action in comparison to various alternatives, including Severe Accident Mitigation Design Alternatives ("SAMDAs")and the alternative of dry storage.
>
> ---
>
> [1] NUREG-0575, Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (August 1979) (hereinafter "GEIS").

BCOC Contentions Request at 1-2.

DISCUSSION: Id. at 1-16; CP&L Contentions Response at 7-20; Staff Contentions Response at 7-26; BCOC Contentions Reply at 8-19.

RULING: With this contention, BCOC challenges the staff's EA conclusion that the proposed CP&L license amendment to use spent fuel pools C and D does not require a complete EIS. In assessing the basis for this contention, we note that all three parties agree

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that the standard for requiring that an EIS be prepared is whether the action at issue, in this case the CP&L license amendment, is a major federal action having a significant impact on the human environment. See BCOC Contentions Request at 3; CP&L Contentions Response at 3 n.3; Staff Contentions Response at 8. Further, all the parties agree that the agency is not required to address in an EIS consequences of an action that are "remote and speculative." See CP&L Contentions Response at 9-10; Staff Contentions Response at 16; BCOC Contentions Reply at 8. What the parties disagree about is whether a possible consequence of the action identified by BCOC -- a severe accident in spent fuel pools C and D -- is remote and speculative.

BCOC discusses a number of different elements that it asserts provide the basis for this contention, including the fact that the number of stored spent fuel assemblies at the Harris facility ultimately may double as a result of the proposed amendment; the purported impact of the use of "administrative measures" such as controlling fuel burnup levels rather than relying solely on "physical measures" such as fuel assembly separation and the presence of solid neutron absorbers to avoid criticality; and new information regarding sabotage risk. In the Board's view, however, the crux of the contention, and the focus of our consideration as to whether it meets the specificity and basis requirements of section 2.714, is whether the accident proposed by BCOC in basis F.1 of the contention has a probability sufficient to provide the beyond-remote-and-speculative "trigger" that is needed to compel preparation of an EIS relative to this proposed licensing action.

To examine whether the contention provides an adequate basis to support further Board consideration of this question, we examine the accident scenario in question, which was first summarized by CP&L, see CP&L Contentions Response at 9-10, with an appropriate modification by BCOC, see BCOC Contentions Reply at 8. In this regard, BCOC postulates the following chain of events:

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(1) a degraded core accident;
(2) containment failure or bypass;
(3) loss of all spent fuel cooling and makeup systems;
(4) extreme radiation doses precluding personnel access;
(5) inability to restart any pool cooling or makeup systems due to extreme radiation doses;
(6) loss of most or all pool water through evaporation; and
(7) initiation of an exothermic oxidation reaction in pools C and D.

Relative to this accident sequence, what BCOC asserts, and what the CP&L and the staff contest, is that BCOC has established an adequate basis to allow merits litigation on whether this sequence is not "remote and speculative" so that a further environmental analysis of the CP&L pool expansion amendment request is required.

In considering this question, we note that the Commission has provided some guidance regarding such an issue statement in its decision in Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), CLI-90-4, 31 NRC 333 (1990). In that case, which also involved the expansion of a spent fuel pool, likewise at issue was the admission of a contention that asserted the license amendment involved required the preparation of an environmental impact statement because the action raised the potential for a substantial release of radioactive material following the occurrence of a specific accident sequence. More specifically, the question in dispute was whether the accident sequence specified was of a sufficiently high probability to put it beyond the "remote and speculative" threshold for the purpose of admitting the contention.

Prior to coming before the Commission, however, that contention was considered by both the Licensing Board and the Appeal Board, with the matter coming before the Appeal Board on referral from the Licensing Board's admission of the contention. See Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), ALAB-919, 30 NRC 29 (1989). The Appeal Board determined that:

> The essence of Environmental Contention 1 . . . is that an environmental impact statement is required for the proposed

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> license amendment to assess the risks of the following hypothetical accident scenario: (1) a severe reactor accident occurs by some unidentified mechanism and involves substantial fuel damage, hydrogen generation, Mark I containment failure, and subsequent detonation in the reactor building where the Vermont Yankee fuel pool is located; (2) the reactor building and the spent fuel pool are assuredly not likely to withstand the pressure and temperature loads generated by such an accident, thereby threatening the pool cooling systems or the pool structure itself, . . . and (3) pool heatup occurs, resulting in a self-sustaining zircaloy cladding fire with increased long-term health effects for the public from the increased fuel pool inventory.

Id. at 43 (citations omitted). The Appeal Board then went on to say that the scenario on which the contention is premised "is obviously not a 'normal' operating event; indeed it can be characterized as a double 'worst case' accident." Id. Consequently, after what it considered to be a careful examination of the bases presented for the accident scenario, the Appeal Board rejected the contention and referred its ruling to the Commission. See id. at 52.

The Commission responded by remanding the issue to the Appeal Board for further consideration, saying:

> The Commission believes that on remand more information on the plausibility or probability of the reactor accident/hydrogen combustion/spent fuel pool cooling failure/cladding fire at issue here . . . is needed before a judgment should be made whether the accident . . . is remote and speculative. As part of our remand we therefore direct the Appeal Board to develop such information further. We leave it to the Appeal Board to decide the procedural means to obtain this information, whether by inviting something akin to summary disposition motions or otherwise. If the Appeal Board finds that an accident probability on the order of 10^{-4} per year is appropriate for the entire accident sequence postulated in this contention, the case should be returned to the Commission for further review. Otherwise, the Appeal Board should modify or confirm its judgment as to the remote and speculative nature of the accident on the basis of the accident probability derived on remand.

CLI-90-04, 31 NRC at 335-36 (citations omitted).

There followed an Appeal Board request for clarification of the Commission's decision. See ALAB-938, 32 NRC 154 (1990). But before the Commission could respond, the

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intervenors asked to withdraw from the proceeding and the licensee moved to dismiss the proceeding. The Commission granted the motion to dismiss, but opined that it was

> concerned that the probability that the Appeal Board found to be so low as to be remote and speculative pertained not to the whole scenario in the contention but to pieces of the scenario in the contention or related scenarios set out in the technical documents, some with probabilities as high as on the order of 10^{-4} per reactor year. In ALAB-919, the Appeal Board bridged the gap between the technical documents and the scenario in the contention by assuming, conservatively, that the probability of that scenario could be no greater than certain scenarios actually analyzed in the documents. If the scenarios in the documents were remote and speculative, then, *a fortiori*, the scenario in the contention must be remote and speculative as well. Our opinion makes clear that future decisions that accident scenarios are remote and speculative must be more specific and more soundly based on the actual probabilities and accident scenarios being analyzed.

CLI-90-7, 32 NRC 129, 132 (footnote omitted).

Certainly, in the intervening decade the Commission has come to rely on probabilistic analysis ever more heavily in the process of making decisions. Indeed, the entire trend in licensing, enforcement, inspection and the granting of amendments has swung gradually toward decision-making by probabilistic risk assessment. We therefore think that the Commission's intent is at present even more firmly directed to deciding what is "remote and speculative" by examining the probabilities inherent in a proposed accident scenario.

In this instance, based on the information now presented by BCOC, including the 1993 Harris facility individual plant evaluation (IPE) of core damage frequency, the accident scenario it has postulated may have a probability in the range of 1×10^{-5} per reactor year, *see* BCOC Contentions Reply at 11-12, a figure that under the Commission's guidance seemingly should not be dismissed automatically as per se "remote and speculative." To be sure, CP&L and the staff dispute various aspects of the BCOC probability analysis and its underlying accident scenario, including whether cooling water restoration would be precluded by onsite radiation

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levels; the availability of water makeup systems; bounding decay heat levels for pools C and D; the age of the spent fuel that will be stored in pools C and D; whether the probability of a substantial SFP release is on a par with the probability of a substantial reactor release; the effect of the use of burnup credit; and an increase in sabotage-related risk. And we agree with CP&L and the staff that BCOC's assertions regarding sabotage risk do not provide a litigable basis for this contention. See Philadelphia Electric Co. (Limerick Generating Station, Units 1 and 2), ALAB-819, 22 NRC 681, 701 (1985), review declined, CLI-86-5, 23 NRC 125 (1986), aff'd, Limerick Ecology Action v. NRC, 869 F.2d 719, 744 (3d Cir. 1989). We find, however, that the information provided by BCOC otherwise is sufficient to establish a genuine material dispute of fact or law adequate to warrant further inquiry relative to the other aspects of the BCOC scenario and the associated probability analysis.[2] Accordingly, we admit contention EC-6 as it relates to this accident sequence.[3]

Finally, in connection with further litigation on this contention, we offer the following additional observations. In its Vermont Yankee decision, the Commission directed the Appeal Board to select a "procedural means" to obtain the risk-informed information and suggested "something akin" to inviting summary disposition motions. CLI-90-04, 31 NRC at 336. In this instance, pursuant to 10 C.F.R. § 2.1109, CP&L has invoked the process set forth in Subpart K

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[2] In this regard, we note that in our decision in LBP-00-12, 51 NRC at 259-60, in ruling on the two admitted BCOC technical contentions, we found CP&L's planned use of so-called "administrative processes," such as use of enrichment/burnup level controls and soluble boron as SFP criticality control measures, is permitted under General Design Criteria (GDC) 62. As a consequence, contrary to BCOC's assertion, the use of such measures does not, in and of itself, trigger the need for an EIS. Whether, and to what extent, the use of these control measures has any relevance to the probability calculation at issue here is a matter for resolution as part of further litigation regarding contention EC-6. The same is true for the question of the heat load for pools C and D, which seemingly includes an associated legal issue concerning appropriate project segmentation relative to NEPA.

[3] In its final sentence, the contention includes a statement about what should be analyzed in an EIS. For the reasons stated below relative to contentions EC-7 and EC-8, we consider this aspect of the contention premature and do not admit it.

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to Part 2 that includes the written summaries and oral argument specified in sections 2.1109 and 2.1113. Certainly, these procedures are sufficiently "akin" to summary disposition to satisfy the Commission's previously stated preference.

Additionally, so that we will be able properly to assess the significance of the materials submitted in the detailed written summaries required by section 2.1113(a), we ask that the parties address the following points:

1. What is the submitting party's best estimate of the overall probability of the sequence set forth in the chain of seven events in the CP&L and BCOC's filings, set forth on page 13 supra? The estimates should utilize plant-specific data where available and should utilize the best available generic data where generic data is relied upon.

2. The parties should take careful note of any recent developments in the estimation of the probabilities of the individual events in the sequence at issue. In particular, have new data or models suggested any modification of the estimate of 2×10^{-6} per year set forth in the executive summary of NUREG-1353, Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools (1989)? Further, do any of the concerns expressed in the ACRS's April 13, 2000 letter suggest that the probabilities of individual elements of the sequence are greater than those previously analyzed (e.g., is the chance of occurrence of sequence element seven, an exothermic reaction, greater than was assumed in the decade-old NUREG-1353)?

3. Assuming the Board should decide that the probability involved is of sufficient moment so as not to permit the postulated accident sequence to be classified as "remote and speculative," what would be the overall scope of the environmental impact analysis the staff would be required to prepare (i.e, limited to the impacts of that accident sequence or a full blown EIS regarding the amendment request)?

2. Contention EC-[7]: EIS Should Consider Cumulative Impacts In Light of New Information

> The EA is deficient because it fails to acknowledge or evaluate the significant cumulative environmental risk posed by the operation of pools A, B, C, and D.

BCOC Contentions Request at 16.

DISCUSSION: Id. at 17-18; CP&L Contentions Response at 20-25; Staff Contentions Response at 26-27; BCOC Contentions Reply at 20-21.

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RULING: We find this contention premature, given that there is still an outstanding question whether the staff correctly concluded in its EA that no environmental impact statement is required. See LBP-99-25, 50 NRC at 39. If, in ruling on the merits of contention EC-6, we should determine that an EIS is necessary, then the proper scope of that EIS would become a matter in controversy based on the CP&L environmental report (assuming the staff requires that one be prepared) and the EIS the staff prepares.

3. Contention EC-[8]: Scope of EIS Should Include Brunswick and Robinson Storage

> The EIS for the proposed license amendment should include within its scope the storage of spent fuel from the Brunswick and Robinson nuclear power plants.

BCOC Contentions Request at 18.

DISCUSSION: Id. at 18-19; CP&L Contentions Response at 25-28; Staff Contentions Response at 27-28; BCOC Contentions Reply at 21.

RULING: As with contention EC-7, we decline to admit this contention as premature.

4. Contention EC-[9]: Discretionary EIS Warranted

> Even if the Licensing Board determines that an EIS is not required under NEPA and 10 C.F.R. § 51.20(a), the Board should nevertheless require an EIS as an exercise of its discretion, as permitted by 10 C.F.R. §§ 51.20(b)(14) and 51.22(b).

BCOC Contentions Request at 20.

DISCUSSION: Id. at 20-23; CP&L Contentions Response at 28-30; Staff Contentions Response at 28-29; BCOC Contentions Reply at 21-22.

RULING: We have carefully considered whether such a discretionary EIS is warranted and we see no reason to require an EIS if one is not required by the rules. We recognize that CP&L and the staff assert that such a requirement is ultra vires for this Board. See CP&L Contentions Response at 28; Staff Contentions Response at 28. We, however, need not rule

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on that point. Suffice it to say that we find no "special circumstances" pursuant to sections 51.20(b)(14) and 51.22(b) that would warrant a discretionary EIS.

III. ADMINISTRATIVE MATTERS

As we previously noted, under 10 C.F.R. § 2.1109, CP&L has invoked the procedural provisions of Part 2, Subpart K, relative to the litigation of this proceeding. Accordingly, the schedule for utilizing the Subpart K procedures in connection with contention EC-5 is as follows:

Discovery Begins	Monday, August 21, 2000
Discovery Ends	Friday, October 20, 2000
Written Summaries Filed	Monday, November 20, 2000

The discovery limitations and guidelines set forth in our July 29, 1999 issuance shall apply.[4] See Licensing Board Memorandum and Order (Granting Request to Invoke 10 C.F.R. Part 2, Subpart K Procedures and Establishing Schedule) (July 29, 2000) at 3-4. Moreover, the Board will establish a date and location for conducting oral argument regarding the parties' written summaries in a subsequent order.

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[4] As with the admitted technical contentions, the Board is not requiring that informal discovery must be used during the discovery period. Nonetheless, the Board notes that the parties need not await the beginning of the discovery period to initiate discussions regarding the nature and scope of the information each will be seeking in discovery and try to reach some agreement on documentary or other materials that can be provided without a formal discovery request.

Also, in connection with discovery in this proceeding, the Board notes that any attempt to obtain discovery materials or testimony from ACRS members, staff, or consultants is subject to the exceptional circumstances showing of 10 C.F.R. § 2.720(h). See Pacific Gas & Electric Co. (Dabble Canyon Nuclear Power Plant, Units 1 and 2), ALAB-519, 9 NRC 42, 43 n.2 (1979). Moreover, the Board directs that any discovery requests regarding ACRS information or personnel must be filed within the first ten days of the discovery period established above.

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IV. CONCLUSION

With these new proposed environmental contentions being filed within forty-five days of the challenged staff EA, the five-factor balancing test set forth in 10 C.F.R. § 2.714(a)(1) favors the admission of BCOC renumbered late-filed contentions EC-6 through EC-8. Additionally, we find that BCOC has established relative to contention EC-6 regarding "remote and speculative" SFP accident sequences that there exists a genuine material dispute of fact or law adequate to warrant further inquiry. We thus admit contention EC-6 and establish a schedule for its further litigation under 10 C.F.R. Part 2, Subpart K. On the other hand, we dismiss contentions EC-7 and EC-8, which concern the scope of any staff EIS that may be needed, as premature, and dismiss contention EC-9, which concerns the need for a discretionary EIS, as lacking adequate support to show there exists a genuine material dispute of fact or law adequate to warrant further inquiry.

For the foregoing reasons, it is this seventh day of August 2000, ORDERED that:

1. The following BCOC contention is admitted for litigation in this proceeding: EC-6.

2. The following BCOC contentions are rejected as inadmissible for litigation in this proceeding: EC-7, EC-8, and EC-9.

3. The parties are to conduct discovery and submit section 2.1113 written presentations in accordance with the schedule established in section III above.

FOR THE ATOMIC SAFETY
AND LICENSING BOARD[5]

Original Signed By
G. Paul Bollwerk, III
ADMINISTRATIVE JUDGE

Original Signed By
Frederick J. Shon
ADMINISTRATIVE JUDGE

Original Signed By
Dr. Peter S. Lam
ADMINISTRATIVE JUDGE

Rockville, Maryland

August 7, 2000

[5] Copies of this memorandum and order were sent this date by Internet e-mail transmission to counsel for (1) applicant CP&L; (2) intervenor BCOC; and (3) the staff.

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

August 21, 2000

MEMORANDUM TO: ACRS Members

FROM: P. Boehnert, Senior Staff Engineer PB

SUBJECT: STATUS OF NRC CORE POWER UPRATE REVIEWS

The purpose of this memorandum is to summarize the status of nuclear power plant licensee applications for core power uprates in excess of 5% of nominal. At the time of its review of the GE Nuclear Energy (GENE)/Fermi Unit 2 (lead-plant) request for a 5% core power uprate (September 1992), pursuant to the GENE generic power uprate program, the Committee stated that it did not wish to review any further BWR power uprate requests unless they exceed 5% of nominal power.

GENE has subsequently initiated a so-called Extended Power Uprate Program which encompasses core power uprates of up to 20% of nominal. Several licensees have indicated strong interest in this Program, and, as noted below, application requests for significant power uprates are to be tendered.

Power Uprate Requests

The following core power uprate requests are of interest to the ACRS:

- Duane Arnold - The Duane Arnold plant (BWR/3, 1658 MWt) licensee (Alliant Energy - AE) has proposed a power uprate program under the GENE Extended Power Uprate Program. AE will be making a submittal for a 15.3% increase in core power. The plant also obtained a 4.7% power uprate, soon after obtaining its operating license, so the overall result is an increase of nearly 20% for the original licensed power level.

  NRR and AE met on May 16, 2000 to discuss this uprate program. A copy of my summary of the meeting is attached. AE plans to make its application submittal in October of this year and is requesting staff approval by May 2001. The staff indicated that this schedule was quite aggressive and approval by May appeared

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in doubt. During the May 16 meeting, I indicated that the ACRS will want to review this submittal as well; AE gave indication that it had not planned on Committee review of this matter.

- **Commonwealth Edison Plants** - Commonwealth Edison plans to submit an application for core power uprates for the Dresden Units 2 & 3 (BWR/3 2527 MWT , each) and Quad Cities 1 & 2 (BWR/3 2511 MWt, each) plants. Commonwealth Edison is seeking a power uprate of 17% for each of the four units. The licensee plans to submit its application at the end of this year, and is requesting approval for the lead unit (Dresden, Unit 2) by November 2001.

  ACRS Fellow G. Chronenberg and I attended a NRR/Commonwealth Edison "kick off" meeting held to discuss the uprate application. A copy of my meeting summary is attached. Aside from the uprate, the licensee plans to transition all four units to a new fuel design - GE14. Key items discussed at the meeting included: concern with adopting the GE critical power ratio methodology to calculation of mixed fuel types, including Siemens fuel; the fact that all four units had been uprated in power by 5% shortly after initial licensing which equates to a 22% power uprate, 2% more than the 20% "limit" specified in the GENE Extended Uprate Program; and, the need for ACRS review of this uprate application.

- **Beaver Valley Plants** - Recently, representatives of the First Energy Nuclear Operating Company (Beaver Valley plant licensee) met with NRR to discuss its so-called "Full Potential Program" for the Beaver Valley plants (W - 3 loop, 2652 MWt, each). The Program includes a total power uprate of 6.4%.[1] Other elements of the Full Potential Program include replacement of Unit 1 steam generators, conversion from a sub-atmospheric to atmospheric containment, use of revised source term, and license renewal (2004 application). It was also noted that FENOC intends to make use of the EPRI MAAP code in conjunction with their source term submittal.

  ACRS Fellow G. Chronenberg attended an August 8, 2000 FENOC/NRR meeting. A copy of his report on this matter is attached for your perusal.

- **Other Uprate Applications** - Aside from the above plants, significant interest has been expressed by the Grand Gulf and Brunswick plant licensees. During the meeting discussions with the Commonwealth Edison representatives, it was stated that the cost of the uprated power was ~ $175/KWe. If this cost is typical for BWR

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[1] The uprate will proceed in two installments: a 1.4% uprate application to be submitted this year pursuant to the recent revision to Appendix K to allow use of more accurate flow instrumentation, and a separate 5% power uprate to be submitted in late-2002.

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units, one can expect a flood of uprate applications under the GENE Extended Uprate Program.

ACRS Concerns Related to Uprate Reviews - G. Chronenberg has raised some issues regarding the staff's reviews of power uprate applications. He presented a paper on this matter at the ANS meeting of June 4-8, 2000 held in San Diego, entitled: "Potential Synergistic Safety Issues Related to Reactor Power Uprate Reviews". A copy of Gus's travel report on the San Diego Meeting which includes a recounting of the "Q & A" associated with his presentation and a reprint of his Paper is attached. Gus has also issued a June 20, 2000 memorandum that was provided to the Committee during the July Meeting that details his concerns. These concerns include the potential for erosion of safety margins/emergency measures, the lack of a Standard Review Plan Section addressing power uprates, and, an apparent lack of adequate staff audits of licensee uprate submittal information/less than rigorous review of same.

ACRS Members were requested to provide comment on Gus's June 20 Memorandum, pursuant to Committee action on this matter. The Planning and Procedures Subcommittee is scheduled to discuss the issue of core power uprate applications during its August 28, 2000 meeting and report to the Committee during the September Meeting.

Attachments: As Stated

cc: Balance of ACRS Members
R. Savio

cc w/o attach (via E-mail):
J. Larkins
H. Larson
S. Duraiswamy
ACRS Technical Staff & Fellows

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

June 23, 2000

MEMORANDUM TO: ACRS Members and Staff

MEMORANDUM #: AWC-106.2000

FROM: A. W. Cronenberg

SUBJECT: Central Issues Related to Power Uprate Reviews

This memo outlines key issues that I believe should be addressed in anticipated ACRS meetings with the staff, related to the conduct/content of agency reviews of License Amendment Requests for power uprates. Recommendations stem from my prior review of operational events noted for uprated plants, and indications of potential synergistic safety implications for aged/uprated plants which involve an extended fuel cycle.

1) Adequacy Of Agency Uprate Review Procedures: The agency's Maine Yankee Lessons Learned effort (*Report of the Maine Yankee Lessons Learned Task Group*, internal NRC document, 1996) indicated the need for a more comprehensive/consistent review of power uprate applications, with a primary recommendation for development of a Uprate Standard Review Plan. A similar recommendation was made by an independent review of power uprates by Scientech Inc, (J. S. Miller, *Power Uprate* Review, Scientech. Inc, SCIE-NRC-249-96, Oct. 1996). My observations lead to a similar recommendation of the need for a more formalized approach to review of power uprate requests. My examination of agency uprate Safety Evaluation Reports (SERs) does not reveal consistency in the scope and level of detail of the subject matter reviewed. The SERs do not generally specify how the review was accomplished, the acceptance criteria for the conclusions reached, or include staff analysis to audit the accuracy of information provided by the licensee. This type of information would normally be expected under stipulations of a Standard Review Plan.

Agency in-action for a more comprehensive uprate review process is being justified by risk arguments of minor changes in CDF for power uprates. This indeed may be the case, nevertheless operational events have been noted for uprated plants, as well as violations of Tech. Specs. In light of these considerations, I recommend that ACRS encourage a more formalized approach to review of power uprates, specifically the development of a Uprate Standard Review Plan.

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<u>2) NRC Audit Analysis of Licensee Submittal Information</u>: A large part of a licensee's submittal for a power uprate centers on a re-analysis of information similar to that found in the original FSAR but at the higher power level; examples being a re-evaluation of design basis accidents (DBA) and off-normal transients at the elevated power, operational core cooling and core thermal-hydraulic conditions, DNB (departure from nucleate boiling) margins, analysis of the thermal capacity of the residual heat removal and emergency core cooling systems. Balance-of-plant thermal-hydraulic analysis must also be provided by the licensee, such as predictions of secondary-side feedwater flow/temperature conditions at the increased thermal load. This information generally takes the form of code predictions which are reviewed by the NRC staff and findings reported in the uprate Safety Evaluation Report (SER).

A review of a number of uprate SERs (i.e. Brunswick, Limerick, Maine Yankee, North Anna, Surry, Callaway, and Wolf Creek plants) reveals little in the way of staff audit analysis of licensee submittal predictions. The question then is how can the staff validate the accuracy of submittal analysis without aid of it's own/independent audit calculations. If this had been done for Maine Yankee, the faulty LOCA analysis might have been reveled by the staff rather than from whistle-blower accusations. I strongly urge the ACRS to press the staff for some sort of audit analysis of Licensee submittal thermal-hydraulic predictions. I would also urge additional staff audit analysis of core neutronics predictions by the licensee, specifically when an uprate request involves use of extended fuel duty times and/or for cores involving new fuel configurations, e.g new fuel designs or reload configurations with a mixture of multi-vendor fuel types (see INPO report on problems noted for restart with multi-vendor fuel; *Design and Operating Considerations for Reactor Cores,* <u>SEOR-96-2</u>, 1996).

<u>3) Potential Synergistic Safety Issues:</u> Several recent operational events for uprated plants point to circumstantial evidence of compounding degradation due aging/uprate and high-burnup/high-power effects, which have not been addressed in prior uprate reviews.

<u>Aging/Uprate Effects:</u> A significant number of major pipe ruptures have been noted for uprated plants, where synergistic effects on pipe corrosion appear largely responsible for such ruptures. A recent example is the Aug. 11/99 event at Callaway-1(PWR), where a double-ended guillotine break occurred in an 8" diameter steam line leading to a feedwater heater. It is noted that power uprates often involve increased feedwater flow/temperature conditions to accommodate the thermal load; where pipe corrosion is exacerbated at increased flow/temperature conditions. Another example is the more recent Susquehanna-2 (BWR) event in 1999, where weld failure in the BWR re-circulation line has been attributed to weld fatigue related to increased vibrations at the higher speed of the re-circulation pumps needed to accommodate the uprated power begun in 1995 (see Nucleonics Week, Vol. 40/No. 51, Dec. 23, 1999). A compilation of reactor pipe ruptures has been recently documented in an EPRI report [EPRI, *Nuclear Reactor Piping Failures at US Commercial LWRs: 1961-1997,* TR-110102, Dec. 1998], indicating in excess of 170 dramatic pipe rupture events in LWRs, ranging from single-ended pipe breaks to full double-ended guillotine ruptures of the Callaway type. The cause of such ruptures is generally due to flow/erosion or flow-assisted corrosion effects. Flow-assisted ruptures would be expected to be exacerbated at the higher flow rates that generally accompany a power uprate (primary and secondary side for BWRs, secondary side for PWRs); thus potential synergistic concerns exist and deserve additional attention in uprate reviews by the staff.

High-burnup/Elevated-power Effects: Control rod insertion problems have also been noted for extended-life fuel assemblies (burnup effect) exacerbated at high-power core locations (uprate effect). At Wolf Creek PWR plant 5 control rods failed to properly insert during a plant trip on January 30, 1996. All of the affected control rods involved Westinghouse VANTAGE-5H fuel assemblies with burnups greater than 47,6000 MWD/t-U. As indicated, the Wolf Creek plant received agency approval in 1993 for a 4.5-% power uprate from 3411MWt to 3565MWt. Root cause analysis revealed that the control rod insertion problems were caused by fuel assembly guide thimble tube distortion resulting from excessive compressive loading. The compressive loading was caused by excessive irradiation induced growth of the Zircaloy thimble tubes at high power/high-burnup core locations, indicative of potential synergistic elevated-power/extended fuel life effects.

At the recent ANS-San Diego meeting, utilities were talking of both power uprates and extended fuel duty times (higher burnup levels) in terms of nuclear plant economics in a deregulated environment. It is noted that Commonwealth Edison plans for both the Quad Cities and Dresden uprates involve not only power increase of 17-%, but in combination for with the use of the new GE-14 extended life fuel. The ACRS should question the staff on issues of potential synergistic high-burnup/high-power effects.

4) Content of Uprate Safety Evaluation Reports (SERs): The uprate SERs reviewed in my study were those for the Brunswick, Limerick, Maine Yankee, North Anna, Surry, Callaway and Wolf Creek plants. In general these SERs did not reveal any particular in-depth probe of potential issues or evidence of independent audit predictions by the staff. Indeed, a reading of these SERs gives one the distinct impression that information contained in the licensee submittal is simply paraphrased in the SERs. The following are excerpts of staff review findings found in a typical uprate-SERs, in this case the Wolf Creek Uprate SER:

Emergency Core Cooling System (ECCS): "From the licensee's study, no adverse impact to ECCS operability or vulnerability to single failure due to the re-rated conditions was identified. The licensee submitted revised ECCS performance analyses in support of Amendment 61, which justified various changes associated with Cycle 7 operation. The licensee performed large and small break analyses at the limiting re-rate conditions and determined that all acceptance criteria continued to be satisfied. The NRC staff has reviewed the licensee's analyses and concludes that the ECCS analyses referenced in support of the re-rate conditions continues to be in compliance with 10CFR50.46 and App. K. The Wolf Creek ECCS is, therefore, acceptable for operation at the re-rated conditions."

Main Steam System: "The main steam system dissipates energy generated by the reactor core to the turbine generator and auxiliary steam loads, the main condenser via the steam dump valves, or to the atmosphere via atmospheric relief valves or main steam safety valves. Isolation of the main steam system is achieved by the main steam isolation valves and main steam bypass isolation valves. The licensee evaluated the capability of the main steam system components to perform their design functions under the proposed re-rate conditions. The licensee determined that the existing set-points and capacity of the main steam safety valves are adequate to prevent exceeding 110-% of design pressure of the main steam system under the most limiting transient. The set-point and capacity of the atmospheric relief valves were found to remain adequate to control the design load shed of 10-% rated thermal power. In addition, the atmospheric relief valves were found to have adequate capacity to achieve a 50 F/hr cool-down if the main condenser was unavailable. The main steam isolation valves were evaluated to ensure the valves will continue to perform their isolation function under the maximum differential pressure conditions and within the time limits assumed in the safety analysis. The staff concludes that the existing main steam system components are adequate to perform their safety functions under the re-rated plant conditions."

Main Feedwater: "The main feedwater system delivers feedwater, at the required pressure and temperature, to the four steam generators. The safety-related portions of the system ensure isolation capability and provide a path to permit the addition of auxiliary feedwater for reactor cool-down following design basis transients. The licensee's evaluation shows that the existing design basis for the main feedwater isolation valves and main feedwater bypass isolation valves is not significantly affected by operation at the re-rate conditions. The piping configurations associated with the feedwater and auxiliary feedwater systems do not change as a result of the re-rate conditions. The ability of the auxiliary feedwater system to perform its heat removal function was addressed by the licensee. The staff finds that the safety functions of the feedwater system will continue to be satisfied during operation at the re-rate conditions."

In each of the above examples, no independent NRC analysis are cited to support the staff conclusions reached in the SER. The SERs did not specify the scope of the subject matter reviewed, how the review was accomplished, or acceptance criteria for the conclusions reached. Such information is required in the review of the original plant FSAR, as specified in the Standard Review Plan. Of particular note are standardization of acceptance criteria. In the FSAR-SRP the technical bases for the acceptance criteria are specified, including the solutions and approaches that are acceptable, which are codified in a form so that staff can rely on uniform and well-understood positions for its review. Standardization of requirements/acceptance criteria is desirable to assure consistency/adequacy of the uprate review process and documentation of staff findings and conclusions in the SER.

<u>5) Safety Margins/Risk Measures:</u> The uprate applications and associated SERs reviewed in my study did not reveal significant efforts related to an assessment of the risk, or change-in-risk, associated with the uprated power. This may be due to the fact that these submittals were reviewed prior to agency efforts at risk-informed regulation. However, future uprate approvals should require some sort of assessment of the change-in-risk or safety margins associated with the uprate. For example, one might estimate the change in failure probability and impact on risk for a piece of equipment, say for a feedwater pump or piping, operated at the higher flow rates/temperatures for uprated conditions, versus the failure probability of the same pump or pipe if it remained at the prior/lower power level. Another example relates to the Susquehanna-BWR experience, where one might estimate the failure probability of a recirculation pump due to increased vibrational fatigue at the higher flow rates of the uprated plant, versus the pump failure probability (and impact on overall risk) at the lower/slower pumping conditions at the prior power level; where delta-risk would be of interest. Some indication of the change-in-risk, or change-in-safety margin, should be required for power uprate applications.

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

June 20, 2000

MEMORANDUM TO: ACRS Members and Staff

MEMORANDUM #: AWC-105.2000

FROM: A. W. Cronenberg

SUBJECT: Travel Report: Power Uprate Paper at American Nuclear Society San Diego Meeting of June 4-8, 2000

Summary: This memo constitutes a travel report related to participation at the San Diego-ANS summer meeting, where I presented a summary paper related to my review of operational events noted for power uprates and potential synergistic safety issues. The meeting also included two embedded topical meetings, one DOE Spent Fuel & Fissile Material Management, the other on Nuclear Installation Safety. Here I briefly comment on my presentation, as well as impressions of several sessions I attended.

PRESENTATION: The paper I gave was based on work completed last fall and entitled: *Potential Synergistic Safety Issues Related to Reactor Power Uprates* (summary attached). The paper was included in a session devoted to Performance Monitoring/Trending in Support of the Maintenance Rule, with approximately 20-25 in attendance. The session was held on Thursday morning, the last day of the meeting, at a time when more than half the attendees had departed; thus a good turnout, all things considered. I used the same overheads as previously presented to the ACRS in February, which need not be repeated here. The presentation generated significant questions/discussions, which are paraphrased, as best I can recall:

a) Why has not the agency developed a formal mechanism for review of power uprates in view of the Maine Yankee experience and expected requests for power increases?

b) Can you comment more on the extent of NRC's audit of the safety analysis for design basis accidents, particularly LOCAs, which are submitted by a utility when requesting a power uprate?

c) Does the ACRS review applications for increases on the order of 1-2%, related to better system measurements, which the Commission has stated will be left for staff approval only?

d) How do you anticipate that your recommendation for inclusion of risk indicators for uprate applications be accomplished and incorporated into a utility submittal?

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e) You included QHOs as one of the risk indicators that might be included in an uprate request. How do you envision this?

My response to these comments/questions are as follows:

Comment-a: Why has not the agency developed a formal mechanism for Review of Power Uprates in view of the Maine Yankee experience and expected applications for power increases?

Reply: The Commission has put License Renewal on the fast track, so the staff has not been able to devote the time needed for development of a more formal approach for power uprate reviews. I noted that with the expected 15-17% power uprate requests for Duane Arnold and the Commonwealth Edison Dresden and Quad Cities plants, this may change.

Comment-b): Can you comment more on the extent of NRC's audit of the safety analysis for Design Basis Accidents, particularly LOCAs, which are submitted by a utility when requesting a power uprate?

Reply: I reiterated my presentation comment, that I have not seen any documentation in the uprate-SERs (Safety Evaluation Report) issued by the staff, of thermal-hydraulic or neutronic audit calculations to benchmark licensee predictions. I stated that the only audit calculations I've seen are those done after the Maine Yankee uprate approval, which were done not as part of the uprate review process but rather in response to, and after the fact, related to *Whistle Blower* allegations of faulty submittal analysis. I mentioned that the allegations were submitted to the Maine State authority, where the alligator indicated nil confidence or willingness by NRC to challenge or uncover faulty analysis. I stated that it was my personal opinion that some sort of audit of a utility's thermal-hydraulic and neutronic predictions should be required of the NRC staff, as an integral part of its review of each power uprate. I stated that maybe the Maine Yankee story might be different if this had been done. I closed with the comment that the Maine Yankee uprate story was a failure not only for the licensee, but more importantly the NRC uprate review process.

c) Does the ACRS review applications for increases on the order of 1-2%, related to better system measurements, which the Commission has stated will be left for staff review only?

Reply: I first asked for clarification of the comment; then responded that ACRS has a memo of understanding with the EDO that it will only review requests for 5-% or more.

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Comment-d): How do you anticipate that your recommendation for inclusion of risk indicators for uprate applications be accomplished and incorporated into a utility submittal?

Reply: I replied that I was really thinking of a "change in risk" or "delta-risk". I stated that one might estimate the change in failure probability (and impact on overall risk) for a piece of equipment, say for a feedwater pump or piping, operated at the higher flow rates/temperatures for uprated conditions, versus the failure probability of the same pump or pipe if it remained at the prior/lower power level conditions. Another example cited was from the Susquehanna-BWR experience, where one might estimate the risk related to failure of the recirculation pump which was thought to be due to the increased vibrational fatigue at the higher flow rates for the uprated plant, versus the risk associated with the pump failure probability at the lower/slower pumping conditions at the prior power level; again where on the delta risk would be of interest.

Comment-e: You included QHOs as one of the risk indicators that might be included in an uprate request. How do you envision this?

Reply: I said that I did not have in mind any particular Quantitative Health Objective (QHO), but rather some risk indicator; where CDF seemed those most amenable for power uprates. I said I just mentioned QHOs, because some in industry believe that QHOs should be the primary measure for assessing the real risk to the public. I also mentioned Rick Sherry's thoughts that LERF (Large Early Release Fractions) might be a better measure for public risk than CDF.

There were no more comments. I closed with the remark that I believed ACRS would, in the near future, be reviewing with the staff the adequacy of agency uprate review procedures in light of expected uprate requests in the range of 15-% or more.

Other paper at the session were entitled:

1) *Performance Monitoring/trending in Support of the Maintenance Rule at the San Onofre Plant: R. Allen of San Onofre.* I asked a question on the proposed "On-line risk monitor" for shutdown operations.....i.e. was it solely an in-house effort, did they feel they had enough risk information for shutdown conditions, and the time frame for the on-line shutdown monitor? The author replied that they were just starting to think out the basics of the shutdown monitor, but replied that he thought it would be as robust as the risk monitor for at-power conditions.

2)*Auxiliary Condenser Circulating Water Flow Optimization Using an Integrated Optimization Procedure: Z. Huang of Penn State.* I made the comment that this optimization tool might be of particular use in power uprate applications, where condenser thermal-hydraulic conditions would be expected to change to accommodate the higher power conditions, and that the condenser conditions at the uprated power might be best optimized with this tool. The author commented that he had not thought of his analysis in terms of uprate conditions, but that yes......it would seem appropriate.

3) My paper followed.

4) *Curricular Developments in Maintenance and Reliability Engineering at the University of Tennessee: Prof. Kerlin* I had no questions/comments on this paper. Others were seeking more information on the details for certification versus an actual university degree in maintenance engineering. The session closed with this paper.

OTHER PAPERS/SESSIONS:

Tues-June 6/Morning: I attended the session entitled *Overview of Space Nuclear Power*, due to personal interest in the subject and prior consulting involvement with Los Alamos Labs during the early 1980s. I came away with the general feeling that the program has been significantly scaled back from what is was in the 1980's, although their seems to be the accepted belief that nuclear power is still a viable option for space station support power. Likewise, nuclear propulsion seems to be the only viable option for deep-probe missions. Nevertheless, I got the distinct feeling that this session was more of a pep talk to the choir, with very little in the way of new technical information.

Tues-June 6/Afternoon: My afternoon was divided between two sessions, one on *Economics of Nuclear Power in a Deregulated Environment (panel discussion)*; the other on *DOE Melter Technology for Nuclear Waste Treatment*. The first session was primarily a panel discussion of how nuclear power fits into a deregulated electric utility environment, although there were several formal presentations. One was given by NEI (I forget the name of the presenter), indicating the general view that NEI expected that about 70 of the approximate 100 N-plants currently operating will survive into the next decade. Most of the surviving plant, if not all, will be owned by several (5-10) large N-plant operators, rather than current situation of numerous utility operators. He presented slides on fuel duty/cycle trends, operation/downtime trends, and indicated that both higher burn-ups and higher-power levels will be important factors in the viability of a plant in a deregulated environment. I asked the question: "We have an idea that BWRs may requests power uprates on the order of 10-20% over the next few years, noting Duane Arnold, and the Dresden and Quad Cities BWR plant; my question is do you have an idea of what can be expected for PWRs". He did not answer my question directly, saying that he did not have any specific numbers on what could be power uprates for PWRs, but that he would expect some increase over current power levels. No one on the panel offered any additional information per my question fro PWRs.

During the latter part of the afternoon, I attended a session devoted to melter technology for nuclear waste remediation, primarily due to my prior work and interest in this area. A significant portion of the session dealt with explosive hazards and off-gassing during the vitrification process, largely centering on the use of vitrification at the Hanford and Savannah River DOE sites. The session moderator also asked a BFNL manager at the session to fill in the audience on details of the recent DOE decision to terminate the BFNL-Hanford contract.

Wed.-June 7/Morning: I attended the session on *Cost Performance for Decommissioned Plants*, which largely centered on presentations by both utilities and DOE contractors on the costs related to plant decommissioning. Each individual presenter gave slides which basically outlined the costs for various elements for plant decommissioning. One of the presentations was by engineers from the Portland General Electric Company for the decommissioned Trojan plant. The most surprising part of the presentation to me was that most of the major cost

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over-runs were for non-nuclear related items, such as the costs of housing personal at the site. Indeed, the cost of actual removal and shipping of the reactor vessel and embedded piping was at or under the original budget estimate. This was also in line with Duke Engineering Company's experience, the contract managers for decommissioning of the Yankee-Rowe plant. Most major over-runs for that plant were likewise non-nuclear costs. The total decommissioning costs for these plants were also quite similar, at about 400-500 million each.

Wed.-June 7/Afternoon: Attended part of the session on *Public Confidence in Nuclear Energy (Panel discussion)*. General consensus is that nuclear power is gaining in acceptance by general public.

Thurs-June 8/Morning: Session where my paper was given (discussed above).

End of conference.

3. Potential Synergistic Safety Issues Related to Reactor Power Uprates, *August W. Cronenberg (NRC)*

During the past several decades, the U.S. Nuclear Regulatory Commission (NRC) has reviewed and approved more than 30 licensee requests for power uprates. Each request has been evaluated to ensure that plant safety and regulatory requirements are satisfied. Recent events, however, point to potential synergistic concerns that may not have been adequately covered in the uprate application-and-review process. Specifically, higher power levels when combined with system/component degradation via plant aging, as well as high power in combination with fuel-life extensions to elevated burnup levels, may affect safety margins. Evidence of these effects stem from recent operational events, including failure to fully insert control rods in high-power high-burnup fuel assemblies and piping failures. This paper examines the potential for synergistic effects (*synergistic*—the cooperative action of discrete agencies such that the total effect is greater than the sum of the individual effects) and the adequacy of the NRC uprate review process.

A utility seeking a power uprate will submit a licensing amendment request (LAR), which contains information similar to that found in the original Final Safety Analysis Report but at the uprated power level. The LAR centers on a reevaluation of design-basis accidents and off-normal transients at the higher power level, the adequacy of safety systems to perform their intended function, and a no-significant-hazard assessment. Changes to plant equipment, operating conditions, and technical specifications to achieve the intended power increase must also be specified. Information presented in the LAR is reviewed by the NRC staff, and its findings are reported in an uprate Safety Evaluation Report (SER). The NRC review is conducted in accordance with 10 CFR-Part 50.59 and encompasses consideration of any new or unreviewed safety concerns. The uprate application is approved if the case has been made that there is no significant degradation in plant safety margins and that all applicable regulations are satisfied.

The uprate applications reviewed in this study include that for the Brunswick, Callaway, Maine Yankee, North Anna, Surry, Susquehanna, and Wolf Creek plants. A review of the LARs and SERs for these plants revealed little documentation with regard to consideration of potential synergistic effects of high-core-power densities when combined with component aging or high-burnup effects. A review of operational events for uprated plants, however, points to potential compounding effects. Examples include the control rod insertion problems noted at the Wolf Creek and North Anna plants, both having received power uprate approvals in the range of 4 to 5%. At the Wolf Creek plant, five control rods failed to fully insert during scram from full power. The affected control rods involved Westinghouse Vantage-5H fuel assemblies in the area of 47 600 MWd/t U. Root-cause analyses indicate distortion of the Zircaloy control rod guide tubes (thimbles) due to irradiation-induced growth. Because Zircaloy irradiation growth is influenced by neutron energy spectrum and flux (power-level effects), as well as total exposure (burnup effect), potential synergisms may exist. Control rod sticking problems have also been noted at North Anna-1. An examination of the Wolf Creek and North Anna uprate documentation (LARs and SERs) for control rod behavior did not reveal consideration of the effects of higher power level when combined with elevated burnup conditions. Other incidents include power offset anomalies for long-cycle/high-power cores tied to crud buildup on high-burnup fuel rods. The crud appears to gather boron, causing a distortion of the axial power profile, particularly in high-power assemblies, indicative of potential elevated power/burnup synergisms.

Aged reactor components and systems, combined with the higher flow rates that often accompany uprates [primary and secondary flow increases for boiling water reactor (BWR) uprates; secondary-side flow increases for pressurized water reactor (PWR) uprates], may likewise produce degradation that is greater than the sum of the individual effects. Research has shown that pipe corrosion can be exacerbated at increased fluid velocities, indicative of a synergistic corrosion (aging)/erosion (flow) process. Root-cause analysis has also pointed to corrosion (aging)/vibration-fatigue (flow)-induced pipe failures. Uprated plants that have experienced recent pipe failures attributed to corrosion/erosion and corrosion/vibrational effects include the Callaway (PWR) break of an 8-in. steam line leading to a feedwater heater and a weld leak in a 1-in. line in the recirculation system at the Susquehanna (BWR) plant.

Inadequacies were also noted in the Maine Yankee and Brunswick uprate reviews, where deficient licensee submittal information was not uncovered during the initial review by the NRC staff. These incidents point to a need for independent agency thermal-hydraulic and neutronic audit analysis to verify licensee submittal information. NRC in-house computational efforts would go a long way in providing an independent check and verification of what is now essentially a licensee effort. In view of these observations, the following recommendations are made:

1. NRC should issue a Standard Review Plan (SRP) for power uprate applications, which should include acceptance criteria that consider the influence of potential synergistic effects, specifically high-fuel-burnup levels and component/system aging effects combined with uprated power conditions. The NRC is in the process of developing a power uprate SRP.

2. NRC uprate review procedures should include requirements for independent NRC staff analysis (i.e., thermal-hydraulic and neutronic code predictions) to verify uprate predictions submitted by the licensee. The results of NRC audit calculations should be part of the SER for each uprate review and include comparisons with licensee submittal analysis.

3. A comparison of probabilistic safety measures (e.g., core damage frequency, QHO, and LERF) at the uprated and prior power levels is recommended for future uprate applications.

4. Reliability Analysis of Aging Effects Considering Imperfect Testing and Maintenance, *Kang M. Park, Young W. You, Chang H. Chung (Seoul Natl Univ–Korea)*

Aging effects are dealt with in the evaluation of periodic test and maintenance, replacement, and life extension. Because the interval-averaged unavailability for a component and a system is used in the existing reliability analyses, continuous time trend of each component's unavailability cannot be analyzed. The use of both extremes, as-good-as-new and as-bad-as-old, in the quantitative evaluation of test and maintenance has difficulty in reflecting the actual maintenance activities. In this paper, time-dependent unavailability is derived under periodic test and maintenance with the discrete renewal process, and accumulated aging effects are evaluated with the introduction of a new factor based on imperfect test and maintenance and its sensitivity analysis.

METHODOLOGY

The unavailability can be derived with the discrete renewal process.[1–3] The result is as follows:

$$Q(t) = R(nT)CF_0(t)$$

$$= \sum_{k=1}^{n} Q(kT)(Q(kT)(1 - p_k) + R((n-k)T)CF_k(t))$$

$$nT < t < (n+1)T\,, \qquad (1)$$

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

May 18, 2000

MEMORANDUM TO: G. Wallis, Chairman, Thermal-Hydraulic Phenomena Subcommittee

FROM: P. Boehnert, Senior Staff Engineer

SUBJECT: NRR MEETING WITH DUANE ARNOLD PLANT LICENSEE ON DUANE ARNOLD ENERGY CENTER POWER UPRATE, MAY 16, 2000 - ROCKVILLE, MARYLAND

Representatives of Alliant Energy (AE - Duane Arnold Energy Center Licensee) and NRR met on May 16, 2000 to discuss the proposed power uprate program. Key points noted during the discussion included the following:

- The proposed power increase is 15.3% above the current license value of 1658 MWt to a level of 1912 MWt. The original power level was 1593 MWt; a previously granted power increase of 4.7% was made. The overall result is a total increase of 20.0% from the original licensed power level.

- In addition to the power increase, AE is proposing to implement the alternative source term (AST). The proposal is to use full implementation pursuant to Draft Regulatory Guide DG-1081[1]. The chemical and physical form of the radionuclides assumed as well as timing will be based on the DG-1081 guidance. AE prefers to provide a stand-alone AST submittal to support the uprate request with subsequent amendments to request AST relaxations. Details are provided in the attached handouts. The cognizant NRR representative present (S. LaVie) indicated that AE's approach for the AST submittal appeared reasonable.

- The analysis conditions will be for an uprate of 120% rated core thermal power, use of a new fuel design (GE 14) and extension to a 24-month operating cycle. For the equilibrium uprate cycles, batch average fuel burnup will be ~50 GWD/MTU, with a 58.8 GWD/MTU peak rod burnup for the equilibrium uprate cycles.

- For the next fuel cycle (Cycle 18), the licensee is proposing use of a "transition core", consisting of a mix of three GE fuel types (GE 10, 12, and 14 design fuel assemblies)

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[1] DG-1081: "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Plants". The Committee will review the proposed final version of this regulatory guide and associated Standard Review Plan Section during its June Meeting.

as well as an intermediate power level (between the current level of 1658MWt and the ultimate 1912 MWt value). This approach gave the staff some pause, particularly with regard to the scope and structure of the Cycle 18 Core Operating Limits Report. In the end, however, NRR opined that this approach should be workable.

- AE's proposed schedule for submittal and staff approval appears extremely optimistic. The submittal is to be made in October and approval was requested by the end of May, 2001. The following points are of note however:

  o AE has not completed all of its analyses pertaining to the uprate, including the evaluations for ATWS and the containment. All analyses are to be complete by July. As a result, AE does not yet have a comprehensive list of all the hardware change-outs/modifications that will be required (items mentioned included installation of a new high-pressure turbine and the separator/dryer assembly located in the vessel).

  o I informed AE that the ACRS intends to review this uprate and that AE and the staff needed to factor Committee review into the schedule. While AE gave indication that they were cognizant of prior Committee uprate reviews (they were aware that the ACRS will want to discuss the risk aspects of the uprate), they appeared to have not planned on Committee review of this matter.

  o AE was requesting approval to use the TRACG code to support aspects of the uprate. NRR indicated that their review of TRACG would not be complete in time to support the licensee's advertised schedule.

Near the conclusion of the meeting, AE began exploring "fall-back" positions centering on what aspects of the uprate review could be accomplished consistent with the above-noted schedule.

NRR proposed holding its next meeting with AE sometime in July, subsequent to the licensee's completion of its uprate analyses.

Attachment: As Stated

cc: Balance of ACRS Members
R. Savio

cc w/o attach (via E-mail):
J. Larkins
H. Larson
S. Duraiswamy
ACRS Technical Staff & Fellows

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DAEC Alternate Source Term Implementation

Chuck Nelson

Principal Engineer

Power Uprate Team

319-851-7778

chucknelson@alliant-energy.com

Conditions for Analysis

- Analyzed Conditions
  - 120% Original Rated Thermal Power
    - (1593 MWt Original, 1658 MWt Current, 1912 MWt Uprate)
  - GE14 Fuel (new fuel)
  - 24 Month Operating Cycle
    converting to

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Scope of Submittal

- Full Implementation per DG-1081 1.2.1
  - Composition - NUREG 1465
  - Magnitude - ORIGEN2 (GE code runs specific to DA)
  - Chemical and Physical Form - DG-1081
  - Timing
    - DG-1081
    - BWROG Report "Prediction of Fission Gas Release from Fuel in Generic BWR"

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Scope of Submittal

- Analyzed Design Basis Accidents
  - Loss of Coolant Accident LOCA - App A
  - Fuel Handling Accident FHA - App B
  - Control Rod Drop Accident CRDA - App C
  - Main Steam Line Break MSLB - App D
- Output - Doses and Acceptance Criteria in TEDE

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Equipment Qualification
Impact of Cesium

- DAEC is performing EQ Evaluations in Power Uprate Analysis using TID-14844 Source Term in accordance with interim NRC staff guidance in SECY-99-240. per pending GI on Cs
- DAEC will address the Cesium Impact in accordance with the resolution of the pending GSI or DG-1081 when issued.

RADTRAD Default 60 Isotope List

- DAEC Reviewed RADTRAD Default NIF files
  - PWR and BWR 60 Isotope Inventories use the same isotopes.
  - It is our understanding that these are the same isotopes used in NRC evaluations.
  - DAEC is using the same list

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RADTRAD Default 60 Isotope List

- Cobalt Isotopes
  - Co-58, Co-60 are corrosion and activation products, not fission products. ORIGEN cutoff at 1E-8 => no Co fission products
  - ORIGEN cutoff 1E-10 => only Co-74 and above
  - DAEC is using the RADTRAD BWR Default NIF values for Co-58 and Co-60 - significantly higher than actual coolant borne concentrations
- EQ Isotopes - 60 isotopes using TID fractions

Submittal Options

- Option 1 Standalone Submittal of AST to support Separate Power Uprate Submittal with Future Amendments to Request AST Relaxations (DAEC Preferred)
- Option 2 Combined AST and Relaxations Submittal
- Option 3 Combined AST, Power Uprate and Relaxations Submittal

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Considerations

The Recommended Option:

- Minimize changes so focus is on effects of power uprate.
- Minimizes schedule impact on power uprate review.
- Minimizes complexity of AST review.

Since the resolution of current industry issues (e.g., EQ GSI, CR Habitability) may impact cost/benefits for some relaxations (e.g., timing, leakage), DAEC will take additional time to determine the mix of exemptions to be pursued.

Submittal Schedule

- Analysis in Progress
- Shell for Submittal being developed from GGNS and Perry Submittals
- Last Engineering Task Report due July 00
- AST Submittal to NRC 3rd Quarter 2000

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

June 6, 2000

MEMORANDUM TO: Graham Wallis, Chairman, Thermal-Hydraulic Phenomena Subcommittee

FROM: P. Boehnert, Senior Staff Engineer

SUBJECT: NRR MEETING WITH COMMONWEALTH EDISON COMPANY (COMED) — CORE POWER UPRATE PROGRAM FOR DRESDEN AND QUAD CITIES PLANTS, MAY 31, 2000, ROCKVILLE, MARYLAND

Representatives of NRR and ComEd met on May 31, 2000 to hold a "kick-off" meeting to discuss ComEd's licensing plan to support extended power uprates for Dresden Units 2 & 3 and Quad Cities Units 1 & 2, as well as transition to use of GE14 fuel. Key points noted during the meeting include:

- ComEd will transition to use of the new GE14 fuel design in all its BWR units. Currently ComEd's BWRs are using a mix of GE and Siemens ATRIUM-9B fuel. Discussion ensued over GE's plans to apply its GEXL critical-power ratio calculation methodology to the Siemens fuel type, absent knowledge of its design/test parameters, which are held proprietary. GE explained that they would need to perform a series of interpolative calculations which ComEd would, in turn, need to evaluate, since they have access to the Siemens proprietary information. NRR raised concerns regarding GE's lack of knowledge of the applicability of the Siemens design parameters to uprate power conditions, the bounding of uncertainties, as well as the overall approach being employed. The staff advised GE of the need to submit a comprehensive report on this methodology for its review, as soon as practicable.

- The Dresden and Quad Cities Units are to be uprated by 17% of the current licensed power level. ComEd maintains that the impact for an uprate of this magnitude is minimal, as substantial design margin exists in both the NSSS and balance-of-plant equipment for units of this vintage (BWR/3). In response to my question, ComEd said that both plants received 5% power uprates shortly after initial licensing, pursuant to AEC practice at that time (early 1970s). Technically, this power increase represents an overall uprate of 22% above the initial licensed level. Given this, the staff requested that ComEd address the applicability of the GE generic analyses supporting the Extended Uprate Program, as this Program was limited to uprates of no more than 20% of nominal power.

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A list of significant plant modifications was provided (Figure 1). Regarding the need for additional cooling towers at the Dresden site, NRR cited a concern with the impact of the site's heat rejection capabilities during high temperature conditions on such plant parameters as suppression pool temperature limits (e.g., elevated spray pond temperatures). The licensee has not yet performed the safety analyses supporting the uprate; therefore, additional modifications may be necessary.

- ComEd discussed its approach for the safety analysis supporting the uprates for the four units. Designated "Unit 5", it will consist of a set of bounding inputs for the safety analyses and use of the MELLL (maximum extended load line limit - Figure 2) for plant operation at the increased power level. Figure 3 provides some additional details on the Unit 5 approach.

- The licensee intends to submit its uprate license amendment request by the end of this year. NRC review would need to be completed within ~ eight to nine month's time to support the proposed restart schedule for the first uprated unit, Dresden Unit 2, in November 2001. I made note of the ACRS's intention to review this uprate application, and the need to include time for Committee review in the above schedule. ACRS Fellow G. Cronenberg indicated that the Committee is concerned with the lack of a NRC review plan (Standard Review Plan Section) for power uprates[1]. ComEd indicated that they will be in a position to uprate the plants in mid-cycle, if necessary, given any review schedule delays.

- During discussion, ComEd noted that the cost of the uprate power is ~$175/kW(e).

Attachments: As Stated

cc: Balance of ACRS Members
R. Savio

cc w/o attach (via E-mail):
J. Larkins
H. Larson
S. Duraiswamy
ACRS Technical Staff & Fellows

[1] Subsequent to the meeting, Dr. Cronenberg and I discussed this matter with Mr. Duraiswamy. I sent you an E-Mail message recommending that the Committee engage the staff in a dialogue on the need for development of a Standard Review Plan for review of power uprates. You indicated support of this approach. The P&P Subcommittee is scheduled to discuss this matter during its June 6, 2000 Meeting.

Significant Modifications

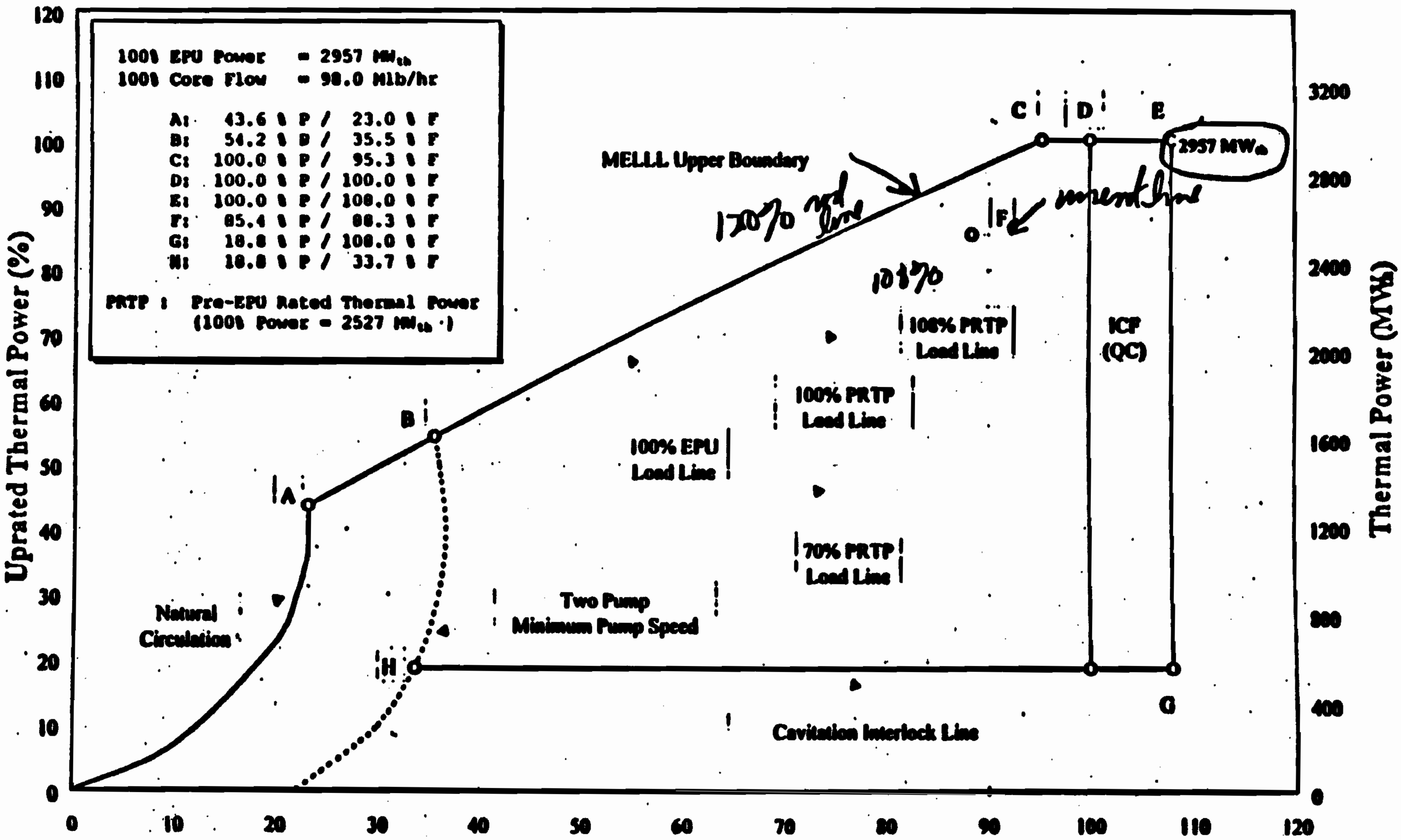
- Replace HP turbines
- Add new condensate demineralizers
- Recirculation pump runback on FW or CD pump trip
- Off gas temperature conditioning
- Heater drain valve replacements
- Auxiliary power system changes
- Instrument setpoint changes
- Additional cooling towers at Dresden

ComEd
A Unicom Company

Fdl Repl. done on fuel outages

Dresden and Quad Cities MELLL Power/Flow Map

FIG. 3



Unit 5 Analytical Approach for DR/QC

F/6.3

– What is Unit 5?
  - A bounding set of analysis inputs for the four Dresden and Quad Cities units for EPU/MELLL SAR (e.g. LOCA, containment analysis)
  - Safety analysis results/impacts due to EPU/MELLL will be presented in the PUSAR for review and approval
  - Unit/cycle specific models will be used for reload safety analyses according to the NRC-approved methods
– Why Unit 5?
  - Only a few differences between the four units (typical BWR3)
  - More efficient analysis and review
  - Common design bases for consistency and maintenance
  - Uprated core thermal power will be the same for all four units
– How?
  - Current safety analysis inputs of the four units were compiled/reviewed
  - Unit 5 model jointly developed by ComEd/GE by selecting the limiting parameter(s)
  - Justification for choice of limiting parameter compiled
  - Parameter choice is dependent on analysis

ComEd
A Unicom Company

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

August 8, 2000

MEMORANDUM TO: P. Boehnert, S. Duraiswamy, J. Larkins, H. Larson, D. Powers, G. Apostolakis

MEMORANDUM #: AWC-110.2000

FROM: A. W. Cronenberg

SUBJECT: Beaver Valley Power Uprate & License Renewal Plans

Attached is a copy of an overview presentation by First Energy Nuclear Operating Company (FENOC) made on Aug. 8, 2000 to NRC, concerning management plans for upgrading the Beaver Valley Units 1 & 2 (W-PWRs), which primarily relate to plans for a Request for License Renewal, Request for Power Uprate, and anticipated plant equipment upgrades (i.e. replacement of the Unit-1 steam generator). It should be noted that the Beaver Valley units were purchased by FENOC from Duquesne Power about a year ago, and that the attached represents the FENOC long-term management plan for the units. The presentation was made for information purposes only, where it is noted that no License Request submittals have been made to date, except for the 1.4-% power uprate related to Appendix-K considerations for measurement of set point parameters (a staff review item only). The key items of interest to ACRS, I believe, are the anticipated plans for:

(a) a License Amendment Request for a total Power Uprate of about 6-%(1.4-% in 2000 related to App.-K set points, and a full-scope 5-% increase in late 2002),
(b) a License Renewal Request in 2004 for 20 year extension.

The schedule for these items is given in the second slide on page 4 of the attached handout. It is noted that the full-scope 5-% power uprate (late-2002) will include a request to allow for changeover from a sub-atmospheric containment to atmospheric conditions, to facilitate future in-containment maintenance activities during at-power operations. In this regard the licensee will submit new source term retention and containment structural response calculations for Design Basis Accidents. I suspect that the containment analysis will require ACRS review (prior unreviewed safety concerns), in addition to that associated with the power uprate itself. The licensee stated that they will base their source term submittal on *best estimate analysis*, using the MAAP code (analysis under contract to Fauske & Associates/Westinghouse-BNFL). This may require prior review of the MAAP code by ACRS.

The NRC/NRR contact for the Beaver Valley units is Dan Collins (415-1427).

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NRC/FENOC MEETING BEAVER VALLEY FULL POTENTIAL PROGRAM

AUGUST 8, 2000

FENOC

DESIRED OUTCOME

- Understanding of focus, scope, and strategy of Beaver Valley's Full Potential Program
- Understanding of relationship of reactor vessel issues to Full Potential Program

FENOC

AGENDA

- Full Potential Program
  - William R. Kline
- Reactor vessel issues
  - Dennis Weakland
- Concluding remarks
  - William R. Kline

FENOC

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GOALS

- Improve plant safety, reduce operating costs through risk informed technologies
- Improve capacity factor at least 15%
- Increase Mwe output at least 6%
- Convert to atmospheric containment
- Convert to Improved Standard Technical Specifications
- Extend operating license period 20 years
- Reliable steam generator operation
- Replace Unit 1 steam generators

FENOC

INDIVIDUAL PROJECTS

- Steam generator management
- Uprates/Atmospheric containment
- Improved Standard Technical Specifications
- Capacity factor/outage improvements
- Steam generator replacement
- License renewal
- Asset management
- Fuel management

FENOC

FULL POTENTIAL PROGRAM STRATEGY

- Phased implementation
  - Efficiency
  - Revenue generation
- Long term focus
  - Improve/preserve assets
  - Extend life

FENOC

PTS SCREENING LIMIT

- PTS Screening limit change
  - Methodology currently in regulations
    - 10CFR50.61 defines method
    - Screening limit being reevaluated by NRC Research
    - Correlations also being reevaluated by NRC Research
  - Timing of regulatory change uncertain
  - Impact of correlation change and screening limit uncertain

FENOC

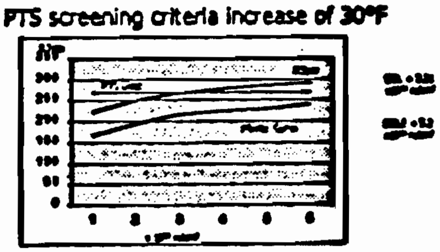
MASTER CURVE

- Master Curve Methodology
  - Methodology has been Codified
    - ASTM E1921-97
    - Code Case N-629 (Unirradiated)
    - Code Case N-631 (Irradiated)
  - RT_{T_0} measured directly from unirradiated and irradiated surveillance specimens
  - Kewanee Nuclear Power Plant submittal to apply this methodology is under active review by NRR

FENOC

PRELIMINARY MASTER CURVE RESULTS

- Master Curve methodology
- PTS screening criteria increase of 30°F



FENOC

CONCLUDING REMARKS

- Full Potential Program
  - Comprehensive, integrated plan
  - Maximize asset potential
  - Showcase of deregulated nuclear generation
- Success paths exist for reactor vessel issues

FENOC

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TRIP REPORT

AMERICAN NUCLEAR SOCIETY

2000 UTILITY WORKING CONFERENCE

"MANAGING THE BUSINESS OF NUCLEAR POWER"

AMELIA ISLAND, FLORIDA

AUGUST 6-10, 2000

Noel Dudley

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TRIP REPORT

American Nuclear Society 2000 Utility Working Conference
Managing the Business of Nuclear Power
Amelia Island, Florida
August 6-10, 2000
by
Noel Dudley

I attended the first two days of the 2000 Utility Working Conference sponsored by the American Nuclear Society. The Conference, as its name implies, focused on operating nuclear power plants as businesses with regulatory oversight as one of the activities that needs to be managed. After the Plenary Session on Monday morning, the Conference offered seven different parallel tacks of break out sessions. I followed the Regulatory Relations track and attended sessions on reactor oversight, risk-informed regulations, and license renewal. The fourth day of the Conference consisted of an ANS Professional Development Workshop concerning the maintenance rule and condition monitoring. The agenda for the Conference and the Workshop is attached [pp 9-23].

Approximately 250 people attended the Conference. The 15 NRC staff members who attended also participated as panel members during selected sessions. Vendors, consultants, industry executives, and licensee regulatory and maintenance managers were well represented. A list of the attendees is available upon request.

OBSERVATIONS

The scenario of a resurgence of nuclear power presented at the Conference was predicated on the following assumptions. With the increasing demand for reliable power being driven by the communications industry, power generators will be motivated to build more base-load electric power plants. Due to deregulation, the price volatility of fossil fuels, and environmental concerns, nuclear power will be competitive with other fuel sources. Since deregulation will allow a better return on capital, investors will be more likely to accept the risk associated with financing construction of new nuclear power plants.

Based on the content of the different sessions at the Conference, issues that nuclear power executives need to consider under the assumed scenario are operating nuclear power plants as businesses, increasing present generating capacity, maintaining aging equipment, retaining a skilled work force, and effectively managing the regulatory environment.

Some speakers noted the importance of protecting public health and safety. Other speakers explained that operating a nuclear plant as a low cost power producer will ensure safety. However, speakers did not talk about how safety will be maintained or how the regulatory structure will ensure public health and safety.

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PLENARY SESSION

The speakers during the plenary session were upbeat about the future of nuclear power. Most speakers focused on the business aspect of operation a nuclear power plant and expressed a belief that nuclear power is competitive with any other source of electric power. Only two of the speakers stressed the need for continuing to operate the plants safely.

Commissioner Merrifield made the case that the outlook for nuclear power is the brightest it has been since the Three Mile Island (TMI) accident. He presented lessons learned from a book entitled "Containing the Atom" by J. Samuel Walker, the NRC Historian. One lesson learned from the 1950's is the need to prevent licensing bottlenecks caused by lack of NRC resources. Commissioner Merrifield noted that the NRC should establish an infrastructure to support the review of license renewal application as a priority over reviewing the applications. Another lesson from the 1960's concerned the loss of public confidence resulting from the fear of radiation from nuclear weapons' fallout. Similarly, the fear created by the plant events at TMI and Chernobyl in the 1970's and 1980's adversely effected nuclear power. He noted that public confidence must be earned and that it is fragile. He warned against advocating further cuts in the NRC that may adversely effect public confidence. A copy of Commissioner Merrifield's remarks are attached [pp 24-31].

In response to questions Commissioner Merrifield made the following statements:

- The Commission will wait until the consolidation of the electric power industry is completed before deciding on changes to the boundaries of NRC Regional Offices.
- Fewer licensees would not necessarily make it easier to regulate the 103 operating plants.
- The NRC will review new plant design applications, such as the pebble bed design, when they are received.
- Based on his discussions with resident inspections, the inspectors are changing their negative views towards the Revised Regulatory Oversight Process.
- The federal government should rescind its prohibition against foreign ownership of nuclear power plants, since nuclear proliferation is no longer an issue.
- Performance indicators enhance public confidence.
- Maintaining adequate electric power is not the NRC's responsibility. The NRC should ensure plants can withstand voltage dips on the power grids.
- The NRC should improve the timeliness of its review of spent fuel storage issues including high burnup fuel and damaged fuel elements in spent fuel pools.

- The NRC has provided the infrastructure necessary to license a new plant within one year. An application for construction of a new nuclear power plant may be submitted within 7 to 10 years.

Mr. Jerry Yelverton, President and CEO of Entergy Nuclear, Inc., presented statistics demonstrating how safely and efficiently nuclear power plants are being operated. He stated that the consolidation of the nuclear power generation industry would continue partly because it reduces a company's risk and provides the least expensive source of base-load electricity. Mr. Yelverton stated that the larger companies would hire the skilled craftsmen they need and would be less reliant on operating service providers. He explained that the costs of electricity from present nuclear power plants is less that the cost of electricity from operating gas plants due to the present high cost of gas. Mr. Yelverton was encouraged by license renewal, which would allow present plants to continue to operate until new plants are built. He concluded that nuclear power is competitive with other sources of electricity because of increased capacity (power uprates), lower staffing and operating costs, and stable fuel cost. Selected slides used during this presentation are attached [pp 32-37].

Dr. Lucian Conway, President, Conway Consulting, provides financial decision making training to industry executives. He explained how the transition from a regulated monopoly business model to a deregulated model will result in reducing expenses and working capital. He stated that the return on the regulated portions of an electric company is 12 percent and that the return on the unregulated portions of a company is about 20 percent. As a result, more funds will flow into nuclear power plants as deregulation progresses. Dr. Conway concluded that if nuclear power plants are run as businesses they will be competitive with other types of electricity generators.

Mr. David A. Christian, Senior Vice President-Nuclear, Dominion Generation, explained that North Anna and Surry nuclear power plants were low cost producers because the organizational vision of safe operations had been adopted by the employees. He stated that increased capacity factors resulted in increased revenues. He noted that increased regulation of fossil fuel emissions would make nuclear power even more competitive. Concerning mergers and takeovers, Mr. Christian observed that quality was added by increased safety and not increased size. He concluded that licensees should be less concerned with safety risk and concentrate on managerial factors. His remarks are attached [pp 38-50].

Mr. Edward Tirello, Jr., Managing Director, Deutche Banc Alex Brown, is a Wallstreet electric power company analyst. He disagreed with the decision to separate power generation, transmission, and distribution into different companies. He stated that as the electric power industry consolidates the nuclear power producers must think like a business since they will be the largest profit centers. He speculated that there eventually will be eight to ten transmission companies that will need to invest billions of dollars in upgrading the national infrastructure. He noted the present regulatory environment discourages investment in transmission lines. Mr. Tirello explained that the reserved margin for operating grids has dropped to 8 percent and peak electricity usage has extended into the evening hours due to the internet. He stated that the demand for electricity by high tech companies presently represents 13 percent of the demand and will increase to 25 percent.

Mr. Tirello projected that five to eight major electric generating companies would compete on each transmission grid and that 30-40 distribution companies would sell different grades of power nation wide. He expected high tech companies would purchase reliable nuclear power and that consumers and warehouse facilities would purchase less reliable power generated by electric peaking stations. Mr. Tirello stated that electric power generating companies must be managed as businesses similar to the vicious competition that occurs in the food industry. He speculated that oil companies would soon become involved in the electric power generation industry.

In response to questions, Mr. Tirello provided the following answers:

- In the present rapidly changing business environment, the NRC needs to approve new plant applications within months and not years.

- Distributive power generation equipment, such as fuel cells, soon will be used at major buildings, farms, and telecommunication towers.

- The government has collected money, which is earning interest, and is doing nothing to build a high level waste repository. A law suit is needed to make progress.

- Regulators can adapt to the new business environment, but change will be slow and training will be needed.

- Craft workers are not drawn to nuclear facilities because the industry is viewed as dying. The industry needs to sell job security and better manage overtime work.

- Competition has not decreased the sharing of information since the industry is interdependent with regards to maintaining a safety focus.

- Since utilities have been purchasing vendor organizations and developing buying groups, the demand for vendor services will decrease.

- Training, which provides an understanding of economic competition, information on how well the company is doing, and what the employee can do to help, should be used to motivate employees to be more productive and efficient.

REACTOR OVERSIGHT PROGRAM

My conclusion from this session is that the Revised Regulatory Oversight Program (RROP) is supported by both the NRC and the industry and will be revised to improve its effectiveness. The differences and issues between the NRC and the industry were clearly identified, the need to revise the present RROP was repeatedly noted, and the quality of the working relationship between the NRC and industry was continually highlighted. The NRC staff stated that the cross-cutting issues will be reflected in the performance indicators and that additional leading indicators are unnecessary. One staff member stated that performance indicator thresholds would be exceeded before a risk significant event occurs.

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The RROP is predicated on the effectiveness of each licensee's corrective action program. In a private discussion, a consultant who reviews licensee corrective action programs stated that most licensees have weak corrective action programs. The primary weaknesses of the programs were the inability to correct identified problems and the inability to identify latent errors, which could eventually result in self-identifying problems.

Mr. William Dean, NRC, described the regulatory framework and the initial implementation of the RROP. He explained that the NRC staff was developing tools to improve the objectivity and consistency of the performance indicators and the Significance Determination Processes (SDPs). Others issues discussed were fault exposure time in relationship to reliability and the need to more closely inspect the cross-cutting issues. Mr. Dean described future NRC initiatives such as establishing an initial implementation evaluation panel, considering the use of risk-based performance indicators, evaluating resources, and holding public workshops. Selected slides used during this presentation are attached [pp 51-53].

Mr. Peter Wilson, NRC, provided an overview of the SDPs and explained that the SPDs for the cornerstones are works in progress. He concluded that due to the SDPs, objectivity has improved and the NRC is better focused on safety significant issues. Selected slides used during this presentation are attached [pp 54-61].

Ms. Donna Alexander, Manager, Regulatory Affairs, Carolina Power and Light Company, presented insights gained during the pilot program for the RROP. She stated that no major process changes were necessary to implement the oversight program. She noted that internal thresholds were lower than the NRC thresholds and that program results were reviewed by a panel before being sent to the NRC. Ms. Alexander stated that the number of inspection hours appeared to increase and that the inspection reports were pretty bland. She suggested that resident inspector observations, similar to those contained in the old style inspection reports, should be conveyed to the licensee. She questioned the appropriateness of the NRC issuing three violations for one event. Ms. Alexander concluded that communications with the NRC had been good and that the RROP was still dynamic as indicated by the continued use of draft documents for program guidance.

Mr. Greg Gibson, Nuclear Oversight and Regulatory Affairs Division, Southern California Edison Company, presented insights from the Shadow Plant Program, which involved licensees not in the pilot program applying the RROP to their plants. He reported that the licensees who participated in the Shadow Plant Program were pleased with the oversight process. Mr. Gibson identified problems with the thresholds used for the health physics and security performance indicators. He stated that there should be a disciplined process for adding new performance indicators. Mr. Gibson indicated that including the responses to frequently asked questions on the NRC web site provided instantaneous up-to-date information to all licensees.

Mr. Stephen Floyd, Nuclear Energy Institute, summarized the latest performance indicator results and identified issues related to the RROP. The slides used to summarize the performance indicator results are attached [pp 62-63].

Mr. Floyd discussed the following issues related to the performance indicators:

- potential adverse effects of counting manual scrams on operator decisions,
- better definition of unplanned power changes,
- safety system availability criteria are more restrictive than maintenance rule criteria, and
- inconsistent use of components, systems, or trains to determine availability.

Mr. Floyd noted that the RROP had significantly reduced the number of enforcement actions. For example, severity level IV violations have dropped from 1037 to about 8 per year. Slides showing the reduction in enforcement actions are attached [pp 64-65]. He indicated that the SDPs still need work to establish consistency and standardization.

RISK-INFORMED REGULATION

During this session it was difficult to determine the level of interest of the industry in risk-informing the regulations either thought Option 2 and/or Option 3. The NEI and South Texas Project representatives were very vocal in their support of both Option 2 and Option 3. In a private discussion, one licensee representative stated that there is no benefit to his facility in maintaining a probabilistic risk assessment or developing risk-informed license amendments partly because the Q List is small.

Mr. Timothy Reed, NRC, presented an overview of the staff's approach to risk-informing 10 CFR Part 50 special treatment requirements. He described the development of 10 CFR 50.69, Appendix T, the associated regulatory guide, and the Nuclear Energy Institute's implementation guidance document. He identified the following technical issues that are still under discussion.

- quality of probabilistic risk assessments (PRA),
- peer certification and expert panels,
- monitoring and providing feedback on RISC III components,
- comparison of commercial practices and 10 CFR Part 50 Appendix B requirements, and
- change controls to ensure PRA assumptions remain valid.

Mr. Eugene Hughes, President , Erin Engineering, presented the role of the regulator in a risk-informed environment. Using dam safety, coal mining safety, and the WASH 1400 report as examples, he explained that as knowledge about a safety hazard increases the need for regulation decreases. Mr. Hughes stated that proposed regulations should address the risk perceived by the public in an effective way. He suggested that risk-informed requirements could be one of the following types:

- prescriptive that would require direct oversight,
- performance-based that would require periodic inspections, and
- incentive-based that would require self-oversight.

Mr. Hughes asserted that rules should be imposed only when a result is expected, such as a defined level of performance, availability, or reliability and when it is necessary to ensure safety. He used the example of components on licensees' Q Lists, and asked the question what is achieved when a component is on the List. Mr. Hughes noted that NASA required highly

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reliable components since there is no redundancy built into many space craft systems. He questioned whether the same level of quality assurance is necessary at nuclear plants where a single component failure is assumed as part of the general design criteria. He recommended further discussion of commercial grade components.

Mr. James Chapman, Director, PSA and Safety Analyses, SCIENTECH, defined risk-informed and presented a case that the insights derived from PRAs over the last two decades have proven to be robust. Mr. Chapman highlighted important risk-informed initiatives that have been completed. He predicted that South Texas Project will be successful in implementing graded quality assurance. He concluded that the regulations do not need to be revised to improve regulatory activities. Selected slides used during the presentation are attached [pp 66-70].

Mr. Mark Reinhart, NRC, described the use of probabilistic safety assessments in the regulatory framework, such as in reviewing deterministic license amendments, quality assurance programs, and technical specification configuration management.

LICENSE RENEWAL

Mr. Stephen Hoffman, NRC, provided an overview of the license renewal activities and 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." He outlined the process for license renewal including the license renewal application, the environmental review, and the opportunity for a hearing. He noted that license renewal is a business decision. Mr. Hoffman summarized the schedule for reviewing generic license renewal guidance documents and expected license renewal applications.

Mr. Barth Doroshuk, President and Chief Operating Officer, Constellation Nuclear Services, provided an overview of the preparation of the Calvert Cliffs Nuclear Power Plant license renewal application. He stated that life cycle management required a good understanding of plant historical behavior, resources, mitigation measures, discovery techniques, and corrective action and follow-up. He noted that the license renewal application is not only a regulatory document but also provides an extended planning horizon for the engineering department. Mr. Doroshuck stated that license renewal lessons learned can be viewed from the perspectives of assessment and planning, management decision support, project implementation, application preparation, and implementation of a plant lifetime program.

Mr. Greg Robison, Project Manager, License Renewal, Duke power Company, provided an overview of the preparation of the Oconee license renewal application. He explained that license renewal was a business decision that required the assessment of plant and equipment mortality. He noted that reconstruction of 30 year old decisions was difficult and some re-engineering of past decisions was necessary. Mr. Robison made the following observations:

- clear definition of terms is important,
- do not assume preparers and reviewers are on the same page,
- the licensee is the application integrator - the NRC staff is the application reviewer,
- the license renewal process must be standardized, and
- a technically sound process for reviewing emerging issues should be developed.

Mr. Tony Pietrangelo, NEI, provided an overview of the license renewal process. He stated that scoping the plant is a labor intensive effort and that licensees need to know their current licensing basis. He recommended that license renewal personnel speak the same language as plant personnel and not attempt to teach license renewal to plant operators. Mr. Pietrangelo explained that NEI 95-10 will be the source document for preparing applications while the Standard Review Plant for license renewal and the Generic Aging Lessons Learned (GALL) report will stabilize the application review process.

The following information was exchanged during the meeting session discussions:

- system engineers do the scoping,
- design engineers identify the aging management programs, and
- due to process requirements 24 months is the minimum possible time to review an application.

Attachments: As Stated

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DR. BONACA, 8/21/2000.

Potential synergistic effects of industry initiatives to extend plant life, increase production and reduce regulatory burden

The License Renewal (LR) Rule rests on the basic regulatory principle that a nuclear power plant (NPP) can continue to operate for as long as it complies with its current licensing basis (CLB), because compliance with its CLB provides assurance of adequate protection.

The LR implementation documents provide details on how an NPP demonstrates that aging degradation will be adequately managed so that plant structures, systems and components (SSCs) will continue to comply with their CLB requirements for as long as the plant continues to operate. Active components are excluded from LR consideration because existing regulation already imposes requirements on timing and level of corrective action required when they fail.

Passive components fall into two different categories.

One category includes passive components subject to periodic replacement under their CLB. These components are identified for the purpose of reviewing existing CLB commitments dealing with age degradation and to assess their adequacy for the extended period of operation.

The other category includes long-lived passive components that are not subject to periodic replacement under their current CLB. This category includes major reactor coolant system components such as reactor coolant system piping, reactor vessel and internals, pressurizer and steam generators in pressurized water reactors (PWR), reactor coolant pump casings, emergency systems piping, secondary side major components such as steam lines, and containment. For these components aging degradation is monitored to assure that it will not exceed aging degradation limits required to support the CLB. In those cases where component operation is supported by a time limited aging analysis that does not extend beyond 40 years, the time limited aging analysis must be modified to qualify the component for the extended period of operation.

In most instances long-lived passive components are expected to operate for the extended period of operation without being replaced. This is possible because these components are designed with excess margin over the regulatory limits that support the CLB. Part of this excess margin is in fact intended to, and used for operating the plants to their currently licensed 40 years life. Extending the life of the plants beyond 40 years involves the recognition that excess margin is still available in most components after 40 years of operation and the acceptance of its use to compensate for aging degradation for the purpose of extending the life of the facility. Since regulatory limits are not exceeded, the plant continues to comply with its CLB, and this provides assurance of adequate protection.

Although regulatory limits are not exceeded, SSCs actual margins to aging degradation limits are being reduced. At the end of 60 years life, mechanical components will be closer to their fatigue limits than at the end of 40 years, the reactor vessel will be more brittle and closer to the PTS limit than at 40 years of life, and so on, and even replacement steam generators, which should be capable of reaching the end of 60 years plant life, will exhibit aging degradation from 20 additional years of service.

If a complete PRA of the plant that would appropriately describe aging effects were performed at 40 years of life, and then again at 60 years, one would expect to observe an increase in risk measures such as CDF and LERF, due to an expected higher failure probability of long lived components subjected to 20 more years of service. Higher failure rates would tend to affect PRA results in several ways:

- By increasing initiator frequency of accidents caused by rupture of passive components,
- By increasing the possibility of cascading failures from physical interaction of ruptured components with adjacent age-degraded components,
- By increasing the probability of failure of engineered safeguards, and
- By reducing the structural capability of the RCS and containment barriers during severe accidents.

This increase in risk measures may not be insignificant and may exceed the guidelines of Regulatory Guide 1.174 at least for plants characterized by relatively high CDF and LERF.

As stated above, the regulatory logic behind the decision to implement the LR rule without further risk consideration seems to be based on the basic concept that a plant complying with the current deterministic regulation meets the requirements for adequate protection even if its risk to the public increases with age. This concept is accepted for the first 40 years of life. The LR rule extends its acceptance beyond the first 40 years. Since the LR rule does not establish a life extension limit, there is an implication that the LR rule will allow as maximum acceptable risk from aging the one associated with a condition where all long-lived components have aged to their regulatory limit without exceeding it. This approach would not be in conflict with the guidelines of RG 1.174 if current PRAs of operating plants already assumed aging of all components to their regulatory limits and met the subsidiary safety goals. But current PRAs have not explicitly and systematically addressed aging effects, and many plants do not meet the subsidiary safety goals of CDF and LERF. Therefore, granting a renewed license without consideration of aging risk may in some cases conflict with the guidelines of RG 1.174.

Even if we accept license renewal without consideration of associated risk, as an extension of the licensing philosophy supporting the first 40 years of operation, concerns remain about the risk implications of concurrent licensing actions proposed by licensees that compete for the same SSC margins used to support life extension and that are likely to be evaluated without explicit consideration of aging. The exclusion of aging risk considerations from the LR rule does not mean that the aging effects due to LR don't need to be considered in risk assessments of other licensing actions that may be affected by the aging of components.

For example, several plants are planning power up-rates. In his June 23 report to the ACRS on this subject, Dr. Cronenberg noted that "several recent operational events for uprated plants point to circumstantial evidence of compounding degradation due to aging/uprate and high-burnup/high-power effects, which have not been addressed in prior uprate reviews." The report provides several examples of pipe failures that have occurred in uprated plants. The report also states "Agency inaction for a more comprehensive uprate review process is being justified by risk arguments of minor changes in CDF for power uprates.".

These power uprate requests will come in for review and approval through licensing actions under the provisions of existing deterministic rules. A study performed by Energy Research, Inc. (ERI) for the Swiss Nuclear Inspectorate in 1997 (Ref.) assessed the risk associated with a 14.7% power increase of the Leibstadt NPP in Switzerland. Leibstadt is a BWR6 with a MarkIII containment. The study showed that the power upgrade would result in minor increases in CDF and LERF, but in a 30% increase in risk as measured by the risk metric of frequency of a release times the activity of the release. The study also showed that the metrics of RG 1.174 are not the most appropriate to assess such risk increase. Even if the NRC were to perform a risk assessment of such uprates using the insight of the ERI study, approval or denial of the licensing request is likely to be based on the merits of the uprate request alone, without consideration of the additional risks associated with other licensing actions such as license renewal and of the potential synergistic effects resulting from the combined licensing actions.

Since many NPPs are planning to extend their life, and many are planning power upgrades, we may face a situation where a plant characterized by high risk (maybe not apparent because its PRA is incomplete or inadequate) could be allowed to raise its power level, and as a separate action go for life extension. Another concurrent separate action could include justifying continued operation for some time with degraded steam generators. The current licensing process does not allow for risk considerations to effectively enter into the decision of whether these plant actions can be supported simultaneously or even individually. PRA is the only tool having the capability of comprehensively exploring the synergies of such proposed plant changes. But its benefits are effectively excluded by

- Current lack of information (from incomplete or inadequate PRAs) on the actual risk associated with operating plants,
- Explicit exclusion of PRA considerations from LR rule,
- Weak understanding of impact of aging on plant risk (no systematic PRA study has been performed, methodology has only partially been developed)
- Lack of complete PRA models to seriously evaluate the synergistic effects of industry initiatives to increase production, extend life and reduce regulatory burden.

The staff needs to be prepared to address the global issue being raised by the Industry's move to aggressively utilize existing plant margin above minimum regulatory limits. Piecemeal review and approval of industry requests may fail to identify important synergies that may result from the separate licensing actions. We need to understand what the NRC in general, and RES in particular are doing about this issue. Depending on the staff's initiatives in this area we may need to recommend a focused effort in our research report. Also, the metrics of RG 1.174 may need to be augmented if CDF and LERF are not sufficient to identify plant risk associated with licensing actions, as the ERI report seems to suggest.

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ANS 2000 Utility Working Conference

Managing the Business of Nuclear Power

Trip Report
By
Noel Dudley

August 31, 2000

PLENARY SESSION

- Commissioner Merrifield
  - Establish Infrastructure to Support License Renewal
  - Strong Regulator and Public Confidence
  - License New Plant Within One year of Application
- Industry, Financial, and Academic Speakers
  - Manage Nuclear Power Plants as a Business
  - Low Cost Producers ⇔ Organizational Vision of Safety
  - Regulated Companies 12% Return; Deregulated 20%
  - License New Plants Within Months of Application

REVISED REACTOR OVERSIGHT PROCESS

- RROP Supported by NRC and Industry
- Cross-cutting Issues Reflected in Performance Indicators
- Leading Indicators are Unnecessary
- Implications to Enforcement Activities
- Need Process to Retain Inspectors' Observations
- Weakness in Corrective Action Programs
  - Inability to Correct Problems
  - Inability to Identify Latent Errors

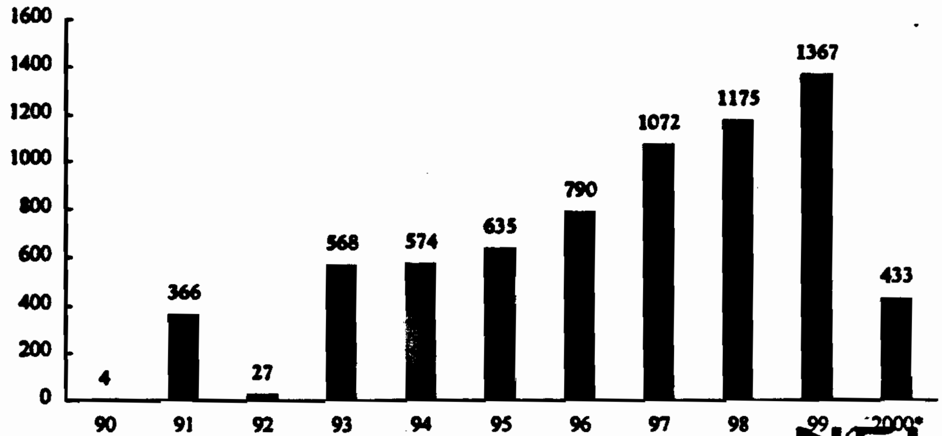
RISK-INFORMED REGULATION

- Technical Issues Still Under Discussion
  - Quality of PRAs
  - Peer Certification and Expert Panels
  - Commercial Grade vs. Appendix B Requirements
- Impose Rule Only When a Result is Expected
  - Prescriptive ⇒ Direct Inspection
  - Performance-Based ⇒ Periodic Inspections
  - Incentive-Based ⇒ Self-Oversight
- Insights From PRAs Have Been Robust
- Level of Industry Interest in Option 2 or 3

LICENSE RENEWAL

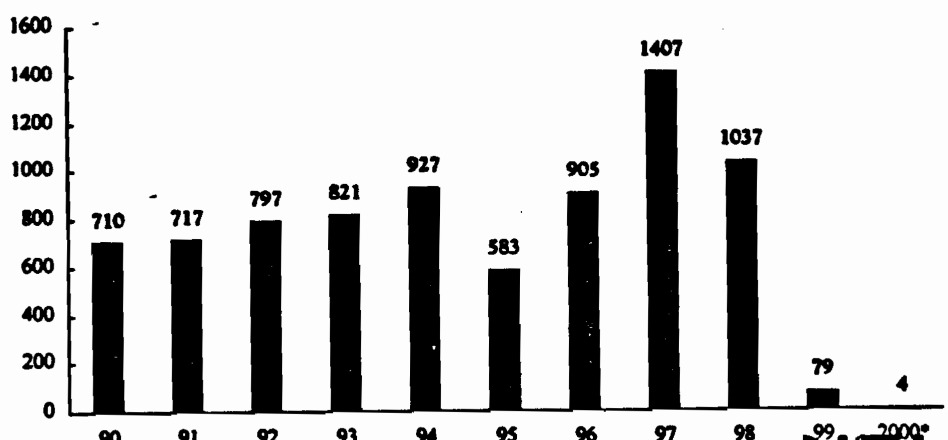
- Applications Reflect Life Cycle Management
- Business Decision Based on Plant Mortality
- Scoping/Screening Process is Labor Intensive
- Reconstruction of 30 Year Old Decisions is Difficult
- License Renewal Engineers Should Speak Language of Plant Operators
- 24 Months is Minimum Time to Review an Application

Total Non-Cited Industry Violations (1990-2000)



Source: NUS *Violations through June 2000

Industry Violations: Severity Level IV (1990-2000)



Source: NUS *Violations through June 2000

DPO PLAN

- SUBCOMMITTEE VETS
- FULL ACRS DECIDES

SUBCOMMITTEE	TOPIC	REVIEWER
D. Powers	Chair, Iodine Spiking	Seale
Bonaca	Risk, Human Factors	Apostolakis
Kress	Severe Accidents, Thermal-hydraulic	Wallis
Shack	Metallurgy	Consultant
Sieber	NDE	Uhrig

Staff Support: Undine Shoop/Sam Duraiswamy
Available Consultants: Ivan Catton, Jim Higgins, Richard Ricker

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Ground Rules

- If we omit an area of research, the omission will be interpreted as an indication the area is unimportant

- ACNW will prepare its own report dealing with research for waste repositories and research done by NMSS

- We will not examine organizational structure

- Our focus will be on the long term research needed to facilitate the execution of NRC's mission in the future

- We should, however, help the Commission to understand when a research effort has yielded enough information for regulatory decisionmaking

21a

PLAN FOR RESEARCH REPORT

- **Develop essays in each of 12 areas on NRC's long-term research needs . . . . October**
- **Subcommittee meeting (November 1) for Q&A sessions with staff**
- **Review essays . . . . . . November**
- **Unified document for ACRS approval . December**

RESEARCH REPORT RESPONSIBILITIES

Topic	Member	Reviewer
Civil/Structural Engineering	Bonaca	
Criticality	Seale	
Fire Protection	Sieber	
Fuel	Cronenberg	
Human Performance	Sorensen	
Digital I&C	Uhrig	
Materials, NDE, Steam Generator	Shack	
Mechanical Engineering	Sieber	
PRA	?	
Radiation Dosimetry	Seale	
Severe Accidents	Kress	
Thermal Hydraulics	Wallis	

Thurs 8/31
@ 9:45

21b

Plan for Research Report

* develop essays in each of 12 areas on NRCs long term research needs .... October

* Subcommittee Nov. 1 for Q&A sessions with Staff

* Review essays - Nov.

* Unified document for ACRS approval - Dec.

Research Report Responsibilities

Topic	Member	Reviewer
Civil/Structural Eng.	Bonaca	
Criticality	Seale	
~~Radiation Dosimetry~~	~~Seale~~	
Fire Protection	Sieber	
Fuel	Cronenberg	
Human Performance	Sorensen	
I&C	Uhrig	
Mat'ls, NDE, Steam Gen	Shack	
Mechanical Eng.	Sieber	
PRA	?	
Radiation Dosimetry	Seale	
Severe Accidents	Kress	
Thermal Hydraulics	Wallis	

21c

Some Thoughts on the Needs for Regulatory Research

D.A. Powers

Chairman
Advisory Committee on Reactor Safeguards

Visual aids used for meeting with Rogers Committee

- **ACRS regularly reviews the NRC Research Program**

- **The most recent of these reviews are in a series NUREG - 1635 Volumes 1-3**

- **We have only started the review for this year**

- **Today, I can only present my own views which have not had the benefit of deliberation within the entire ACRS**

World View

- **Contrary to view of many, neither the nuclear industry nor the Nuclear Regulatory Commission is a mature static institution**

  - Industry:

    - **power uprates cutting into margin**
    - **extended burnup fuel**
    - **license extension & PTS**
    - **best estimate accident analysis**
    - **shortened outages**

  - NRC

    - **risk informed regulation**
    - **Reg. Guide 1.174**
    - **revised source term**
    - **performance-based monitoring and inspection**
    - **risk-informed enforcement**

The Paradigm for Research

- **In the past, research at NRC has been focused on finding and mitigating the residual risks posed by the commercial use of nuclear power**

  - NRC interest in a topical area was often a source of consternation to the industry though often the research saved the industry money and time

- **Now research needs to be used to develop the processes and procedures to carry out the NRC function to assure adequate protection of public health and safety**

  - NRC can carry out its regulatory mandate without research

  - But, without research, it may not do this efficiently and it certainly will be quite conservative

A Question

To identify NRC research needs, a question that must be addressed is:

> *"When will the NRC do independent assessments rather than simply reviewing licensee submittals?"*

ACRS does not now have a satisfactory answer to this question.

Do line organizations at NRC now have ready access to risk information to support a risk-informed regulatory process?

- line organizations cannot not readily carry out risk assessments and uncertainty analyses for license amendments for specific plants.

- inspectors cannot not independently evaluate risks associated with licensee plans for shutdown operations, yet scoping assessments by NRC and others suggest risk during shutdown is comparable to risk during normal operations.

- line organizations cannot do fire risk assessments to resolve debates with licensees on topics such as the ongoing discussion of hot shorts during fires, yet the IPEEEs suggest risk from fire is comparable to risk during normal operations.

Does NRC have the technical capability to assure that the combination of power uprates, extended burnup fuel, and best estimate accident analyses do not erode safety margins unacceptably?

- NRC's thermal hydraulics research program seems to be well founded and well pursued, though it may be underfunded and thus may not have goals that will meet all the agency needs in this area.

- There is not a well-researched Standard Review Plan for power uprates that considers the synergisms of all the changes taking place in the ways plants are run.

- Fuel research, which had atrophied, has been revived in a limited way. NRC is conceding that it will rely on licensee submittals for fuel burnups beyond current limits.

Digital Electronic Reactor Control Systems

There is a consensus among the informed technical community that software-based digital electronic systems for reactor control and reactor safety functions offer tremendous improvements over the existing analog systems. Yet, the nuclear industry has been slow to join the digital revolution underway in nearly all other industries.

- Digital systems are susceptible to common mode failures.

- Regulatory review is hostage to standards more appropriate for far more complicated digital systems.

- NRC has lacked the research resources to develop more appropriate regulatory requirements for digital safety systems.

Revised Reactor Source Term

- **NRC has used the results of its past research on accident source terms to provide a more realistic accident source term for safety analyses.**

  - NRC lacks the research resources to fully participate in an international collaborative experimental project that will allow experimental verification of many of the models that lead to the revised source term.

  - NRC is losing the technical capacity to independently assess licensee submittals dealing with source term behavior in the reactor containment and the efficacy of engineered safety systems. The problem is especially acute in dealing with the chemistry of radioactive iodine.

Analysis versus Experiment

- **NRC lacks the resources to support extensive or routine experimental verification of its analyses.**

  - A case in point is the analysis of spent fuel pool fires that will establish the risk basis for revised safety rules for decommissioning plants.

  - Another example of where analysis may be used without adequate experimental support may be arising in connection with the issue of hot shorts.

Conclusions

- NRC needs research to develop the risk analysis tools that can be used by line organizations in the agency to support a risk-informed regulatory process.

- NRC needs research to develop an appreciation of the synergisms of developments in the industry dealing with power upgrades, higher fuel burnups and extended licensing periods.

- NRC needs research to develop more workable regulatory requirements for digital electronic systems for reactor instrumentation and safety functions.

- NRC needs research to provide tools to support the use of the revised accident source term.

- NRC needs resources to more fully participate in international collaborative research into reactor safety issues.

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
475	16	16-2

Title **RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS**

Author **Howard Larson**

SUBJECT	EDO LTR	ACRS LTR	Analysis	
NEI LETTER DATED JANUARY 16,200, ADDRESSING NRC PLANS FOR RISK-INFORMING THE TECHNICAL REQUIREMENTS IN 10 CFR PART 50	8/30/00 (PP 1-2)	7/20/00 (PP 3-6)	9/1/00 (P 7)	16

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

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

August 30, 2000

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

SUBJECT: ACRS LETTER DATED JULY 20, 2000, "NUCLEAR ENERGY INSTITUTE LETTER DATED JANUARY 19, 2000, ADDRESSING NRC PLANS FOR RISK-INFORMING THE TECHNICAL REQUIREMENTS IN 10 CFR PART 50"

Dear Dr. Powers:

The subject letter to Chairman Meserve discussed the Nuclear Energy Institute's (NEI's) January 19, 2000, letter, as well as the NRC staff's work to risk-inform the technical requirements contained in 10 CFR Part 50. More specifically, your letter provided two recommendations with respect to this work. These recommendations, and our responses, are:

> <u>ACRS Recommendation 1</u>: The staff should proceed with finalizing the framework for risk-informing the technical requirements of 10 CFR Part 50, including the prioritization criteria, and use the information in the NEI letter, as appropriate.
>
> <u>Staff Response</u>: We agree. We are continuing to use the framework, to revise it to reflect your comments as well as the comments of others, and to apply it in the evaluation of 10 CFR 50.44 ("Standards for combustible gas control system in light-water-cooled power reactors"), 10 CFR 50.46 ("Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"), and other sections of 10 CFR Part 50. We intend to provide a revised version of the framework to the Commission for their information, and as background for recommendations on modifying 10 CFR 50.44, at the end of August 2000.
>
> <u>ACRS Recommendation 2</u>: The staff will want to interact further with the industry to determine the benefits and burden reduction that could result from changes in rules in light of risk information.
>
> <u>Staff Response</u>: We agree. We are planning to have additional public meetings and workshops to discuss our work, including obtaining input on the benefits of possible changes to various sections of 10 CFR Part 50.

/

Your letter also discussed alternative approaches to risk-informing 10 CFR 50.46, including potential changes to ECCS success criteria and the definition of challenges to the ECCS. As you know, we are now considering possible changes to Section 50.46; your ideas on this are thus particularly timely and will be considered as we proceed.

Sincerely,

William Travers

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

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

July 20, 2000

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

Dear Chairman Meserve:

SUBJECT: NUCLEAR ENERGY INSTITUTE LETTER DATED JANUARY 19, 2000, ADDRESSING NRC PLANS FOR RISK-INFORMING THE TECHNICAL REQUIREMENTS IN 10 CFR PART 50

During the 474th meeting of the Advisory Committee on Reactor Safeguards, July 12-14, 2000, we discussed the subject letter to NRC Chairman Meserve. In addition, we discussed with representatives of the staff and the Nuclear Energy Institute (NEI) the NRC plans for risk-informing the technical requirements in 10 CFR Part 50. During our discussions, we had the benefit of the documents referenced.

This report responds to the Commission's request in the April 5, 2000 Staff Requirements Memorandum (SRM) that the ACRS review the subject letter.

Recommendations

1. The staff should proceed with finalizing the framework for risk-informing the technical requirements of 10 CFR 50, including the prioritization criteria, and use the information in the NEI letter, as appropriate.

2. The staff will want to interact further with the Industry to determine the benefits and burden reduction that could result from changes in rules in light of risk information.

Background

The Commission directed the staff to develop a plan for risk-informing technical requirements in 10 CFR Part 50. In response to staff activities in this area, NEI conducted an industry survey to identify regulations that are prime candidates for assessment and change or possible candidates for improvement. This was the subject of an NEI letter dated January 19, 2000, to Chairman Meserve. In an SRM dated April 5, 2000, the Commission requested that:

> The ACRS review the January 19, 2000, letter from the Nuclear Energy Institute (NEI) to Chairman Meserve, that addresses NRC plans for risk-informing the technical requirements in 10 CFR Part 50. In particular, the ACRS, in coordination with the NRC staff, should evaluate the priority listing of regulatory requirements that might be modified based on consideration of risk. This includes review of interim staff reports on the activities described in SECY-99-256 and SECY-99-264.

In SECY-98-300, "Options for Risk-Informed Revisions to 10 CFR Part 50 - Domestic Licensing of Production and Utilization Facilities," the staff proposed three options for modifying regulations in 10 CFR Part 50 to make them risk informed. These options were:

1. Continue with ongoing rulemaking, but make no additional changes to Part 50.

2. Make changes to the overall scope of systems, structures, and components (SSCs) covered by those sections of Part 50 requiring special treatment (such as quality assurance, technical specifications, environmental qualification, and 10 CFR 50.59 by formulating new definitions of safety-related and important-to-safety SSCs).

3. Make changes to specific requirements in the body of regulations, including general design criteria.

In the SRM of June 8, 1999, the Commission approved proceeding with the current rulemaking in Option 1, implementing Option 2, and proceeding with a study of Option 3. For Option 3, the Commission requested that the staff determine how best to proceed and provide a detailed plan outlining its recommendations regarding specific regulatory changes that should be pursued. SECY-99-256 provides the staff's plans for implementing Option 2. SECY-99-264 provides the staff's plans with respect to the Commission request to proceed with a study of Option 3.

The letter of January 19, 2000, which is the primary subject of this report, provided the industry's initial response to SECY-99-264. In this letter, NEI stated that there is general industry support for the overall approach. NEI also reported the results of a survey to which 61 units responded. This survey identified what the industry considers as prime candidate regulations for assessment and change and provided estimates of the financial benefits expected from risk-informing each identified regulation.

Discussion

It is appropriate that the staff consider the industry's priorities and seek information from the industry on the expected benefits. The industry priority list appears to be primarily driven by burden reduction and the associated cost savings. This is an important input in the prioritization process. The industry presumably is the best judge of the burden associated with a regulation, and this input will be valuable to the staff in developing its own priority listing. Many of the NEI priority items seem to relate to the scope of SSCs important to safety, quality assurance, and in-service inspection. These items are already incorporated under Option 1 and Option 2 and, thus, are already being given priority. The staff has also accelerated its preparation of a risk-informed revision to 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors."

In SECY-00-0086, "Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3)," the staff proposed a framework for prioritization, consideration of defense in depth, safety margins, and uncertainties. Because this framework is still under development, it is premature for us to comment. We believe, however, that this framework is appropriate and its development should continue.

If the staff is to have reliable estimates of the benefits of risk-informing selected parts of 10 CFR Part 50, there must be some sort of determination of the possible plant changes that will result. This determination appears to require first developing the risk-informed version of the rule and then identifying the possible changes on a plant-by-plant basis. After the staff has decided on the risk-informed version of a particular rule, it may want to further interact with the industry to determine the ranges of benefits – including uncertainties. For risk/benefit decisions, uncertainties in benefits are just as important as uncertainties in risk.

The highest priority candidate in the NEI letter is 10 CFR 50.46 and Appendix K related to emergency core cooling system (ECCS). The NEI letter provided information on the potential benefit (of up to $3 million per unit per year) as one of the bases for this selection. In our view, 10 CFR 50.46 and Appendix K can be considered as a deterministic specification on how good the ECCS cooling capability must be after it is activated. Its risk implications relate primarily to success criteria – will the ECCS be good enough to provide assurance that the accident will be terminated and long-term shutdown cooling provided. Probabilistic risk assessment insights, however, also suggest that the proposed challenge to the ECCS, an instantaneous double ended guillotine break (DEGB), is an extremely unlikely event.

It is not clear that substantial changes can be made in terms of the success criteria. Successful continued cooling involves evaluation of the effects of potential local hot spots, possible geometry changes as a result of rod bowing and clad swelling, local dry out, steam-zirconium chemical reactions, and possible propagation of loss of coolant from local to substantial involvement of the core. Such phenomena are highly uncertain and, therefore, must have proper criteria to provide the required confidence to be attached to the success criteria that the accident will be terminated and the core damage frequency acceptance value will be achieved. In our view, then, this is an area with a strong defense-in-depth component related to the proper balance between prevention and mitigation in a highly uncertain phenomenological area.

There appear to be greater benefits from reconsidering changes in the definition of the challenges to the ECCS, i.e., replacement of the DEGB, with an alternative large-break loss-of-coolant accident. It has long been recognized that the DEGB has led to undesirable consequences in the structural design of piping systems. It may also have negative consequences when used as the design basis for ECCS. It could, for example, result in a greater likelihood of pressurized thermal shock and lead to unrealistic startup times for emergency equipment that can reduce reliability.

On the other hand, the use of the DEGB can be considered as a sort of margin on the acceptable performance of ECCS. A systematic assessment, therefore, of the consequences of this change must be considered. Although the staff's framework is still under development, it does include a proposed process to appropriately consider the impacts of changes to the regulations. We look forward to interacting with the staff in its development of the final framework.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated January 19, 2000, from Joe F. Colvin, President and Chief Executive Officer, NEI, to Richard A. Meserve, Chairman, NRC, regarding Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.
2. Memorandum dated April 5, 2000, from Annette L. Vietti-Cook, Secretary, NRC, to John T. Larkins, ACRS/ACNW, Subject: Staff Requirements - Meeting with ACRS on Risk Informing 10 CFR Part 50, March 2, 2000.
3. Memorandum dated June 8, 1999, from Annette L. Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-98-300 - Options for Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
4. Memorandum dated April 12, 2000, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-00-0086, Subject: Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3).
5. Memorandum dated October 29, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-99-256, Subject: Rulemaking Plan for Risk-Informing Special Treatment Requirements.
6. Memorandum dated November 8, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-99-264, Subject: Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.
7. Memorandum dated December 23, 1998, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-98-300, Subject: Options for Risk-Informed Revisions to 10 CFR Part 50 - "Domestic Licensing of Production and Utilization Facilities."

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

September 1, 2000

MEMORANDUM TO: ACRS Members

Noel Dudley

FROM: Noel Dudley, Senior Staff Engineer

SUBJECT: ANALYSIS OF THE EDO'S RESPONSE TO THE ACRS REPORT ON THE NEI LETTER ADDRESSING NRC PLANS FOR RISK-INFORMING 10 CFR PART 50

The purpose of this memorandum is to provide an analysis of the August 30, 2000 memorandum from the Executive Director for Operations (EDO) that responded to the Committee's report dated July 20, 2000. The Committee's report concerned a letter from the Nuclear Energy Institute that addressed NRC plans for risk-informing the technical requirements in 10 CFR Part 50.

The Committee recommended that the staff finalize the risk-informed framework and continue to interact with the industry.

The staff stated that it was continuing to revise the framework to reflect the Committee's comments and to apply the framework in the evaluation of specific 10 CFR Part 50 requirements. The staff plans to have additional public meetings and workshop on this matter. In addition, the staff is considering possible changes to 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," as suggested by the Committee.

Analysis

The staff has initiated activities that are responsive to the Committee's recommendations and plans to keep the Committee informed of its progress in developing the regulatory framework.

cc via e-mail:
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