SL-0484

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 represen-tatives 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.

<u>OTHER RELATED ACTIVITIES OF THE COMMITTEE</u>

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

C 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.

C 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.

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

- The Committee decided that it was satisfied with the EDO's response concerning the ACRS/ACNW Joint report concerning defense in depth in NMSS activities. However, the Committee recommended that the ACRS/ACNW Joint Subcommittee follow-up on selected issues during future meetings 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 decided to review the NEI 00-02 document "Industry PRA Peer Review Process Guidelines."

PROPOSED SCHEDULE FOR THE 476TH ACRS MEETING

The Committee agreed to consider the following topics during the 476th ACRS meeting, October 5-7, 2000:

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

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

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

Pressurized Thermal Shock Technical Bases Reevaluation Project
Briefing by and discussions with representatives of the NRC staff regarding the pressurized thermal shock technical bases reevaluation project.

Discussion of Topics for Meeting with the NRC Commissioners
Discussion of topics and preparation for meeting with the NRC Commissioners scheduled for 9:30 a.m. - 12:00 Noon, Friday, October 6 concerning:

— Risk Informing 10 CFR 50
  - S NEI Letter of January 19, 2000
  - S 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)

— Quality of PRAs
  - S Assessment of the Quality of PRAs
  - S ASME Standard on PRAs

— Spent Fuel Pool Fire Safety Study

— More Realistic (Best Estimate) Thermal-Hydraulic Codes

— Status of ACRS Activities on License Renewals

Discussion of Industry Issues
Presentation by R. Beedle, Senior Vice President, NEI, on issues of mutual interest.

Proposed Resolution of GSI-168, Equipment Qualification of Electrical Equipment
Briefing by and discussions with representatives of the NRC staff regarding the proposed resolution of GSI-168, Equipment Qualification.

ACRS Review of Generic Guidance Documents Associated with License Renewal
The Committee members will discuss concerns identified during their initial review of the draft guidance documents.

Annual Report to the Commission on the NRC Safety Research Program
Discussion of the current status of the review by the members regarding the topical areas previously assigned.

Sincerely,

/RA/

Dana A. Powers
Chairman