UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

SL-0548

February 26, 2007

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Klein:

SUBJECT: SUMMARY REPORT - 539th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, FEBRUARY 1-3, 2007, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 539th meeting, February 1-3, 2007, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letter, and memorandum:

REPORTS:

Reports to Dale E. Klein, Chairman, NRC, from William J. Shack, Chairman, ACRS:

- Browns Ferry Nuclear Plant, Unit 1, 5-Percent Power Uprate, dated February 16, 2007

- Report on the Safety Aspects of the License Renewal Application for the Oyster Creek Generating Station, dated February 8, 2007

LETTER:

Letter to Luis A. Reyes, Executive Director for Operations, NRC, from William J. Shack, Chairman, ACRS:

- Draft Final Revision 1 to Regulatory Guide 1.189 (DG-1170), "Fire Protection for Nuclear Power Plants," dated February 14, 2007

MEMORANDUM:

Memorandum to Luis A. Reyes, Executive Director for Operations, NRC, from Frank P. Gillespie, Executive Director, ACRS:

- Proposed Revisions to Standard Review Plan Sections in Support of New Reactor Licensing, dated February 6, 2007

HIGHLIGHTS OF KEY ISSUES

1. Final Review of the Power Uprate Application for the Browns Ferry Nuclear Plant, Unit 1

The Committee met with representatives of the Tennessee Valley Authority and the NRC staff to discuss the proposed 5-percent uprate application for Browns Ferry Nuclear Plant, Unit 1. The discussions focused on the modifications to the plant to make it ready for restart after being shut down since 1985, the plans for the uprate, and details of several of the analyses that were used to support the uprate request. This application is the first step in uprating all three Browns Ferry units to 120-percent of their original licensed thermal power levels. The discussion was focused on the licensee's request to change its licensing-basis methodology to include credit for containment accident pressure in calculating the net positive suction head for the residual heat removal (RHR) and core spray pumps during a variety of scenarios. The licensee explained the basis for the analysis assumptions and described the results of testing that had been performed on a similar RHR pump at Browns Ferry Unit 3 to demonstrate that the pumps can operate successfully in cavitation mode for a limited period of time.

Committee Action

The Committee issued a report to the NRC Chairman on this matter dated February 16, 2007, recommending that the 5-percent power uprate for Browns Ferry Unit 1 be granted. The Committee also noted that granting of containment overpressure credit during long-term loss-of-coolant accident (LOCA) and 10 CFR Part 50 Appendix R fire scenarios at 120-percent power will require support by more complete evaluations.

2. Final Review of the License Renewal Application for the Oyster Creek Generating Station

The Committee met with representatives of the NRC staff and its contractor (Sandia National Laboratories [SNL]), members of the public, and AmerGen Energy Company, LLC (AmerGen) and its contractors to review the license renewal application for the Oyster Creek Generating Station (OCGS) and the updated Safety Evaluation Report (SER) prepared by the NRC staff. The applicant, AmerGen, has requested approval for continued operation for a period of 20 years beyond the current license expiration date of April 9, 2009.

AmerGen representatives described the leakage from the reactor cavity liner that caused corrosion of the exterior surface of the drywell shell and the corrective actions taken to prevent water from entering the sand bed region. The applicant concluded that the corrective actions taken to mitigate the drywell shell corrosion have been effective, the corrosion in the embedded portion of the drywell shell is not significant, the drywell shell meets the ASME Code requirements, and an effective aging management program is in place to ensure continued safe operation.

The staff described buckling analyses of the OCGS drywell shell performed by General Electric (GE) and SNL. The key difference between these analyses is the inclusion of hoop tensile stresses. The staff concluded that if the SNL analysis included these hoop tensile stresses, the minimum thickness results would be similar to the GE analysis. In the updated SER, the staff concluded that the applicant has appropriately identified the structures, systems, and components within the scope of license renewal and that the aging management programs described by the applicant are appropriate and sufficient to manage aging of long-lived passive components that are within the scope of license renewal.

A representative from the Coalition to Stop the Relicensing of Oyster Creek expressed concerns regarding the drywell shell, the analysis methods used to evaluate the drywell shell, and the adequacy of the inspection data used in the analyses.

The ACRS members received a letter from Jon S. Corzine, Governor of the State of New Jersey (NJ), inviting the Committee to tour OCGS and hold its public meeting in NJ to facilitate public attendance. The ACRS members also received a letter signed by Senator Frank Lautenberg (NJ), Senator Robert Menendez (NJ), Congressman Christopher Smith (NJ), and Congressman Jim Saxton (NJ) asking the Committee to ensure that the safety issues regarding the drywell are fully resolved before it makes any decisions regarding the OCGS license renewal application. The NRC is in the process of responding to these letters.

Committee Action:

The Committee issued a report to the NRC Chairman on this matter dated February 8, 2007, recommending that the application for license renewal for OCGS be approved with the incorporation of certain license conditions. These license conditions are (1) to increase the frequency of the drywell inspections and to monitor the two drywell trenches to ensure that the sources of water are identified and eliminated; (2) to ensure that the applicant fulfills its commitment to perform an engineering study prior to the period of extended operation to identify options to eliminate or reduce the leakage in the OCGS refueling cavity liner; and (3) to ensure that the applicant fulfills its commitment to perform a 3 dimensional finite-element analysis of the drywell shell prior to entering the period of extended operation.

3. Development of the TRACE Thermal-Hydraulic Code

The Committee met with representatives of the NRC staff concerning the development of the TRACE thermal-hydraulic system analysis code. This development effort is intended to establish a new state-of-the-art tool for the analysis of reactor thermal-hydraulic performance during transients and accidents. The staff expects this effort will result in the consolidation of the capabilities of four separate computer programs into one code. The Committee members expressed concern about the pace of the code development and the slow rate of introduction of the code into the NRC analytical community. They also noted that the code documentation is not complete and no outside peer-review of the code capability has been performed.

Committee Action

The Committee plans to consider a letter to the Executive Director for Operations on this matter during its March 8-10, 2007 meeting.

4. Proposed Revision to 10 CFR 50.46 LOCA Criteria for Fuel Cladding Materials

The Committee met with representatives of the NRC staff and its contractor, Argonne National Laboratory, and the Electric Power Research Institute (EPRI) regarding the development of the technical basis for a revision to the fuel cladding embrittlement criteria in 10 CFR 50.46. The objectives of this effort are to remove alloy-specific references in the regulation and establish a more performance-based standard that would not impede the introduction of new cladding materials. The staff described the experiments used to establish the underlying phenomena of fuel cladding embrittlement that are relevant to zirconium alloys containing tin and/or niobium during a LOCA. These experiments used samples from a number of operating reactors with different cladding materials and different geometries. These experiments also considered the effect of cladding performance as a function of fuel burnup because some early experiments had indicated that the current acceptance criteria might not be appropriate for high-burnup fuel. A representative from EPRI commented that the research results were impressive but additional experiments need to be performed with modern alloys such as Zirlo and M5 to confirm that the proposed technical criteria are appropriate for the newer materials.

Committee Action

The Committee plans to consider a letter to the Executive Director for Operations on this matter during a future ACRS meeting.

5. Draft Final Revision 1 of Regulatory Guide 1.189 (DG-1170), "Fire Protection for Nuclear Power Plants"

The Committee met with representatives of the NRC staff to discuss the draft final Revision 1 to Regulatory Guide 1.189 and resolution of public comments. In a letter dated November 17, 2006, the ACRS requested the opportunity to review the draft final version of this Guide after the resolution of public comments. Regulatory Guide 1.189 provides comprehensive guidance on the scope and depth of fire protection programs that the staff considers acceptable for existing and new plants. The staff noted that this technical guidance has also been incorporated into Standard Review Plan (SRP) Section 9.4.1, "Fire Protection Program."

Committee Action:

The Committee issued a letter to the Executive Director for Operations on this matter dated February 14, 2007, recommending that Revision 1 to Regulatory Guide 1.189 be issued.

6. Wolf Creek Pressurizer Weld Flaws

The Committee met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss the weld flaws discovered in the Wolf Creek pressurizer during an October 2006 inservice inspection. The staff presented an evaluation of the safety significance of these circumferential flaws and the implications to other plants with similar dissimilar metal butt welds. The staff also described the analysis of the growth of these flaws. The results of this analysis revealed that in some cases the flaws may lead to a rupture soon after they begin to leak. As a result, the staff has determined that the currently scheduled inspections or mitigations of these welds may need to be accelerated for some plants. NEI representatives provided the basis for the current schedule of inspections or mitigations of these flaws. They stated that the staff's analysis is extremely conservative and there is time between pipe leakage and rupture to allow plant personnel to take preventive actions.

Committee Action

This was an information briefing. No Committee action was necessary. The Committee plans to review the technical basis associated with the proposed NRC staff action for dealing with dissimilar metal butt weld issues during its March 8-10, 2007 meeting.

7. Proposed Revisions to Regulatory Guides and Standard Review Plan (SRP) Sections in Support of New Reactor Licensing

The Committee discussed "high-priority" SRP sections that are being revised or developed in support of new reactor licensing. The Committee noted that it is awaiting receipt of additional high priority SRP Sections from the staff.

Committee Action

The Committee plans to conduct an accelerated review of all Regulatory Guides and SRP Sections that it determines warrants ACRS review.

8. Subcommittee Report on Reliability and Probabilistic Risk Assessment

The Chairman of the Reliability and Probabilistic Risk Assessment (PRA) Subcommittee provided a report to the Committee summarizing the results of the December 14-15, 2006, meeting with the NRC staff and representatives of GE to discuss the PRA for the Economic Simplified Boiling Water Reactor that is in the design certification process. During the meeting, the Subcommittee reviewed several topics identified at a prior meeting, including the dominant accident sequences, the common cause failure method, the effects of thermal-hydraulic uncertainties on the PRA, the regulatory treatment of non-safety systems, and staff requests for additional information. The Subcommittee raised several issues to discuss at future meetings, and decided that no interim letter was necessary at this time. The next Subcommittee meeting will focus on the effects of thermal-hydraulic uncertainties on the PRA, the Level 2 PRA, and severe accident phenomena.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of January 11, 2007, to comments and recommendations included in the December 12, 2006, ACRS report on draft final Regulatory Guide DG-1145, "Combined License Applications for Nuclear Power Plants (LWR Edition)." The Committee decided that it was not satisfied with the EDO's response related to the Committee's recommendation that "the proposed final rule, 10 CFR Part 52, should include the requirements that a PRA be submitted with the design certification application and that a plant-specific PRA be submitted with the combined license (COL) application." The EDO's response articulated the staff's basis for deleting these requirements from the draft final Part 52 rule and stated that the Commission will decide on this matter when it votes on the final rule. The Committee reiterates its previous position with regard to including a requirement in 10 CFR Part 52 for submitting PRAs to the staff.

The staff committed to inform the ACRS of any significant changes to the final regulatory guide prior to publication.

- The Committee considered the EDO's response of January 19, 2007, to comments and recommendations included in the November 16, 2006 ACRS report on the proposed rulemaking to modify 10 CFR 50.46, "Risk-informed Changes to Loss-of-Coolant Accident Technical Requirements." The Committee decided that is was satisfied with the EDO's response.

  The staff committed to inform the Commission of the impact of the Committee's recommendations on its resources and schedule.

- The Committee considered the EDO's response of January 19, 2007, to comments and recommendations included in the December 15, 2006 ACRS letter on the proposed revision to SRP Section 13.3, "Emergency Planning." The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from December 9, 2006, through January 31, 2007, the following Subcommittee meetings were held:

- Reliability and Probabilistic Risk Assessment - December 14-15, 2006

The Subcommittee reviewed the PRA for the Economic Simplified Boiling Water Reactor.

- Power Uprates - January 16-17, 2007

The Subcommittee discussed the proposed 5-percent power uprate for the Browns Ferry Nuclear Plant, Unit 1.

- Plant License Renewal - January 18, 2007

The Subcommittee reviewed the license renewal application for the Oyster Creek Generating Station and the associated updated Safety Evaluation Report prepared by the NRC staff.

- Materials, Metallurgy, and Reactor Fuels - January 19, 2007

The Subcommittee discussed the proposed technical basis for revising the embrittlement criteria in 10 CFR 50.46.

- Planning and Procedures - January 31, 2007

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

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee would like to be kept informed of any significant changes made to the SRP Sections, prior to issuing them in final form, listed in the February 6, 2007 memorandum from Frank P. Gillespie, Executive Director, ACRS, to Luis A. Reyes, Executive Director for Operations, NRC.

- The Committee is awaiting receipt of additional high priority SRP Sections from the staff.

- The Committee plans to review the draft final version of Generic Letter 2007-XX, "Managing Gas Intrusion in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," during a future meeting.

- The Committee would like to be briefed by the staff on the results of the 3-dimensional finite element analysis of the Oyster Creek Generating Station drywell shell.

- The Committee plans to review the extended power uprate applications for Browns Ferry, Units 1, 2, and 3 during a future meeting.

- The Committee plans to review the technical basis associated with the proposed NRC staff actions for dealing with the dissimilar metal butt weld issue during its March 8-10, 2007 meeting.

- The Committee stated that granting of containment overpressure credit during long-term LOCA and 10 CFR Part 50 Appendix R fire scenarios at 120-percent of the original licensed thermal power for Browns Ferry Nuclear Plant Units 1, 2, and 3 will require support by more complete evaluations.

PROPOSED SCHEDULE FOR THE 540th ACRS MEETING

The Committee agreed to consider the following topics during the 540th ACRS meeting, to be held on March 8-10, 2007:

- Technical Basis Associated with the Proposed NRC Staff Action for Dealing with the Dissimilar Metal Butt Weld Issue (Open/Closed)
- Proposed Revisions to SRP Sections 15.0, "Accident Analysis - Introduction," and 15.9, "BWR Core Stability"
- Final Results of the Chemical Effects Head Loss Tests Related to the Resolution of the PWR Sump Performance Issues
- Technology Neutral Licensing Framework and Related Matters
- Proposed Revisions to Regulatory Guides and SRP Sections in Support of New Reactor Licensing

- Safeguards and Security Matters (Open/Closed)
- Proposed ACRS Report on the Development of the TRACE Thermal-Hydraulic System Analysis Code.

Sincerely,

/RA/

William J. Shack
Chairman

- Safeguards and Security Matters (Open/Closed)
- Proposed ACRS Report on the Development of the TRACE Thermal-Hydraulic System Analysis Code.

Sincerely,

/RA/

William J. Shack

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OFFICE	ACRS/ACNW	ACRS/ACNW
NAME	CSantos	FGillespie
DATE	02/26/07	02/26/07