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

SL-0054

October 2, 2007

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

Dear Chairman Klein:

SUBJECT: SUMMARY REPORT – 545th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, SEPTEMBER 6-8, 2007, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 545th meeting, September 6-8, 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:

- Development of a Technology-Neutral Regulatory Framework, dated September 26, 2007.
- Report on the Safety Aspects of the License Renewal Application for the Pilgrim Nuclear Power Station, dated September 26, 2007.

LETTER

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

- Proposed Recommendation for Resolving Generic Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment," September 26, 2007.

MEMORANDUM

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

- Draft Final Amendment to 10 CFR 50.55a, "Codes and Standards," and Revisions to Regulatory Guides Regarding ASME Code Cases, dated September 13, 2007.

HIGHLIGHTS OF KEY ISSUES

1. Final Review of the License Renewal Application for the Pilgrim Nuclear Power Station

The Committee met with the representatives of the Entergy Nuclear Operations, Inc. (Entergy, the applicant) and the NRC staff to discuss the license renewal application for the Pilgrim Nuclear Power Station (PNPS) and the associated final Safety Evaluation Report (SER). The operating license for PNPS expires on June 8, 2012. The applicant has requested approval for continued operation for a period of 20 years beyond the current license expiration date. The applicant stated that Pilgrim does not have the same scoping issues identified as the Vermont Yankee (VY) license renewal application because different scoping methodologies were used for PNPS. The applicant described the resolution of the open items related to the containment inservice inspection program, neutron fluence, and intrusion of groundwater into the torus room, as well as the actions taken and the commitments made to resolve these issues.

The applicant addressed the open item associated with neutron fluence by committing to complete the benchmarking of the code used in the fluence calculation. The applicant committed to submit a correctly benchmarked fluence calculation to the NRC on or before June 8, 2010, to confirm that the limiting fluence value will not be reached during the period of the extended operation. The staff is making the applicant's commitment a license condition to ensure adequate resolution of this issue. Regarding groundwater intrusion into the torus room, the applicant committed to enhance the structures monitoring program and perform periodic testing of the water for aggressiveness to concrete.

The staff described its review and inspection of the applicant's scoping, screening, and aging management programs; the program implementation at PNPS; and resolution of the open items. The staff also confirmed that the applicant has committed to follow the Generic Aging Lessons Learned Report, without exceptions, regarding monitoring of the cumulative usage factor for environmentally assisted fatigue. The staff stated that it plans to issue a supplemental SER to document this commitment.

Committee Action

The Committee issued a report to the NRC Chairman on this matter, dated September 26, 2007. The Committee concluded that the license conditions proposed by the staff are appropriate and recommended that the application of Entergy Nuclear Operations, Inc., for renewal of the operating license for PNPS be approved with the proposed license conditions.

2. Revisions to Standard Review Plan (SRP) Sections 19.0 and 19.2

The Committee met with representatives of the NRC staff to discuss revisions to SRP Sections 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," and 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance."

SRP Section 19.0 is associated with Regulatory Guide 1.206 (Combined License Applications for Nuclear Power Plants) and provides guidance to NRC staff reviewers for evaluating the content of combined license (COL) applications. The staff stated that design certification (DC) and COL applicants are required to submit a "description of their PRA and its results." The staff outlined the scope and level of detail that a COL applicant's probabilistic risk assessment (PRA) must meet. The staff also summarized the requirements for updating and upgrading COL holders' PRAs. The staff elaborated on what the description of an applicant's PRA should include and what results the staff expects to see in an applicant's submittal. The staff mentioned that additional guidance in several areas related to PRA is needed and it plans to issue interim staff guidance (ISG) to convey this additional guidance to industry.

SRP Section 19.2 is associated with Revision 1 to Regulatory Guide 1.174 (An Approach for Using Probabilistic Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis) and provides guidance to NRC staff reviewers for evaluating risk-informed changes to a plant's licensing basis. A member expressed concern that the modeling of digital instrumentation and control (I&C) systems in PRAs may not be adequate because the failure modes of digital I&C systems are not well understood. Another member expressed concern that the word "large" as used in the expressions "large early release frequency" (LERF) and "large release frequency" (LRF) is not well defined.

Committee Action

This was an information briefing. No Committee action was necessary.

3. Proposed Recommendation for Resolving Generic Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment"

The Committee met with representatives of the NRC staff to discuss the proposed recommendation for resolving Generic Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment." The staff described the history of the issue, its prioritization through the Generic Issues Program, the specific investigations performed for Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs), and the outcome of these analyses. This issue was relevant to reactors that were designed and licensed prior to the issuance of the first SRP. Investigations narrowed the focus of the possible effects of pipe whip and jet impingement inside containment to the possible breach of the containment shell in BWR Mark 1 plants and the possible failure of instrumentation and control systems in PWRs. More detailed and quantitative analyses showed that for the 51 plants that originally fell within the scope of this issue, the designs were satisfactory and that no further actions were required on the part of licensees. Therefore, staff is recommending that this issue be closed.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated September 26, 2007, concurring with the staff's recommendation that Generic Issue 156.6.1 be closed out and that no further actions on the part of the staff or licensees are necessary.

4. Status of NRR Activities in the Fire Protection Area

The Committee met with representatives of the NRR staff to discuss the ongoing NRC activities in the fire protection area. The staff described several major activities, including those associated with implementation of the National Fire Protection Association (NFPA) Standard 805, "Fire-Induced Multiple Spurious Actuations and Manual Operator Actions." The staff discussed the plants that are transitioning to the NFPA 805 Standard and the lessons learned, as well as the status of industry guidance development for fire modeling. The staff also provided its views on the Nuclear Energy Institute multiple spurious actuation methodology and recent interaction with industry in addressing this issue. In addition, the staff discussed post-fire manual operator actions and recent staff guidance for addressing this issue. The staff also provided an update on the Hemyc and MT fire barrier issue, and the industry progress in addressing Generic Letter 2006-03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations."

Committee Action

This was an information briefing. No Committee action was necessary.

5. Subcommittee Report on Plant License Renewal

The Chairman of the Plant License Renewal Subcommittee provided a report to the Committee summarizing the results of the September 5, 2007, meeting with the NRC staff and representatives of Entergy to review the draft SER with Open Items related to the license renewal application for the James A. Fitzpatrick Nuclear Power Plant. The current operating license expires on October 17, 2014. Entergy submitted the license renewal application on July 31, 2006. The staff's draft SER was issued on July 31, 2006, and contains two open items and no confirmatory items. The two open items are related to reactor vessel neutron fluence and environmentally assisted fatigue. For determining reactor vessel neutron fluence, the staff finds that the projected fluence values are unacceptable. Entergy stated that it will submit a new fluence calculation to the staff for review. Entergy also stated that it will demonstrate that cumulative usage factors (CUF) of the most fatigue sensitive locations are less than 1.0 throughout the license renewal period, and it will submit the results of the CUF calculations to the staff for review and approval. Other discussion topics included drywell and torus monitoring, and torus repair.

Committee Action

The Committee plans to discuss the final SER related to the license renewal application for the James A. Fitzpatrick Nuclear Power Plant in a future meeting.

6. Draft Report on Quality Assessment of Selected NRC Research Projects

The Committee was briefed by the members of the ACRS panels regarding the results of their assessment of the quality of the NRC research projects on Cable Response to Live Fire (CAROLFIRE) Testing, Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping, and Technical Review of the Online Monitoring Techniques for Performance Assessment.

Committee Action

The Committee plans to complete a final report on the results of its assessment of the quality of the above NRC research projects during its October 2007 meeting.

7. Draft ACRS Report on the NRC Safety Research Program

The ACRS provides the Commission a biennial report that presents the Committee's observations and recommendations concerning the overall NRC Safety Research Program. During the September 2007 meeting, the Committee was briefed by the lead members of ACRS regarding the status of their evaluation of research activities in specific technical disciplines.

Committee Action

The Committee plans to continue its discussion of the draft ACRS report on the NRC Safety Research Program during its October 2007 meeting.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of August 20, 2007, to comments and conclusions included in the July 24, 2007, ACRS report concerning the staff's approach to verifying the closure of inspections, tests, analyses, and acceptance criteria (ITAAC) through a sample-based inspection program. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of July 30, 2007, to comments and recommendations included in the June 22, 2007, ACRS letter concerning General Electric (GE) Licensing Topical Reports on Maximum Extended Load Line Limit Analysis Plus (MELLLA+) and Applicability of GE Methods to Expanded Operating Domains. The Committee decided that it was satisfied with the EDO's response. In its response, the EDO committed to the following:

  - The modifications identified during ACRS discussions associated with anticipated transients without scram (ATWS) instability will be included in the final Safety Evaluation prepared by the staff.

  - The Committee will be provided the opportunity to review the first few plant-specific MELLLA+ applications, and any significant changes in the final Safety Evaluation prepared by the staff, including any changes to the limitations. The Committee will be provided the opportunity to review any future significant changes to the limitations currently applied to the safety limit already defined in the minimum critical power ratio (MCPR), the operating limit MCPR, and on bypass voiding.

- The Committee considered the EDO's response of August 29, 2007, to conclusions and recommendations included in the July 27, 2007, ACRS letter on draft NUREG/CR, titled, "Review of NUREG-0654, Supplement 3, 'Protective Action Recommendations for Severe Accidents'." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of July 23, 2007, to comments and recommendations included in the June 18, 2007, ACRS letter concerning the final draft NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of July 11, 2007, to comments and recommendations included in the May 23, 2007, ACRS letter regarding proposed technical basis for the revision to 10 CFR 50.46 LOCA embrittlement criteria for fuel cladding materials. The Committee decided to discuss this matter during a future meeting.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

The following Subcommittee meetings were held during the period from July 13, 2007, through September 5, 2007:

- Plant Operations – August 14 – 16, 2007

The Subcommittee visited NRC Region IV offices and the San Onofre Nuclear Generating Station (SONGS) to discuss plant operations issues.

- Planning and Procedures — September 5, 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.

- Plant License Renewal - September 5, 2007

The Subcommittee reviewed the license renewal application and the associated NRC staff's Safety Evaluation Report with Open Items for the James A. Fitzpatrick Nuclear Power Plant.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee would like the opportunity to review the AP1000 technical reports related to PRA (and associated draft safety evaluation) that will form the basis, in part, for anticipated amendment to the AP1000 certified design.

- The Committee plans to discuss the final draft report on the results of its assessment of the quality of the selected NRC research projects during its October 2007 meeting.

- The Committee plans to continue discussion on its draft report on the NRC Safety Research Program during its October 2007 meeting.

- The Committee plans to discuss the final SER related to the license renewal application for the James A. Fitzpatrick Nuclear Power Plant during a future meeting.

- The Committee plans to discuss the report on the proposed technical basis for the revision to 10 CFR 50.46 LOCA embrittlement criteria for fuel cladding materials during a future meeting.

PROPOSED SCHEDULE FOR THE 546th ACRS MEETING

The Committee agreed to consider the following topics during the 546th ACRS meeting, to be held on October 4-6, 2007:

- Digital I&C Project Plan and Interim Staff Guidance
- Draft final Generic Letter 2007-XX, “Managing Gas Intrusion in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems”
- Dissimilar Metal Weld Issue
- Draft ACRS Report on the NRC Safety Research Program
- Draft final Report on Quality Assessment of Selected NRC Research Projects
- Meeting with NEI, EPRI, and INPO to discuss Industry Activities

Sincerely,

/RA/

William J. Shack
Chairman

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William J. Shack
Chairman

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