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

March 31, 2009

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

SUBJECT: SUMMARY REPORT – 560th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, MARCH 5-7, 2009, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

Dear Chairman Klein:

During its 560th meeting, March 5-7, 2009, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report, letters, and memoranda:

REPORT

Report to Dale E. Klein, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Draft Final Regulatory Guide 5.71, "Cyber Security Programs For Nuclear Facilities," dated March 19, 2009

LETTERS

Letters to R. W. Borchardt, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Draft Final Rule 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," dated March 13, 2009

- Draft Final Regulatory Guide 5.73, "Fatigue Management for Nuclear Power Plant Personnel," dated March 19, 2009

- Crediting Containment Overpressure in Meeting the Net Positive Suction Head Required to Demonstrate that the Safety Systems can Mitigate the Accidents as Designed, dated March 18, 2009

MEMORANDA

Memoranda to R. W. Borchardt, Executive Director for Operations, NRC, from Edwin M. Hackett, Executive Director, ACRS:

- Draft Final Revision 3 to Regulatory Guide 10.4, dated March 12, 2009

- Proposed Revision to Regulatory Guide 1.141 (DG-1213), dated March 12, 2009
- Proposed Revisions to Regulatory Guide 1.205 (DG-1218) and Standard Review Plan Section 9.5.1.2, dated March 12, 2009

HIGHLIGHTS OF KEY ISSUES

1. Draft Final Regulatory Guide 5.71 (formerly DG–5022), "Cyber Security Programs for Nuclear Facilities"

The Committee met with representatives of the NRC staff to discuss the Draft Final Regulatory Guide (RG) 5.71, "Cyber Security Programs for Nuclear Facilities," and NRC staff's resolution of stakeholders' comments. 10 CFR 73.54 establishes performance-based requirements to ensure that the functions of critical systems and critical digital assets are protected from cyber attack using a graded approach. RG 5.71 conveys NRC staff positions for developing a program that provides an effective protection mechanism against cyber attacks. It also provides NRC staff positions regarding the minimum set of elements needed within a program to protect a facility, the networks within it, the plant systems, the digital assets that implement system functions, and operating systems and applications within those digital assets that accomplish the functions to be protected. A generic cyber security plan template, NEI-08-09, is also being jointly-developed between staff and industry for later consideration as a reference in RG 5.71.

Committee Action

The Committee issued a report to the NRC Chairman, dated March 19, 2009, recommending that RG 5.71 not be published until it is revised to: (i) provide a reference Digital I&C (DI&C) computer, communication, and network security framework that identifies assets, associated plant functions, vulnerabilities, and interaction and access pathways; (ii) include examples and more specific guidance on how the stated requirements can be met; (iii) ensure that the guidance distinguishes between DI&C system and non-real-time information technology system architectures; and (iv) address the issues of threat assessment, dependency analysis, and the use of probabilistic risk assessment (PRA).

2. Draft Final Rule 10 CFR 50.61a, "Alternative Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events"

The Committee met with representatives of the NRC staff to discuss Draft Final rule 10 CFR 50.61a, "Alternative Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." The discussion on the technical basis of the rule centered on the studies used to show that the rule was generally applicable across the entire fleet of pressurized water reactors (PWRs). The screening limits in the proposed rule were developed through detailed studies of three PWRs: Beaver Valley, Oconee, and Palisades. The generalization studies then considered five additional plants in order to infer the broader applicability of the rule. The events that offered the most challenge to vessel were found to be driven by factors that are similar across the fleet.

The main features of 10 CFR 50.61a include: limitations on applicability; less restrictive screening criteria; evaluation of plant-specific flaw distributions; and implementation of new embrittlement models and surveillance data evaluations. The staff's conclusion was that 10 CFR 50.61a is appropriate for providing protection of public health and safety and for reducing unnecessary regulatory burden.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated March 13, 2009, recommending that the rule be approved and that the staff verify and document the capability of nondestructive examination procedures used to characterize flaw distributions in reactor vessels. The Committee also concluded that an effort is needed to plan for the most effective use of surveillance samples in tracking embrittlement trends.

3. Draft Final Regulatory Guide 1.200 (formerly DG-1200), "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"

The Committee met with representatives of the NRC staff to discuss the Draft Final Revision 2 of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." Revision 2 of Regulatory Guide 1.200 endorses a national consensus standard, jointly developed by the American Nuclear Society (ANS) and the American Society of Mechanical Engineers (ASME), regarding the content of a technically acceptable PRA. It also endorses PRA peer review guidance developed by the Nuclear Energy Institute (NEI). The staff discussed the resolution of public comments and the major outstanding industry issues. One of the industry comments is regarding independence of the peer reviewers. Industry claims that achieving the level of independence recommended by this guide could be a problem because of a lack of PRA experts. The staff also discussed the future work on standard development that would require future revisions to RG 1.200. The ACRS members' questions were related to alternative risk metrics, non-light water reactor PRA standards, and handling of uncertainty.

Committee Action

The Committee plans to issue a letter on this matter during its April 2-4, 2009, meeting.

4. Draft Final Regulatory Guide 5.73 (formerly DG-5026), "Fatigue Management for Nuclear Power Plant Personnel"

The Committee met with representatives of the NRC staff, the Professional Reactor Operator Society (PROS), the Nuclear Energy Institute (NEI), and the International Brotherhood of Electrical Workers (IBEW) to discuss Draft Final Regulatory Guide 5.73, "Fatigue Management for Nuclear Power Plant Personnel." RG 5.73 endorses NEI 06-11, Revision 1, "Managing Personnel Fatigue at Nuclear Power Reactor Sites," with certain exceptions. The first exception relates to periodic overtime. The staff feels that allowing overtime on the day off could lead to rule violations, and that there is already sufficient flexibility to make this a rare situation. The second exception relates to outage activities at multi-unit sites. The staff disagrees with the NEI position that licensed operators responsible for an operating unit (and the operators who provide relief for them) can be considered to be "working on outage activities."

A representative of PROS stated that RG 5.73 will make utilities change the way they schedule staff on the front and back end of an outage, possibly having a negative impact on their ability to safely execute an outage. PROS recommended that instead of defining an outage as having a reactor unit disconnected from the grid, an outage be defined as the time period beginning one week prior to disconnecting the unit from the grid and ending when the unit is reconnected and achieves 75 percent power. This would require a rule change as well as a change to the regulatory guide. The PROS representative also recommended that for multi-unit sites, the concept of “outage unit” be replaced with a concept of “outage site.” This would allow multiple unit facilities with combined control rooms to modify all site personnel schedules to accommodate the outage. The advantages of this include a simpler set of work rules and more opportunity to keep work teams intact over the outage.

A representative of NEI discussed industry implementation of the fatigue management rule and stated that the regulatory guide’s exceptions to NEI 06-11 are not necessary. It is not clear what problem is being addressed by the exception to outage activities. Regarding overtime, the NRC staff position adds to the complexity of schedule changes and would result in a significant distraction for first-line supervision.

A representative of IBEW agreed with the PROS proposal on the definition of outages. Regarding the issue of periodic overtime, IBEW supports the NEI position. Regarding multi-unit outages, IBEW is still working to address the issue of multi-unit outages.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated, March 19, 2009, recommending that Draft Final Regulatory Guide 5.73 be issued as final. The Committee also recommended that the staff closely track industry pilot applications and confirm that practical scheduling and time monitoring programs achieve the desired fatigue management objectives within an integrated framework that maintains stable shift manning and workforce controls throughout all plant operating modes.

5. International Human Reliability Analysis (HRA) Empirical Study

The Committee met with representatives of the NRC staff, Sandia National laboratories (SNL), the Organization for Economic Cooperation and Development Halden Reactor Project (HRP), and Paul Scherrer Institute (PSI) to discuss International Human Reliability Analysis (HRA) Empirical Study. The NRC staff described the benchmark study of HRA methods using control room simulator data; the NRC and industry efforts to improve HRA guidance; interactions with national and international experts to pursue testing and benchmarking of HRA methods; and further support for Halden Reactor Project initiatives. The NRC staff also discussed the value of simulator exercises in providing insights on issues related to human behavior that can be used to improve HRA methods.

Representatives of HRP discussed the new data analysis approach that was developed to optimize the comparison of HRA methods predictions to empirical observations and improve the usefulness of simulator data for HRA purposes. They also discussed experimental work that identified the extent of crew-to-crew variability, the significance of teamwork factors, and the importance of event dynamics in determining crew performance.

The representative of SNL discussed the qualitative analysis of HRA methods and its comparison with Halden crew data. All HRA methods identified some of the important factors that affected crew performance, but most HRA analyses failed to identify important factors for some Human Failure Events (HFEs). Additionally, several methods have significantly over- or underestimated the difficulty of some HFEs.

The representative of PSI discussed the quantitative comparisons between HRA predictions and the empirical data. There were significant limitations on the quantitative results. The quantitative comparisons supplement the qualitative comparisons and insights. The overall evaluation of the HRA methods is based on both qualitative and quantitative insights; however, the qualitative insights should be weighted more strongly.

Committee Action

This was an information briefing and no Committee action was necessary at this time.

6. Plant License Renewal Subcommittee Report (Indian Point License Renewal Application)

The Chairman of the Plant License Renewal Subcommittee provided a report to the Committee summarizing the results of the March 4, 2009, meeting with the NRC staff and representatives of Entergy Nuclear Operations Inc. (Entergy) to review the Draft Safety Evaluation Report (SER) with Open Items related to the license renewal application for the Indian Point Nuclear Generating Station, Units 2 (IP2) and 3 (IP3).

The current operating licenses for the IP2 and IP3 expire on September 28, 2013, and December 12, 2015, respectively. Entergy submitted the license renewal application on April 23, 2007. The staff's draft SER, issued in January 2009 contained 20 open items. During the meeting, Entergy described the plant, its operating history, the license renewal review methodology, the aging management programs, and its commitment tracking system. The staff discussed the open items and stated that 13 out of the 20 open items have been closed. The staff and Entergy are in the process of resolving the remaining seven open items.

A public interest group, Riverkeeper, provided written comments regarding its concerns on the Indian Point license renewal and made oral statements during the meeting.

Committee Action

The Committee plans to review the final SER related to the license renewal application for the Indian Point Nuclear Generating Station, Units 2 and 3 in September 2009.

7. US-APWR Subcommittee Report

The Chairman of the US-APWR Subcommittee provided a report to the Committee summarizing the results of the February 18-20, 2009, site visits to Westinghouse offices in Monroeville, PA, and Mitsubishi Electric Power Products, Inc. (MEPPI) facilities in Cranberry Township, PA. On February 18, 2009, the Subcommittee participated in a tour and demonstration of the Westinghouse AP1000 digital control room simulator. On February 19, 2009, the Subcommittee met with representatives of the NRC staff and MEPPI to discuss three topical reports related to large-break loss-of-coolant-accident (LOCA), small-break LOCA, and non-LOCA methodologies associated with the US-APWR Design. On February 20, 2009, the subcommittee participated in a tour and demonstration of the Mitsubishi APWR digital control room simulator.

Committee Action

The Committee plans to continue its review of the topical reports and draft SERs related to the US-APWR design certification in future meetings.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of January 23, 2009, to conclusions and recommendations included in the December 18, 2008, ACRS letter on the technical basis and rulemaking strategy for the revision of 10 CFR 50.46(b), "Loss of Coolant Accident Embrittlement Criteria for Fuel Cladding Materials." The Committee decided that it was satisfied with the EDO's response.

PROPOSED SCHEDULE FOR THE 561st ACRS MEETING

The following topics are scheduled for the 561st ACRS meeting to be held on April 2-4, 2009:

- License Renewal Application for the Vogtle Nuclear Plant

- Digital I&C Interim Staff Guidances on Highly Integrated Control Room - Human Factors and Licensing Process Issues

- License Renewal Application for the National Institute of Standards and Technology Reactor

- Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants"

- Risk Metrics for New Light-Water Reactor Risk-Informed Application

- Proposed ACRS Report on Revision 2 to Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"

Sincerely,

/RA/

Mario V. Bonaca
Chairman

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Chairman

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