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

February 3, 2011

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: SUMMARY REPORT – 579th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, JANUARY 13-15, 2011

Dear Chairman Jaczko:

During its 579th meeting, January 13-15, 2011, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports and letters:

REPORTS

Report to Gregory B. Jaczko, Chairman, NRC, from Said Abdel-Khalik, Chairman, ACRS:

- Report on the Safety Aspects of the Aircraft Impact Assessment for the Westinghouse Electric Company AP1000 Design Certification Amendment Application, dated January 19, 2011

Report to Gregory B. Jaczko, Chairman, NRC, from J. Sam Armijo, Vice-Chairman, ACRS:

- Report on the Safety Aspects of the Southern Nuclear Operating Company Combined License Application for Vogtle Electric Generating Plant, Units 3 and 4, dated January 24, 2011

LETTERS

Letters to R. W. Borchardt, Executive Director for Operations, NRC, from Said Abdel-Khalik, Chairman, ACRS:

- Draft Final Revision 2 to Regulatory Guide 1.174 and Revision 1 to Regulatory Guide 1.177, dated January 24, 2011

- Draft Final Rule, "Enhancements to Emergency Preparedness," and Related Regulatory Guidance Documents, dated January 24, 2011

- Review of RAMONA5-FA for Use in BWR Stability Calculations, dated January 31, 2011

HIGHLIGHTS OF KEY ISSUES

1. Aircraft Impact Assessment for the Revised AP1000 Design

The Committee met with representatives of the NRC staff and Westinghouse Electric Company (WEC) to discuss the AP1000 Aircraft Impact Assessment (AIA). The results of the AP1000 AIA are a part of the AP1000 Design Certification Amendment (DCA) application. As required by 10 CFR 50.150, applicants for new nuclear power plants must perform an assessment of the effects of the impact of a large commercial aircraft. Using realistic analyses, applicants must identify and incorporate into the facility those design features and functional capabilities needed to show that, with reduced use of operator action (1) the reactor core remains cooled or the containment remains intact and (2) spent fuel cooling or spent fuel pool integrity is maintained. WEC representatives presented the AIA results and concluded that the assessments satisfy the NRC requirements. The assessments were performed using the guidance in NEI 07-13, Revision 7, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs." WEC representatives also addressed ACRS subcommittee meeting follow-up items associated with additional impact locations and the effects of the shield plate dropping on the containment vessel. The staff performed an inspection of the AIA using NRC Inspection Procedure 37804. Both WEC and the staff presented the staff's inspection findings. The inspection revealed that WEC did not use realistic analyses for certain aspects of its AIA and did not fully identify and incorporate into the design control document those design features and functional capabilities credited. The resolution of the inspection findings were presented during the meeting.

Committee Action

The Committee issued a letter to the NRC Chairman on this matter dated January 19, 2011, concluding that the WEC AIA for the design described in the AP1000 DCA application, as modified to resolve NRC inspection findings, complies with the requirements of 10 CFR 50.150. Analyses show that the containment remains intact following the impact of a large commercial aircraft. The reactor core remains cooled, and spent fuel pool integrity is maintained. The Committee also recommended that the staff evaluate information and analyses presented to the ACRS, but not subjected to staff review or inspection, to determine if there is a need for further revision of the design control document, or a need for further inspections.

2. Final Safety Evaluation Report Associated with the Vogtle Units 3 and 4 Combined License Application

The Committee met with representatives of the NRC staff, Southern Nuclear Operating Company (SNC), and two members of the public to discuss the Combined License Application (COLA) for the Vogtle Electric Generating Plant (VEGP), Units 3 and 4. This COLA

incorporates by reference the Westinghouse Electric Company AP1000 Design Certification Amendment application and SNC VEGP Early Site Permit (ESP). SNC representatives described highlights of the COLA including: departures from the AP1000 DCD; exemptions from the regulations; ESP combined license items; the resolution of open items identified by the NRC staff; and plant-specific inspection, test, analysis and acceptance criteria (ITAAC) items. SNC representatives also addressed the following technical questions which were raised during the ACRS AP1000 subcommittee meetings: containment vessel cleanliness program, containment interior debris limitation, in-service inspection/in-service testing (ISI/IST) program requirement for Automatic Depressurization System (ADS-4) squib valves, and VEGP plant-specific seismic margin analyses. The members of the public commented that the potential for corrosion or cracking in the AP1000 steel containment structure should be carefully evaluated.

Committee Action

The Committee issued a letter to the NRC Chairman on this matter dated January 24, 2011, concluding that there is reasonable assurance that VEGP, Units 3 and 4 can be built and operated without undue risk to the health and safety of the public. The SNC COLA for VEGP, Units 3 and 4 should be approved following its final revision. The Committee recommended that containment interior cleanliness limits on debris be included in the Technical Specifications, a requirement on the development of an ISI/IST program for squib valves be established, and a requirement to assure the accuracy of feedwater flow measurements be established. The Committee also recommended that the staff review with the ACRS the changes in design or commitments that are not yet incorporated in the COLA or referenced in the design control document, which significantly deviate from those presented during the ACRS review.

3. Draft Final Revision 2 to Regulatory Guide (RG)1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and Draft Final Revision 1 to RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications"

The Committee met with representatives of the NRC staff to discuss the proposed changes to RG 1.174 and RG 1.177. The staff's presentation described the proposed changes to these RGs, the resolution of public comments, and items not considered as part of these revisions. The terminology in the RGs was revised to be consistent with Revision 2 of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and the 2009 American Nuclear Society/American Society of Mechanical Engineers probabilistic risk assessment (PRA) standard RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications." The changes also included updating the discussion of uncertainty to incorporate NUREG-1855, "Guidance on Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," and removing outdated discussion topics. A paragraph stating that changes

in risk that are not captured by core damage frequency (CDF) or large early release frequency (LERF) should be addressed qualitatively as part of defense-in-depth was added to the draft RG that was issued for public comment, but was subsequently removed based on public comments and further staff consideration. The staff stated that the proposed revisions do not address risk metrics for new (advanced light-water) reactors because they are waiting for Commission guidance regarding SECY-10-0121, "Modifying the Risk-Informed Regulatory Guidance for New Reactors." The proposed revisions do not incorporate safety/security interface guidance, which is currently under development by the Office of Nuclear Reactor Regulation.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter dated January 24, 2011, recommending that RG 1.177 be issued as final. The Committee recommended that RG 1.174 be revised to reinstate guidance on the consideration of late containment failure before being issued as final. The Committee also recommended that the staff continue to investigate approaches for addressing the interfaces between measures taken for safety and measures taken for security, and to identify revisions and adaptations that might be required for new reactors.

4. Draft Final Rule and Regulatory Guidance Regarding Enhancements to Emergency Preparedness Regulations

The Committee met with representatives of the NRC staff to discuss the draft Final Rule, "Enhancements to Emergency Preparedness," and related regulatory guidance documents. The draft Final Rule proposes to amend certain Emergency Preparedness (EP) requirements in 10 CFR Parts 50 and 52, and related guidance documents to codify the EP related security improvements previously made through NRC Orders and Bulletin, in response to the September 11, 2001, incident. The staff's presentation identified 12 high priority EP issues and discussed how each was addressed. Six of these issues are security related. The staff described the regulatory guidance documents associated with this rulemaking: Regulatory Guide 1.219, "Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors;" NUREG/CR-7002, "Criteria for Development of Evacuation time Estimate Studies;" and Interim Staff Guidance (ISG) NSIR/DPR-ISG-01, "Interim Staff Guidance on Emergency Planning for Nuclear Power Plants." Finally, the staff described the comment resolution process and discussed how some of the public comments were addressed.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter dated January 24, 2011, recommending that the draft final EP rule and the associated RG 1.219 not be issued until the NRC staff resolves the issues associated with the location and sharing of an Emergency Operations Facility by several nuclear power plants. The Committee also

recommended that in future revisions of the rule and associated guidance documents, the NRC staff should consider: (a) expanding NUREG/CR-7002 to include evacuation time estimates during conditions of external environmental duress, such as seismic events, extreme weather conditions, or terrorist activity external to the site and (b) developing an approach to risk-inform emergency classifications and emergency action recommendations using site-specific PRA and insights from other severe accident studies.

5. Staff Assessment of the RAMONA5-FA Code

The Committee met with representatives of the NRC staff, AREVA, and a member of the public to discuss Topical Report EMF-3028(P), "RAMONA5-FA, A Computer Program for BWR Transient Analysis in the Time Domain." The NRC staff presented its safety findings for the application of RAMONA5-FA for AREVA's BWR power oscillation detect and suppress calculations, using the DIVOM methodology. DIVOM is an acronym for **D**elta CPR (critical power ratio) over **I**nitial CPR **V**ersus **O**scillation **M**agnitude. It correlates the loss in CPR in the hot channel corresponding to the power oscillation amplitude measured by the oscillation power range monitor (OPRM). The DIVOM correlation is used to define the OPRM amplitude scram setpoint for the long term stability solutions. At expanded operating domains, AREVA uses an enhanced method, which also relies on the DIVOM methodology, but includes additional features to preclude instabilities. Currently, a 10 percent penalty is applied to the DIVOM correlation until the adequacy and performance of RAMONA5-FA to calculate the DIVOM correlation at expanded operating domains is reviewed. The staff described its evaluations of the RAMONA5-FA code predictions against plant data and loop test data. In addition, the staff provided evaluations of the impact of void fraction uncertainties on the DIVOM correlations. The staff's review and approval was limited to the narrow scope of using RAMONA5-FA in generating the DIVOM correlations and not for transients, special events, or accidents.

The Committee and its consultant identified numerous documentation errors in the RAMONA5-FA theory manual (Topical Report EMF-3028(P) Volume 2). A list of errors in the topical report was provided to the staff. Prior to our meeting, AREVA prepared a revised version of the RAMONA5-FA theory manual and stated that a subsequent review of the RAMONA5-FA source code indicated that the documentation errors in the theory manual had not been introduced into the code.

A member of the public commented on the Rod Bundle Heat Transfer Test Program conducted at the Pennsylvania State University and a petitioner's request to revise 10 CFR 50.46 requirements.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter dated January 31, 2011, concluding that the staff's recommendation to remove the 10 percent penalty on the DIVOM correlation slope calculated using RAMONA5-FA for extended flow window

operating domains is acceptable subject to the satisfactory resolution of the following recommendation: the staff should review Volume 2 of the revised RAMONA5-FA Topical Report EMF-3028(P), to ensure that all errors have been corrected and that the documentation errors do not reflect errors in the source code.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of November 26, 2010, to conclusions and recommendations included in the October 26, 2010, ACRS report on the safety aspects of the license renewal application for the Duane Arnold Energy Center. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of December 21, 2010, to conclusions and recommendations included in the November 17, 2010, ACRS letter on the draft final revisions to generic license renewal guidance documents. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of December 10, 2010, to conclusions and recommendations included in the August 9, 2010, ACRS report on the closure of design acceptance criteria for new reactors. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of December 17, 2010, to conclusions and recommendations included in the November 16, 2010, ACRS letter on the standard review plan for renewal of spent fuel dry cask storage system licenses and certificates of compliance (NUREG-1927). The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of November 19, 2010, to conclusions and recommendations included in the October 20, 2010, ACRS report on the draft final rule for risk-informed changes of loss-of-coolant accident technical requirements (10 CFR 50.46a). The Committee decided that it was satisfied with the EDO's response.

SCHEDULED TOPICS FOR THE 580th ACRS MEETING

The following topics are scheduled for the 580th ACRS meeting, to be held on February 10-12, 2011:

- Final Safety Evaluation Report Associated with the License Renewal Application for the Palo Verde Nuclear Generating Station

- Final Safety Evaluation Report Associated with the Virgil C. Summer Units 2 and 3 Combined License Application

- Comparison of Integrated Safety Analyses (ISAs) for Fuel Cycle Facilities and Probabilistic Risk Assessments (PRAs)

- Current State of Licensee Efforts to Transition to National Fire Protection Association (NFPA)-805

- Draft Final Regulatory Guide (RG)1.34, “Control of Electroslag Weld Properties;" RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components;" RG 1.44, "Control of the Processing and Use of Stainless Steel;" and RG 1.50, "Control of the Preheat Temperature for Welding of Low-Alloy Steel"

- Commission Paper on the Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

- Final Safety Evaluation Report Associated with the Virgil C. Summer Units 2 and 3 Combined License Application

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/RA/

Said Abdel-Khalik
Chairman

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Letter to the Honorable Gregory B Jaczko, Chairman, NRC, from Said Abdel-Khalik, Chairman, ACRS, dated February 3, 2011

SUBJECT: SUMMARY REPORT – 579th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, January 13-15, 2011

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