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

May 3, 2013

The Honorable Allison M. Macfarlane
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: SUMMARY REPORT – 603rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, APRIL 11-12, 2013

Dear Chairman Macfarlane:

During its 603rd meeting, April 11-12, 2013, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following letters and memoranda:

<u>LETTERS</u>

Letters to R. W. Borchardt, Executive Director for Operations, NRC, from J. Sam Armijo, Chairman, ACRS:

Chapters 4, 15, 17, and 19 of the Safety Evaluation Report with Open Items for Certification of the US-APWR Design and Safety Evaluations of Supporting Topical Reports, dated April 29, 2013

- Chapters 4, 13, 15, 16, 17, and 19 of the Safety Evaluation Report with Open Items for the Comanche Peak Nuclear Power Plant, Units 3 and 4, US-APWR Reference Combined License Application, dated April 26, 2013

- WCAP-17116-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 5 - Application to ABWR," dated May 2, 2013

<u>MEMORANDA</u>

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

- Proposed Regulatory Guide DG-7009, dated April 15, 2013

- Draft Final Regulatory Guide 1.68, dated April 15, 2013

HIGHLIGHTS OF KEY ISSUES

1. Selected Chapters of the Safety Evaluation Reports (SERs) with Open Items Associated with the US-APWR Design Certification Application and the Comanche Peak Combined License Application (COLA)

The Committee met with representatives of Mitsubishi Heavy Industries, LTD. (MHI), Luminant Generation Company (Luminant), and the NRC staff to review the following chapters of the SER related to the US-APWR Design Certification Document (DCD): Chapter 4, “Reactor;” Chapter 15, “Transient and Accident Analyses;” Chapter 17, “Quality Assurance and Reliability Assurance;” and Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation.” The Committee also reviewed the SERs related to the following supporting Topical Reports: MUAP-07008, “Mitsubishi Fuel Design Criteria and Methodology;” MUAP-07009, “Mitsubishi Thermal Design Methodology;” MUAP-07010, “Non-LOCA Methodology;” MUAP-07011, “Large Break LOCA Code Applicability Report for US-APWR;” and MUAP-07013, “Small Break LOCA Methodology for US-APWR.” Representatives of MHI and the NRC staff presented a summary of the major technical issues associated with these SER chapters. The main issues discussed for the US-APWR design certification were the adequacy of the US-APWR probabilistic risk assessment (PRA) to support plant-specific risk-informed applications and operational programs, and the MHI thermal-hydraulic models used for PRA success criteria.

The Committee also reviewed the following chapters of the SER with open items associated with the Comanche Peak reference COLA: Chapter 4, “Reactor;” Chapter 13, “Conduct of Operations;” Chapter 15, “Transient and Accident Analyses;” Chapter 16, “Technical Specifications;” Chapter 17, “Quality Assurance and Reliability Assurance;” and Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation.” Representatives of Luminant and the NRC staff presented a summary of the major technical issues associated with these SER chapters. The main issues discussed for the Comanche Peak reference COLA were the minimum shift staffing at Comanche Peak Nuclear Power Plant, and comprehensive audit and independent peer review of the Comanche Peak PRA.

Committee Action

The Committee issued two letters to the Executive Director for Operations on these matters. In the first letter dated April 29, 2013, on the SER related to DCD Chapters 4, 15, 17, 19, and the supporting Topical Reports, the Committee did not identify any additional issues that would preclude certification of the US-APWR design. The Committee did identify two items that merit additional attention from the staff: (i) substantial technical improvements to the PRA would be needed to support plant-specific risk-informed applications and operational programs and (ii) the staff should ensure that the MHI thermal-hydraulic models used for PRA success criteria and the initial stages of severe accident progression are benchmarked against values from an NRC-approved thermal-hydraulic code. Separate from the US-APWR design certification review, the Committee will examine the adequacy of current regulatory guidance, analysis methods, and industry practices to evaluate and manage the risk from the pellet cladding interaction (PCI) failure mechanism for PWR fuel during Anticipated Operational Occurrences (AOOs).

In the second letter dated April 26, 2013, on Chapters 4, 13, 15, 16, 17, and 19 of the SER for the Comanche Peak reference COLA, the Committee did not identify any additional issues that would preclude issuance of the COLA. The Committee recommended: (i) the staff evaluate whether the planned minimum shift crew composition with one Shift Technical Advisor and one Radiation Protection Technician shared between Units 3 and 4 is adequate to effectively manage the response to site-wide events that affect both units and (ii) the staff conduct a comprehensive audit to confirm that all technical elements of the full-scope, plant-specific PRA that is required before fuel load have received an independent peer review before that PRA is used to support any risk-informed licensing applications and operational programs.

For both the design certification and reference COLA, the Committee plans to review the staff's resolution of the open items and will comment on safety implications of any system interactions in future interim letters and final report.

2. WCAP-17116-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 5 - Application to ABWR"

The Committee met with representatives of the NRC staff, Westinghouse, and Nuclear Innovation North America, LLC (the applicant for South Texas Project, Units 3 and 4) to discuss the licensing topical report, WCAP-17116-P, and the associated NRC staff safety evaluation. Westinghouse requested extension of the current NRC-approved Westinghouse BWR loss of coolant accident (LOCA) methodology and suite of codes to the ABWR design. Westinghouse discussed (1) the unique features of the ABWR design relative to the BWR3 through 6 designs, (2) qualification data supporting the application to the ABWR plant design, and (3) ABWR plant response to a LOCA coincident with a loss of offsite power. The applicant stated that the ABWR design improvements such as the internal recirculation pumps (located in the annulus region) and the elevations of the penetrations above the top of the active fuel decreased the vulnerability for core uncovery. Therefore, the ABWR design has large margins to the Appendix K and 10 CFR 50.46 limits. The NRC staff reviewed the modeling of the ABWR design and the supporting qualification data. The NRC staff also performed confirmatory analyses, which showed trends similar to the Westinghouse analyses results. Both the NRC staff's best estimate calculations and Westinghouse's more conservative Appendix K calculations show that there are large margins of peak cladding temperature (PCT) for the ABWR design.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated May 2, 2013 recommending that licensing topical report WCAP-17116-P be approved for application to ABWR designs, subject to the conditions and limitations imposed by the staff.

3. Update on the Electric Power Research Institute (EPRI) Ground Motion Model Project

The Committee met with representatives of the NRC staff to discuss updates to the EPRI Ground Motion Model Project which industry has proposed to use to perform the seismic reevaluations requested in the 10 CFR 50.54(f) letters issued as a result of Task 2.1 of the Fukushima Near-Term Task Force Report. The 50.54(f) letter identifies the 2004 version of the ground motion model as the accepted methodology to perform the reevaluations, but industry updated the model to take advantage of new information and data available since the 2004 model was issued. The staff presentation explained the origins of the 2004 model, the new data and information available that warrants an update to the model, the changes proposed to update the model, and the Senior Seismic Hazard Analysis Committee (SSHAC) process that was used to perform the update. Staff also explained its initial review of the proposal from industry and the path forward and schedule to use the updated model in the Task 2.1 seismic reevaluations.

Committee Action

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

4. Proposed Revision 3 to RG 7.10 (DG-7009), "Establishing Quality Assurance Programs for Packaging used in Transport of Radioactive Material"

The Committee considered proposed draft Regulatory Guide (DG)-7009 and decided not to review it. The Committee has no objection to the staff's proposal to issue this guide for public comment but would like an opportunity to review the draft final version of this guide following the public comment period.

5. Draft Final Revision 4 to Regulatory Guide 1.68 (DG-1259), "Initial Test Programs for Water-Cooled Nuclear Power Plants"

The Committee considered draft Final Revision 4 of Regulatory Guide (RG) 1.68 and decided not to review it. The Committee has no objection to the staff's proposal to issue this document as final.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

The Committee considered the EDO's response of March 25, 2013, to comments and recommendations included in the February 14, 2013, ACRS letter on Draft NUREG-2125, "Spent Fuel Transportation Risk Assessment." The Committee was satisfied with the EDO's response.

SCHEDULED TOPICS FOR THE 604th ACRS MEETING

The following topics are scheduled for the 604th ACRS meeting, to be held on May 9-10, 2013:

- Next Generation Nuclear Plant (NGNP) Key Licensing Issues
- Consequential Steam Generator Tube Rupture (C-SGTR)
- Generic Issue GI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion during a Severe Accident"

Sincerely,

/RA/

J. Sam Armijo
Chairman

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J. Sam Armijo
Chairman

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