February 1, 2008

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: INTERIM LETTER: CHAPTERS 2, 5, 8, 11, 12, AND 17 OF THE U.S. NUCLEAR REGULATORY COMMISSION STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE ESBWR DESIGN

Dear Dr. Shack:

This is in response to the Advisory Committee on Reactor Safeguards' (ACRS or the Committee) November 20, 2007, letter regarding the review of the General Electric-Hitachi Nuclear Energy Americas, LLC, (GEH) application for certification of the economic simplified boiling-water reactor (ESBWR) plant design. During the ACRS meeting on November 2, 2007, the staff discussed its safety evaluation reports (SERs) with open items (OIs) for Chapters 2, 5, 8, 11, 12, and 17 of the ESBWR design certification application with the full committee. These discussions included the status of OIs identified in the SERs as well as the technical concerns associated with them. The ACRS raised specific concerns on Chapter 5 associated with minimizing the potential for stress-corrosion cracking of austenitic stainless steels and nickel-based alloys and measures to minimize and mitigate post-welding processes that could contribute to this type of corrosion. In addition, the Committee raised concerns associated with the use of water chemistry controls as a measure to minimize irradiation-assisted stress-corrosion cracking. The enclosure to this letter discusses the staff's responses to these specific ACRS concerns. The staff continues to work with GEH to obtain satisfactory resolution to the OIs presented in the SERs and looks forward to presenting the resolutions to these OIs to the ACRS during future presentations on the final safety analysis report for the ESBWR design certification application.

The ACRS also stated that, although the basis for the estimated source term for radioactive materials released from fuel into the RCS seems reasonable, the Committee would like to review the data and the analysis procedure used to develop the source team. The staff has sent a request for additional information to GEH to obtain this material and will provide this information to ACRS once received.

Thank you for your comments. I appreciate the willingness of the ACRS to engage with the staff on a chapter-by-chapter review process for the SERs with OIs and believe this process has greatly facilitated the staff's review. My staff looks forward to continued interactions with the Committee on the SERs with OIs for the remaining chapters of the ESBWR design certification application.

Sincerely,

/RA Martin J. Virgilio for/

Luis A. Reyes
Executive Director
for Operations

Enclosure:
Staff Response to ACRS Comments

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
SECY

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The U.S. Nuclear Regulatory Staff Response to the Advisory Committee on Reactor Safeguards Interim Letter Dated November 20, 2007, Regarding Safety Evaluation Reports with Open Items on the ESBWR Design Certification Application

The staff prepared responses to comments from the Advisory Committee on Reactor Safeguards (ACRS) on the staff's safety evaluation report (SER) with open items for Chapter 5, "Reactor Coolant System and Connected Systems," of the economic simplified boiling-water reactor (ESBWR) design certification application. The staff responded to these concerns during the ACRS ESBWR subcommittee meeting on January 16 and 17, 2008, and continues to work with the applicant to develop satisfactory resolution to these concerns and to revise the ESBWR design control document accordingly. The staff plans to discuss final resolution of these concerns during the ACRS full committee meeting on the final SER for the ESBWR design certification application.

ACRS Comment: The staff should further investigate the adequacy of controls on post-weld grinding. GE-Hitachi Nuclear Energy Americas, LLC, (GEH) has placed controls on the use of grinding wheels and wire brushes in the fabrication of the ESBWR components and structures to prevent potentially degrading materials from entering the system. However, post-weld grinding can degrade the resistance of austenitic stainless steels and nickel-based alloys to various stress-corrosion cracking (SCC) mechanisms when exposed to the reactor coolant. The controls on welding practice should be revised to eliminate such practices to the extent possible and to mitigate their consequences in those instances in which grinding is unavoidable.

Staff Response: The staff recognizes that excessive cold working of austenitic stainless steels and nickel-based alloys makes them more susceptible to SCC even when using materials (i.e., low-carbon stainless steel and niobium-modified Alloy 600) that are considered to be resistant to SCC. However, post-weld grinding of austenitic stainless steel and nickel-based alloy welds during the fabrication of reactor coolant pressure boundary components is unavoidable in many instances, such as, during the removal of temporary attachments, surface contouring of welds to facilitate nondestructive examinations, and removal of welding defects. The staff notes that welding defects discovered during the fabrication process by the various examination methods that are in excess of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) acceptance criteria must be repaired. The staff makes use of review guidance in the standard review plan and design and inspection criteria in the ASME Code to provide an adequate basis to ensure the long-term integrity of structures, systems, and components (SSCs) important to safety. Revision 4 of the ESBWR design control document (DCD) partly addresses this issue for austenitic stainless steels used for reactor vessel internals and the reactor coolant pressure boundary. That is, during the fabrication, cold working will be controlled by applying limits in hardness, bend radii, and surface finish on ground surfaces. Revision 4 of the ESBWR DCD is silent, however, on the control of cold working of nickel-based alloys. The staff has been discussing additional controls on grinding with GEH, which the staff will consider if they are proposed by either GEH or a combined operating license (COL) applicant.

ACRS Comment: Although the materials chosen for the pressure boundary are resistant to SCC under normal boiling-water reactor water chemistry, experience indicates that core internals will be susceptible to irradiation-assisted stress-corrosion cracking (IASCC) unless more controls are placed on water chemistry. ACRS would like the opportunity to review ESBWR reactor coolant system chemistry controls in future meetings.

Staff Response: The staff recognizes the potential benefits of controls on water chemistry. The staff makes use of review guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," and design and inspection criteria in the ASME Code to provide an adequate basis to ensure the long-term integrity of SSCs important to safety. This review guidance and the design and inspection codes have evolved over time, recognizing the benefits of tighter controls on water chemistry. However, specific requirements to address IASCC through water chemistry controls have not been developed as part of the U.S. Nuclear Regulatory Commission's regulatory requirements. Such controls have been a subject of discussion with GEH and will be considered by the staff if they are proposed by either GEH or a COL applicant. Although there are no regulatory or ASME Code requirements for a design certification applicant, like GEH, to require the use of a hydrogen water chemistry system, the staff still considers the reactor internals less susceptible to IASCC for several reasons, which are summarized here and described in more detail in the staff's safety evaluation report for Chapter 4 of the ESBWR DCD. Only low-carbon stainless steel and nickel alloys modified for high SCC resistance will be specified for reactor internals. Strict controls of the fabrication and installation processes for the reactor internals will be used. Application of surface finishing techniques will be used to remove surface cold work in the weld heat-affected zones of the major structural welds in the large internals.