APPENDIX F. REPORT BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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

April 17, 2014

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

SUBJECT: SUPPLEMENTAL FINAL SAFETY EVALUATION REPORT ON THE GENERAL ELECTRIC-HITACHI NUCLEAR ENERGY (GEH) APPLICATION FOR CERTIFICATION OF THE ECONOMIC SIMPLIFIED BOILING WATER REACTOR (ESBWR) DESIGN

Dear Chairman Macfarlane:

During the 613th meeting of the Advisory Committee on Reactor Safeguards (ACRS), April 10-11, 2014, we reviewed the supplemental Final Safety Evaluation Report (FSER) for certification of the ESBWR passive nuclear power plant design. In particular, we reviewed the staff's evaluation of the revised analysis procedure for the structural and functional integrity of the ESBWR steam dryer. In our review and our subcommittee meeting on March 5, 2014, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

CONCLUSIONS

The ESBWR steam dryer design is adequate, and the associated structural analysis and planned startup test program are acceptable. There is reasonable assurance that the ESBWR design can be constructed and operated without undue risk to the health and safety of the public.

BACKGROUND

The NRC staff issued the ESBWR FSER on March 9, 2011, to document their review of the ESBWR design. Subsequent to the issuance of the ESBWR FSER, the staff raised additional questions with respect to the GEH analysis procedure for computing oscillating pressure loads acting on the ESBWR steam dryer during normal operation. Following an audit, the staff concluded that there were errors and omissions in the referenced licensing topical reports (LTRs) that GEH needed to correct in order to support the ESBWR application and the final issuance of the design certification.

Steam dryer structural analyses and associated power ascension testing are an integral part of any extended power uprate (EPU) for current boiling water reactors (BWRs). In these plants with increased power, the increased steam flow velocities might cause flow-induced vibrations that generate oscillating pressure loads acting on the steam dryer during operation at higher thermal power, potentially leading to high cycle fatigue failure. Although the steam dryer does not perform a safety function, it must retain its structural integrity to avoid generating loose parts that can affect operation of other components such as the main steam line isolation valves.

We reviewed the supplemental FSER with respect to the ESBWR steam dryer analysis. This FSER supplement documents the NRC staff's review of the changes to the steam dryer analysis process. The overall design of the ESBWR and its steam dryer was not changed.

DISCUSSION

GEH withdrew the initial LTRs and submitted revised engineering reports to explain, substantiate, and benchmark their procedure for computing oscillating acoustic pressure loads acting on the steam dryer. GEH applied a plant-based load evaluation method, which is based on operating experience from existing BWR plants, as well as the advanced boiling water reactor (ABWR) steam dryer design on which the ESBWR steam dryer design is based.

The basic process for determining the acoustic structural loads on the dryer is similar to previous analyses that we have reviewed for EPUs. Acoustic pressure sources are postulated at the junction of the main steam lines and the reactor vessel to determine the relationship between these sources and dryer structural load response. However, in contrast to some steam dryer analyses performed to date, the strength of these acoustic sources is not determined from strain gage measurements on the main steam lines, but rather from direct measurements on the dryer. The design procedure still calls for acoustic analyses of the main steam lines, but only for the purpose of avoiding any resonant conditions.

The detailed design of the ESBWR dryer will be based on estimates of acoustic loads derived from measurements on existing plants. Conservative procedures will be used to develop the design loads from the available data. Final acceptance of the steam dryer is dependent on successful completion of a startup test program for confirming the steam dryer design analysis results as the plant performs power ascension testing. Prior to startup, the acceptance criteria for the peak design stresses will include a factor of two margin relative to ASME Code allowable stresses. This gives a high likelihood that when the startup measurements are made, actual stresses will be below the ASME allowable limits. The engineering reports provide a good description of this analysis process.

After the initial plant startup is complete, the pressure sensor and strain gage instrumentation on the dryer may no longer be available, as has been the case for most of the plants with instrumented dryers during EPU startup testing. We agree with the staff position that once it is verified that the acoustic loads are acceptable in the initial cycle, there is no further need for such instrumentation.

The bias and uncertainties determined from the strain gage measurements on the steam dryer provide confidence in the adequacy of the overall model. However, the overall model may not adequately characterize peak stresses, which are strongly influenced by very local geometries. In response to the staff audit and request for additional information, GEH has improved its requirements for demonstrating adequacy of finite element analysis mesh refinement. Even detailed mesh refinement cannot completely characterize the geometries that affect the peak stresses because they can be affected by local imperfections in welds. Thus, empirical fatigue strength reduction factors are introduced in the refined models. The magnitudes of the factors depend on the detail of the finite element analysis. Such an approach is consistent with usual ASME Code design practice and is acceptable.

In summary, the ESBWR steam dryer design is adequate, and the associated structural analysis and planned startup test program are acceptable. The process agreed to by the staff and GEH provides a good basis for satisfactory operation of the ESBWR steam dryer. In light of this reevaluation, there is reasonable assurance that the ESBWR design can be constructed and operated without undue risk to the health and safety of the public.

Sincerely,

/RA/

John Stetkar
Chairman

REFERENCES

1. Supplemental Safety Evaluation Report for the Economic Simplified Boiling-Water Reactor Standard Plant Design, February 12, 2014 (Ml13330A950)

2. Final Safety Evaluation Report for the Economic Simplified Boiling-Water Reactor Standard Plant Design, March 9, 2011 (ML103470210)

3. NRO Memorandum, Subject: Economic Simplified Boiling-Water Reactor, Design Certification – Supplemental Safety Evaluation, February 12, 2014 (ML14042A261)

4. GE Hitachi Nuclear Energy, “ESBWR Steam Dryer Acoustic Load Definition,” NEDE-33312P, Class III (Proprietary), Revision 5, December 2013 (ML13344B163), and NEDO-33312, Class I (Non-proprietary), Revision 5, December 2013 (ML13344B157)

5. GE Hitachi Nuclear Energy, “ESBWR Steam Dryer Structural Evaluation,” NEDE-33313P, Class III (Proprietary), Revision 5, December 2013 (ML13344B164), and NEDO-33313, Class I (Non-proprietary), Revision 5, December 2013 (ML13344B158)

6. GE Hitachi Nuclear Energy, “ESBWR Steam Dryer – Plant Based Load Evaluation Methodology, PBLE01 Model Description,” NEDE-33408P, Class III (Proprietary), Revision 5, December 2013 (ML13344B176 and ML13344B175), and NEDO-33408, Class I (Non-proprietary), Revision 5, December 2013 (ML13344B159)

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In summary, the ESBWR steam dryer design is adequate, and the associated structural analysis and planned startup test program are acceptable. The process agreed to by the staff and GEH provides a good basis for satisfactory operation of the ESBWR steam dryer. In light of this reevaluation, there is reasonable assurance that the ESBWR design can be constructed and operated without undue risk to the health and safety of the public.

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/RA/

John Stetkar
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1. Supplemental Safety Evaluation Report for the Economic Simplified Boiling-Water Reactor Standard Plant Design, February 12, 2014 (Ml13330A950)

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4. GE Hitachi Nuclear Energy, “ESBWR Steam Dryer Acoustic Load Definition,” NEDE-33312P, Class III (Proprietary), Revision 5, December 2013 (ML13344B163), and NEDO-33312, Class I (Non-proprietary), Revision 5, December 2013 (ML13344B157)

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All safety-related structures, systems, and components (SSCs) are located on the nuclear island and are included in the design certification. Three aspects of the plant design (instrumentation and control systems, human factors engineering, and some piping) will be completed by the combined license (COL) applicant using the Design Acceptance Criteria (DAC) described in the DCD. A final issue relates to assuring long-term recirculation cooling following the limiting design basis accident. This issue was confirmed by our review of the DCD and associated analysis using NRC guidance and documented in our letter report dated September 22, 2010.

ESBWR Design Description

The ESBWR design includes a boiling-water reactor (BWR) nuclear steam supply system (NSSS). It could be constructed at any location that meets the parameters identified in Chapter 2 of the DCD, Tier 2, Revision 7. The ESBWR design utilizes a low-leakage containment vessel, which is comprised of the drywell and wetwell. The containment vessel is a cylindrical steel-lined reinforced concrete structure integrated with the reactor building. The DCD describes a nuclear plant with a NSSS thermal power rating of up to 4,500 megawatts thermal (MWt). Based on this reference design, the plant has a rated gross electrical power output of 1,594 megawatts electric (MWe) and a net electrical power output of approximately 1,535 MWe. The COL applicant will establish the rated electrical power output based on the turbine island design selected and site-specific conditions and may base the COL application on a lower rated thermal power output to satisfy site-specific environmental parameters. While the COL license period is for 40 years, GEH stated that the plant has a design life objective of 60 years without a replacement of the reactor vessel.

Safety Enhancement Features

The ESBWR is a direct-cycle, natural circulation BWR and has passive safety features to cope with a range of design basis accidents (DBAs). Within the containment structure are the isolation condensers (IC), the elevated gravity-driven cooling system (GDCS) water pools, a passive containment cooling system (PCCS), and an elevated suppression pool. These systems can remove decay heat under all conditions. The ESBWR standard design includes a reactor building that surrounds the containment, as well as buildings dedicated exclusively or primarily to housing related systems and equipment.

The limiting ESBWR DBA is a Main Steam Line Break (MSLB). In this DBA, water and steam are initially discharged from the break into the drywell. As the drywell pressure increases, the horizontal vents between the drywell and wetwell clear. Subsequently, a steam-water mixture from the break flows through the vents into the wetwell suppression pool, where the steam is condensed, and the water is cooled to the pool temperature. As the primary system pressure falls to the drywell pressure, water makeup to the reactor vessel is provided by actuation of the GDCS; i.e., the GDCS squib valves open and water flows by gravity head into the vessel from the GDCS pools. This occurs approximately ten minutes after the initiation of the accident. The reactor core is never uncovered during the limiting DBA. Steam condensation in the suppression pool and pressure equilibration between the drywell and wetwell through the vacuum breakers reduce the drywell pressure causing the horizontal vents to close. The remaining noncondensible gases and steam in the drywell then flow up through the PCCS heat exchanger. The steam is condensed as it passes through the PCCS tubes. Water condensate is collected and returned to the GDCS pools, and the noncondensible gases flow into the wetwell gas space. This establishes a passive long-term recirculation cooling mode for over 72 hours. Non-safety-related recirculating fans are credited after 72 hours and result in a further reduction in the containment pressure. However, calculations show that even in a purely passive mode, the containment pressure remains below the design pressure for over 30 days.

Probabilistic Risk Assessment

The ESBWR design certification application included a PRA in accordance with regulatory requirements. The ESBWR PRA is a Level 3 PRA that covers full power operation and shutdown conditions. The scope of initiating events includes internal events and assessments of internal plant fires and floods. The only quantified external events are high winds and tornadoes. A seismic margin analysis was performed, but the risk from seismic events and other possible external events was not quantified. Although many of the analysis elements are consistent with the ASME-RA-Sb-2005 Capability Category 2 Standard, those attributes were not consistently achieved at this stage of the PRA development. For example, some aspects of human performance, models for equipment testing and maintenance, and details of fire and flood damage cannot be analyzed in the absence of a physical plant, procedures, and operations staff. In these cases, surrogate analyses were performed and assumptions were applied to encompass potential plant configurations, operations and maintenance programs, and organizations. In addition, any analyses requiring site-specific characteristics were treated in a generic manner.

Our review found that this PRA was acceptable for design certification purposes. The estimated frequencies of core damage and large releases provide confidence that the ESBWR design achieves NRC staff expectations for advanced plants. The PRA was an integral part of the ESBWR design process, and risk insights influenced a number of design changes throughout the review. This integrated risk perspective was an important contribution to achieving the estimated low risk.

The limited scope, varying level of modeling detail, and lack of specificity with respect to "as-built, as-operated" plant conditions limit direct use of the current ESBWR PRA for risk-informed applications. Therefore, it is important that any future use of the PRA results during the COL process, such as the use of calculated risk importance measures for selection of SSCs for the Design Reliability Assurance Program, should be carefully examined and supplemented by appropriate engineering expertise.

ACRS Review Approach

Our review activities for the ESBWR design certification are listed in the appendix to this report. These activities should be viewed in concert with all our review activities conducted for topical reports on analysis methods used by GEH for the ESBWR. We had numerous subcommittee and full-committee meetings to review the ESBWR as listed in the Appendix. Our reviews did not address security-related issues.

During these reviews, we issued 6 letters identifying issues of concern and areas for which we needed additional discussion. The applicant has submitted additional proposed revisions to the DCD to resolve all the open issues from the NRC staff and of interest to us. It is intended that these revisions be incorporated in Revision 8 of the DCD. Some of the issues included:

- Combustion control of flammable noncondensible gases in the PCCS: GEH revised the design of the IC and PCCS to address the potential for hydrogen detonations within the condenser tubes or the lower plenum. The IC system configuration was modified to isolate it from the ESBWR vessel for loss of coolant accident (LOCA) events and to vent it for non-LOCA events in order to address the possibility of combustion events in the IC. The primary structural material of the PCCS was changed to a high strength stainless steel, and component wall thicknesses were significantly increased so that the PCCS can withstand multiple combustion events under bounding conditions. In addition, a passive catalytic recombiner was added to the PCCS drain line to remove combustible gases from piping to the wetwell.

- Clarification and detailed explanation of digital instrumentation and control (DI&C) systems for ESBWR: GEH provided more detailed explanations and tabular information in the DCD revisions to give us confidence that the four fundamental principles are inherent in the hardware and software DI&C architectures, i.e., redundancy, independence, determinate behavior, and diversity and defense in depth. Finally, additional DAC/ITAAC were developed for the ESBWR to confirm that the final system design would meet these principles.

We agreed with the staff's resolution of all the open items for the ESBWR in regard to the specific safety issues, but our discussions identified a few generic issues that may require further consideration.

Level of Detail for DAC/ITAAC

The DCD and associated ITAAC are designed to ensure that a specific plant will be constructed and operated to conform to the certified design in all areas that are safety significant. The staff has interpreted this to mean that the design certification application must be complete, with two exceptions:

- Items for which the technology is rapidly changing and may be significantly different at the COL stage.
- Items for which the level of detail cannot be provided at the time of certification review (or for which the as-procured and as-built characteristics are needed).

For these exceptions, DAC as part of the ITAAC can be used in lieu of detailed design information. The DAC provide acceptance criteria that assure the design requirements for particular systems and components have been met in the final design and construction. DAC have been used with prior reactor certifications starting with the ABWR and including the AP1000 in 2004. Specifically, DAC have been used for the instrumentation and control (I&C) system, for the control room design with regard to human factors, and for piping design details.

For the ESBWR, the proposed additional information to be included in Revision 8 of the DCD provides expanded detailed functional descriptions and DAC/ITAAC for the DI&C hardware and software architectures which support the conclusion that the design will meet requirements. However, there is a class of descriptive information, i.e., integrated system logic diagrams, that is not included. These diagrams would simplify the review and make the safety judgment more robust. Such functional descriptions would also aid in the inspection of DAC/ITAAC for final I&C qualification. Under current practice, the NRC staff does not require that such integrated system logic diagrams be included in the Tier 2 information. We suggest that staff consider requiring such information.

In summary, we agree with the staff's resolution of all the open items for the ESBWR in regard to the specific safety issues. The ESBWR design is robust and there is reasonable assurance that it can be built and operated without undue risk to the health and safety of the public.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

<u>References</u>:

1. Memoranda from David Matthews, transmitting "Final Safety Evaluation Reports Chapters 1 – 22," (ML102850502 package)

2. Letter to U.S. Nuclear Regulatory Commission, transmitting "Transmittal of ESBWR DCD Markups to Tier 1 and Chapter 2, 3, and 19 Related to GEH Internal Corrective Actions and Discussions with the NRC," (ML102730795) 09/24/2010

3. Letter to Gregory B. Jaczko, transmitting "Long-Term Core Cooling for the ESBWR," (ML102560364) 09/22/2010

4. Letter to U.S. Nuclear Regulatory Commission, transmitting "ESBWR Design Control Document, Tier 2 Chapter 7 and Tier 1 Changes to Respond to ACRS Remarks," (ML102700297) 09/23/2010

5. Letter to U.S. Nuclear Regulatory Commission, transmitting "Revised Response (Revision 2) to NRC Request for Additional Information Letter No. 411 Related to ESBWR Design and Certification Application – Engineered Safety Features – RAI Number 6.2-202, Supplement 1," (ML102670082) 09/21/2010

6. Letter to R.W. Borchardt, transmitting "Applicability of TRACE Thermal-Hydraulic System Analysis Code to Evaluate the ESBWR Design and Related Matters," (ML091940352) 07/29/2010

7. Letter to U.S. Nuclear Regulatory Commission, transmitting "ESBWR Design Control Document, Revision 7, Tier 1 and Tier 2," (ML1013401430 and ML101340380) 03/29/2010

8. Letter to U.S. Nuclear Regulatory Commission, transmitting "Licensing Topical Report NEDO-33201, ESBWR Design Certification Probabilistic Risk Assessment," (ML100740287) 03/02/2010

9. Letter to R.W. Borchardt, transmitting "Interim Letter 6: Chapters 7 and 14 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design," (ML083460306) 12/22/2008

10. Letter to R.W. Borchardt, transmitting "Interim Letter 5: Chapters 19 and 22 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design," (ML082810703) 10/29/2008

11. Letter to R.W. Borchardt, transmitting "Interim Letter 4: Chapter 3 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design," (ML081930777) 07/21/2008

12. Letter to R.W. Borchardt, transmitting “Interim Letter 3: Chapters 4, 6, 15, 18, and 21 of the NRC Staff’s Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design,” (ML081330447) 05/23/2008

13. Letter to Dale E. Klein, transmitting “Digital Instrumentation and Control System Interim Staff Guidance,” (ML081050636) 04/29/2008

14. Letter to Luis A. Reyes, transmitting “Interim Letter: Chapters 9, 10, 13, and 16 of the NRC Staff’s Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design,” (ML080670596) 03/20/2008

15. Letter to William D. Travers, transmitting “Draft Safety Evaluation Report for the ESBWR Pre-Application Review,” (ML040440487) 02/12/2004

16. Letter to Luis A. Reyes, transmitting “Interim Letter: Chapters 2, 5, 8, 11, 12, and 17 of the NRC Staff’s Safety Evaluation Report With Open Items Related to the Certification of the ESBWR Design,” (ML073070006) 11/20/2007

17. Letter to Luis A. Reyes, transmitting “Application of the TRACG Computer Code to Evaluate the Stability of the ESBWR,” (ML061110458) 04/21/2006

18. ASME-RA-Sb-2005, “Standard for Probabilistic Risk Assessment for Nuclear Plant Application,” December 2005

APPENDIX

CHRONOLOGY OF THE ACRS REVIEW OF THE GEH APPLICATION FOR THE ESBWR DESIGN CERTIFICATION

The extensive ACRS review of the ESBWR design and its interactions with representatives of the NRC staff and GEH are discussed in the minutes and transcripts of the following ACRS meetings.

ACRS MEETING/DATES	SUBJECT
Thermal-Hydraulic Phenomena Subcommittee 1/14-15/2004	ESBWR Analytical Methods
509th ACRS Meeting 2/5-6/2004	Draft Safety Evaluation Report for the ESBWR pre-application review
Thermal-Hydraulic Phenomena Subcommittee 1/19/2006	ESBWR Stability, Regulatory Guide 1.82
Thermal-Hydraulic Phenomena Subcommittee 3/14/2006	ESBWR Stability Methodology
531st ACRS Meeting 4/5-7/2006	NRC Staff's Draft SER related to the use of TRACG computer code to evaluate the stability of the ESBWR
ESBWR Subcommittee 10/2-3/2007	ESBWR DCD and select portions of Chapters 2, 8, and 17 of the NRC Staff's SER with Open Items
ESBWR Subcommittee 10/25/2007	ESBWR DCD and NRC Staff's SER with Open Items for select portions of Chapters 5, 11, and 12
547th ACRS Meeting 11/1-3/2007	Chapters 2, 5, 8, 11, 12, and 17 of NRC Staff's SER with Open Items related to the certification of the ESBWR Design
ESBWR Subcommittee 11/15/2007	ESBWR DCD and NRC Staff's SER with Open Items for select portions of Chapters 9, 10, 13, and 16
550th ACRS Meeting 3/6-8/2008	Chapters 9, 10, 13, and 16 of the NRC Staff's SER with Open Items related to the certification of the ESBWR Design

ESBWR Subcommittee 1/16/2008	ESBWR DCD and NRC Staff's SER with Open Items for select portions of Chapters 4, 6, 15, and 21
ESBWR Subcommittee 4/9/2008	ESBWR DCD Containment/Reactor Thermal-Hydraulic issues from ACRS review of NRC Staff's SER with Open Items for Chapters 4, 6, 15, 18, and 21
552nd ACRS Meeting 5/8-9/2008	Chapters 4, 6, 15, 18, and 21 of the NRC Staff's SER with Open Items related to the certification of the ESBWR Design
ESBWR Subcommittee 6/18-19/2008	ESBWR DCD and NRC Staff's SER with Open Items for select portions of Chapter 3
554th ACRS Meeting 7/9-11/2008	Select portions of Chapter 3 of the NRC Staff's SER with Open Items related to the certification of the ESBWR Design
ESBWR Subcommittee 6/3/2008	ESBWR DCD and NRC Staff's SER with Open Items for select portions of Chapters 19 and 22
ESBWR Subcommittee 8/21-22/2008	ESBWR DCD and NRC Staff's SER with Open Items for select portions of Chapters 19 and 22, including selected PRA Accident Sequences
556th ACRS Meeting 10/2-3/2008	Select portions of Chapters 19 and 22 of the NRC Staff's SER with Open Items related to the certification of the ESBWR Design
ESBWR Subcommittee 10/21-22/2008	ESBWR DCD and NRC Staff's SER with Open Items for select portions of Chapter 14
ESBWR Subcommittee 12/3/2008	ESBWR DCD and NRC Staff's SER with Open Items for select portions of Chapter 7
558th ACRS Meeting 12/4-6/2008	Select portions of Chapters 7 and 14 of the NRC Staff's SER with Open Items related to Certification of the ESBWR Design
Thermal-Hydraulic Phenomena Subcommittee 2/27/2009	TRACE applicability to ESBWR LOCA

ESBWR Subcommittee 6/17/2009	ESBWR Design Basis Containment Analysis and related open items identified in NRC Staff's SER Open Items, Chapter 6
564[th] ACRS Meeting 7/8-10/2009	Applicability of TRACE thermal-hydraulic system analysis code to evaluate the ESBWR design and related matters
ESBWR Subcommittee 10/20-21/2009	ESBWR DCD and NRC Staff's SER with Open Items related to various topics
ESBWR Subcommittee 11/17-18/2009	ESBWR DCD and NRC Staff's SER with Open Items related to various topics including long-term core cooling
ESBWR Subcommittee 5/18-19/2010	ESBWR DCD and Various Topical Reports
ESBWR Subcommittee 6/22/2010	ESBWR DCD and NRC Staff's FSER for select portions of Chapters 5, 8, 11, 17, 19, and 22
ESBWR Subcommittee 7/13/2010	ESBWR DCD and NRC Staff's Review of various SER Open Items for Chapter 6 regarding long-term core cooling
ESBWR Subcommittee 8/16-17/2010	ESBWR DCD and NRC Staff's FSER for select portions of Chapters 2, 3, 9, 10, 12, 14, 15, 16, 18, 20, and 21
575[th] ACRS Meeting 9/9-11/2010	NRC Staff's evaluation of the adequacy for long-term cooling as it applies to the ESBWR design certification application
ESBWR Subcommittee 9/23-24/2010	ESBWR DCD and NRC Staff's FSER for select portions of Chapters 3, 4, 6, 7, and 9
ESBWR Subcommittee 10/6/2010	ESBWR DCD various topics and Security Related AIA Information and NRC Inspections

to the specific safety issues. The ESBWR design is robust and there is reasonable assurance that it can be built and operated without undue risk to the health and safety of the public.

Sincerely,

/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 October 20, 2010

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE GENERAL ELECTRIC-HITACHI NUCLEAR ENERGY (GEH) APPLICATION FOR CERTIFICATION OF THE ECONOMIC SIMPLIFIED BOILING WATER REACTOR (ESBWR) DESIGN

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