EDO Principal Correspondence Control

FROM: DUE: 11/22/10 EDO CONTROL: G20100653

DOC DT: 10/20/10

FINAL REPLY:

Said Abdel-Khalik, ACRS

TO:

Chairman Jaczko

FOR SIGNATURE OF : ** GRN ** CRC NO: 10-0471

Borchardt, EDO

DESC:

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 (EDATS: SECY-2010-0501)

DATE: 10/22/10

ASSIGNED TO: NRO

CONTACT: Johnson

ROUTING:

Borchardt
Weber
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EDATS
Electronic Document and Action Tracking System

EDATS Number: SECY-2010-0501 **Source:** SECY

General Information

Assigned To: NRO **OEDO Due Date:** 11/22/2010 11:00 PM

Other Assignees: **SECY Due Date:** NONE

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

Description:

CC Routing: NONE

ADAMS Accession Numbers - Incoming: NONE **Response/Package:** NONE

Other Information

Cross Reference Number: G20100653, LTR-10-0471 **Staff Initiated:** NO

Related Task: **Recurring Item:** NO

File Routing: EDATS **Agency Lesson Learned:** NO

OEDO Monthly Report Item: NO

Process Information

Action Type: Letter **Priority:** Medium

Sensitivity: None

Signature Level: EDO **Urgency:** NO

Approval Level: No Approval Required

OEDO Concurrence: NO

OCM Concurrence: NO

OCA Concurrence: NO

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Document Information

Originator Name: Said Abdel-Khalik **Date of Incoming:** 10/20/2010

Originating Organization: ACRS **Document Received by SECY Date:** 10/22/2010

Addressee: Chairman Jaczko **Date Response Requested by Originator:** NONE

Incoming Task Received: Letter

OFFICE OF THE SECRETARY
CORRESPONDENCE CONTROL TICKET

Date Printed: Oct 22, 2010 09:38

PAPER NUMBER:	LTR-10-0471	**LOGGING DATE:**	10/21/2010
ACTION OFFICE:	EDO		

AUTHOR: Said Abdel--Khalik

AFFILIATION: ACRS

ADDRESSEE: Gregory Jaczko

SUBJECT: Report on the safety aspects of the GE-Hitachi Nuclear Energy (GEH) application for certification of the economic simplified boiling water reactor (ESBWR) design

ACTION: Appropriate

DISTRIBUTION: RF

LETTER DATE: 10/20/2010

ACKNOWLEDGED No

SPECIAL HANDLING:

NOTES:

FILE LOCATION: ADAMS

DATE DUE: **DATE SIGNED:**

EDO --G20100653

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

October 20, 2010

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

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

Dear Chairman Jaczko:

During the 576th meeting of the Advisory Committee on Reactor Safeguards, October 7-9, 2010, we completed the safety review of the GEH application for certification of its ESBWR passive nuclear power plant design. This letter report is intended to fulfill the requirements of 10 CFR 52.53. During our review, we had the benefit of discussions with representatives of the NRC staff and GEH and its consultants. We also had the benefit of the documents referenced.

CONCLUSION

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.

DISCUSSION

<u>ESBWR Application</u>

On August 24, 2005, GEH submitted its application to the NRC for certification of the ESBWR design. This application was submitted in accordance with Subpart B, "Standard Design Certifications," of 10 CFR Part 52. The NRC formally docketed the application for design certification (Docket No. 52-010) on December 1, 2005. The application consists of the ESBWR Design Control Document (DCD) and the ESBWR probabilistic risk assessment (PRA) report.

The DCD information is divided into two parts, denoted as Tier 1 and Tier 2. Tier 1 contains the portion of the generic design-related information that is proposed for approval and certification in the rule including, among other things, the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). Tier 2 contains the portion of the generic design-related information that is proposed for approval but not certification. Tier 2 information includes, among other things, a description of the design of the facility for a final safety analysis report, as required by 10 CFR 52.47. Subsequently, GEH supplemented the information in the DCD by providing revisions to the DCD. The applicant submitted the most recent version, DCD Revision 7, on March 29, 2010. 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.

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

References:

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
564th 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
575th 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