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NUREG-1966
Supplement 1

Final Safety Evaluation Report

Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design

Supplement 1

Office of New Reactors

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Office of New Reactors

ABSTRACT

This report supplements the final safety evaluation report (FSER) for the Economic Simplified Boiling-Water Reactor (ESBWR) standard plant design. The FSER was issued by the U.S. Nuclear Regulatory Commission (NRC) as NUREG-1966 in April 2014 to document the NRC staff's technical review of the ESBWR design. The application for the ESBWR design was submitted on August 24, 2005, by General Electric-Hitachi (GEH) in accordance with Subpart B, "Standard Design Certifications," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52. This supplement documents the NRC staff's review of GEH's changes to the ESBWR design documentation in the design control document (DCD) since the issuance of the FSER. On the basis of the evaluation described in the ESBWR FSER (NUREG-1966) and this report, the NRC staff concludes that the changes to the DCD (up to and including Revision 10 to the ESBWR DCD) are acceptable and that GEH's application for design certification meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the ESBWR standard Plant design.

TABLE OF CONTENTS

APPENDICES

1.0 INTRODUCTION AND GENERAL DISCUSSION

1.0 Introduction

This report supplements the final safety evaluation report (FSER) for the Economic Simplified Boiling-Water Reactor (ESBWR) standard plant design. The U.S. Nuclear Regulatory Commission (NRC) staff issued the FSER on March 9, 2011 (Agencywide Documents Access and Management System (ADAMS)[1] Accession No. ML103470210) to document the NRC staff's review of the ESBWR design. This supplement documents the NRC staff's review of the changes to the ESBWR design documentation since the issuance of the FSER, as well as other regulatory issues that emerged since the FSER was completed. Each section of this supplement is numbered and titled the same as the section of the FSER that is being updated. The discussions are supplementary to, but not in lieu of, the discussions in the FSER, unless otherwise noted.

General Electric-Hitachi Nuclear Energy (GEH or the applicant) submitted the ESBWR design documentation under Subpart B, "Standard Design Certifications," of Part 52 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 52), "Licenses, Certifications, and Approvals for Nuclear Power Plants." The ESBWR design documentation includes the ESBWR Design Control Document (DCD) and a description of the ESBWR probabilistic risk assessment. Changes to the ESBWR DCD (DCD Revision 10) (Docket No. 52-010) were submitted on April 1, 2014, after the FSER was issued. The staff's review of these DCD changes, as well as, other items that emerged since the FSER was completed, is discussed in Section 1.1.5 of this report.

Throughout the review, the NRC staff (staff) requested that the applicant submit additional information to clarify the description of the ESBWR design. This report discusses some of the applicant's responses to these requests for additional information (RAIs) and the responses are listed in the reference section.

This report references several GEH reports. Some of these reports and communications include information that the applicant requested be exempt from public disclosure, as provided by 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." For each such report, the applicant provided a nonproprietary version, similar in content except for the omission of the proprietary information. The staff based its findings on the proprietary versions of these documents, which are those primarily referenced throughout this report.

Within certain chapters of this report, the staff needed to present proprietary information for completeness. In these chapters, the proprietary information was subsequently redacted in order to make this report publicly available but references are provided to the proprietary version of the chapter for those individuals permitted to review the proprietary information.

This supplement is issued by the Division of New Reactor Licensing in the Office of New Reactors (NRO), NRC. The NRC's project managers for this part of the ESBWR design certification review are Tekia Govan and David Misenhimer. They may be reached by calling 301-415-6197 or

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[1] ADAMS is the NRC's information system that provides access to all image and text documents that the NRC has made public since November 1, 1999, as well as to bibliographic records (some with abstracts and full text) that the NRC made public before November 1999. Documents available to the public may be accessed via the Internet at http://www.nrc.gov/reading-rm/adams.html. Documents may also be viewed by visiting the NRC's Public Document Room at One White Flint North, 11555 Rockville Pike, Rockville, MD. Telephone assistance for using Web-based ADAMS is available at 800-397-4209 between 8:30 a.m. and 4:15 p.m., eastern standard time, Monday through Friday, except Federal holidays.

301-415-6590, or by writing to them at the Office of New Reactors, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. The ESBWR design documentation and all revisions are available for public inspection at the NRC's Public Document Room and the NRC's Public Electronic Reading Room (ADAMS).[1] The NRC's Public Electronic Reading Room is at http://www.nrc.gov/reading-rm/adams/web-based.html. Through this Web site, the public can gain access to ADAMS, which provides text and image files of NRC's public documents. The ESBWR FSER and this supplement are also available for public inspection at the NRC's Public Document Room and Electronic Reading Room.

1.1.5 Summary of Principal Review Matters

Seven principal review matters are addressed in this supplemental FSER. These include:

1. Protection of Offgas System within the Turbine Building
2. Reactor Pressure Vessel Internals
3. Bulletin 2012-01 and GDC 17
4. Seismic Analysis of Fuel in Spent Fuel and Buffer Pools
5. ESBWR DCD Tier 1, ASME Code Definition
6. ESBWR DCD Tier 1, ASME Code Component Design Verification ITAAC
7. ESBWR DCD Tier 2, Editorial Corrections in Chapters 16 and 16B

1.1.5.1 Protection of the Offgas System within the Turbine Building

In the DCD Tier 2, Revision 9, Section 11.3, the applicant described the Offgas System (OGS) as part of the Gaseous Waste Management System (GWMS). The OGS provides for holdup, and thereby decay of radioactive gases in the offgas from the main condenser air removal system, and consists of process equipment along with monitoring instrumentation and control components. DCD Table 3.2-1 showed the OGS as located in the Turbine Building (TB), and described the OGS as non-safety related (N), seismic category NS (not seismic category I or II), with a safety-related classification S indicating that special quality requirements are applied commensurate with the importance of the item's function. Note (5)d of Table 3.2-1 stated the OGS is required to be designed in accordance with Radioactive Waste Management requirements from Regulatory Guide (RG) 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," for Category RW-IIa. The OGS components were reviewed by the staff according to the guidance provided in RG 1.143, RW-IIa. However, prior to issuance of the FSER on March 9, 2011, the staff had not evaluated the TB structure for providing protection of the OGS components per RW-IIa classification under RG 1.143.

Information in various sections of the DCD provided information regarding protection of the OGS to satisfy the criteria in RG 1.143 for Category RW-IIa. Section 3.8.4 of this supplemental FSER documents the NRC staff's review and evaluation of information included in the DCD to determine if the TB structure satisfies the design criteria in RG 1.143 for Category RW-IIa and therefore provides adequate protection for the OGS components. The discussions are supplementary to, but not in lieu of, the discussions in FSER Section 3.8, unless otherwise noted.

1.1.5.2 Reactor Pressure Vessel Internals

This supplemental FSER documents the NRC staff's review of information included in the DCD, Revision 10 referenced engineering reports, and GEH responses to RAIs on the design and evaluation of the structural integrity of ESBWR reactor pressure vessel internals.

Following the issuance of the FSER on March 9, 2011, the NRC staff identified issues applicable to the ESBWR steam dryer structural analysis based on information obtained during the NRC review of a license amendment request for a power uprate at an operating Boiling Water-Reactor (BWR) nuclear power plant. As a result of the resolution of those issues at that plant, GEH revised the DCD to withdraw the original topical reports that addressed the ESBWR steam dryer structural evaluation, and to reference new engineering reports that describe the updated ESBWR steam dryer analysis methodology. The NRC staff reviewed the revised DCD sections, the new GEH engineering reports, and RAI responses; and conducted several public meetings with GEH to discuss the ESBWR steam dryer analysis. The staff also performed an audit of the GEH steam dryer analysis methodology at the GEH facility in Wilmington, NC, in March 2012, and performed a vendor inspection at that facility of the quality assurance program for GEH engineering methods in April 2012.

Because the original topical reports on the ESBWR steam dryer have been withdrawn by GEH, this supplemental FSER completely replaces the NRC staff review and conclusions in the applicable NRC safety evaluation reports for the original GEH topical reports on the ESBWR steam dryer. The NRC safety evaluation reports on the original GEH topical reports for the ESBWR steam dryer are hereby withdrawn. For the same reason, this supplemental FSER completely replaces the discussion of RAI 3.9-58 in FSER Section 3.9.2.3.3, "Staff Evaluation," and Section 3.9.2.3.4, "Conclusion." The new engineering reports are addressed in the applicable sections of this Supplemental FSER. Section 3.9.5 of this supplemental FSER replaces in its entirety Section 3.9.5, "Reactor Pressure Vessel Internals," of the FSER issued on March 9, 2011. Information related to ESBWR reactor pressure vessel internals other than the steam dryer (such as core support structures) have been copied from the FSER and placed in this supplemental FSER to provide the description of the NRC staff review of all ESBWR reactor pressure vessel internals in one location.

1.1.5.3 Bulletin 2012-01 and GDC 17

On July 27, 2012, the NRC issued Bulletin 2012-01 to all holders of operating licenses and combined licenses for nuclear power reactors. The Bulletin requested information about the facilities' electric power system designs, in light of the then-recent operating experience that involved the loss of one of the three phases of the offsite power circuit (single-phase open circuit condition) at Byron Station, Unit 2, and to verify compliance with applicable regulations. The Bulletin indicated that the NRC would determine if further regulatory action was warranted, based on the responses to the information request. In order to verify the applicants of new reactors have addressed the design vulnerability identified at Byron, RAIs were issued. RAI 8.1-22 was issued to GEH for the ESBWR design and requested the applicant to provide applicable design basis information (Chapter 8, Tier 2) and ITAAC information (Chapter 2, Section 2.13 and 4.2) in accordance with 10 CFR 52.47, "Contents of Applications; Technical Information," and 10 CFR 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report." The RAI requested GEH to elaborate how it satisfies the requirements of GDC 17, if a loss of single phase occurs on the credited offsite power circuit.

On November 1, 2013, the NRC conducted a public meeting with representatives from the Nuclear Energy Institute (NEI) and industry to discuss the industry initiative associated with resolving the issues identified in Bulletin 2012-01. During the meeting, industry representatives provided feedback regarding their review of an offsite power two-phase open circuit event that occurred at Forsmark 3 nuclear power plant in Sweden. NEI informed the NRC staff that their detailed analyses of this condition indicated that the proposed single-open phase detection system may not be sensitive enough to detect a two-phase open circuit condition. Therefore, NEI has taken the position that a two-phase open condition must also be considered when developing a resolution to the Bulletin open phase issue.

In response to RAI 8.1-22, new information was provided in DCD Tier 1, Revision 10, Table 2.13.1-2 and DCD Tier 2, Revision 10, Sections 2.13 and 8.2. Sections 8.2 and 14.3.6 of this supplemental FSER documents the NRC staff's review of information included in the ESBWR DCD in connection with Bulletin 2012-01, and the staff determination regarding whether the new information satisfies the requirements of GDC 17. The discussions are supplementary to, but not in lieu of, the discussions in FSER Sections 8.2 and 14.3.6, unless otherwise noted.

1.1.5.4 <u>Seismic Analysis of Fuel in Spent Fuel and Buffer Pools</u>

In support of the application for the ESBWR DC, GEH issued Revision 4 of topical report NEDO-33373, "Dynamic, Load-Drop and Thermal-Hydraulic Analyses for ESBWR Fuel Racks," in March 2010. (Unless otherwise noted, references to NEDO-33373 refer to Revision 4.) NEDO-33373 documented the results of the structural and thermal-hydraulic analyses for the design of fuel storage racks (FSRs) located in the ESBWR spent fuel pool and the buffer pool. The staff concluded in the Safety Evaluation (ADAMS Accession No. ML101600135) for NEDO-33373 that the ESBWR FSRs met the relevant requirements of General Design Criterion (GDC) 2, "Design Bases for Protection against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and that the FSRs are designed to withstand the forces (loads) associated with the safe shutdown earthquake (SSE) that has been established as the certified seismic design response spectra (CSDRS) for the ESBWR certified design. Because the FSR design is adequate, fuel assemblies stored in the FSRs will be protected from excessive damage in the event of an SSE. Subsequently, the staff determined that NEDO-33373 did not specifically address fuel assembly structural design, and sought confirmation that the fuel assemblies housed in the FSRs maintain structural integrity, should they be subjected to the design-basis seismic events defined for the ESBWR certified design.

In an evaluation report entitled "ESBWR Spent Fuel Seismic Qualification," Revision 4, dated September 23, 2011, (GEH Spent Fuel Seismic Qualification (SFSQ) Report), GEH provided a supplemental evaluation to confirm that the consequent loads on the fuel assemblies in the FSRs induced by a design-basis seismic event do not lead to unacceptable structural damage of the fuel. Section 9.1.2 of this supplemental FSER documents the NRC staff's review and evaluation of the applicant's SFSQ Report. The discussions are supplementary to, but not in lieu of, the discussions in FSER Section 9.1, unless otherwise noted.

1.1.5.5 ESBWR DCD Tier 1, ASME Code Definition

The Tier 1 definition of "ASME Code requirements" in Revision 9 of the DCD did not specifically include alternatives to the Code that are authorized by the NRC pursuant to 10 CFR 50.55a(a)(3). Because the definition was not explicit in this regard, a concern was raised regarding whether a COL holder referencing the ESBWR DCD might need an exemption to use an alternative to the Code under 10 CFR 50.55a(a)(3). To remove all doubt with respect to this concern, DCD Tier 1, Revision 10 was revised to state that "ASME Code requirements" means the American Society of Mechanical Engineers (ASME) Code or any NRC-authorized alternative under 10 CFR 50.55a(a)(3). Section 14.3 of this supplemental FSER documents the NRC staff's review and evaluation of these changes. The discussions are supplementary to, but not in lieu of, the discussions in FSER Section 14.3, unless otherwise noted.

1.1.5.6 ESBWR DCD Tier 1, ASME Code Component Design Verification ITAAC

While confirming the inspectability and consistency of design certification inspection, test, analysis, and acceptance criteria (ITAAC), a concern was raised that ESBWR ASME Code component design verification ITAAC, as written in Revision 9 of the DCD, might be viewed as requiring design verification on as-designed ASME Code components, rather than as-built ASME Code components, which is the underlying purpose of these ITAAC. In DCD Tier 1, Revision 10, a number of ASME Code component design and as-built reconciliation ITAAC were consolidated and rewritten to make explicit that they apply to design verification of as-built ASME Code components, and to ensure efficient ITAAC closure. Section 14.3.3 of this supplemental FSER documents the NRC staff's review and evaluation of these changes. The discussions are supplementary to, but not in lieu of, the discussions in FSER Section 14.3.3, unless otherwise noted.

1.1.5.7 ESBWR DCD Tier 2, Editorial Corrections in Chapters 16 and 16B

The applicant made minor editorial changes in DCD Tier 2, Revision 10, Chapters 16 and 16B. Section 16.0 of this supplemental FSER documents the NRC staff's review and evaluation of these changes. The discussions are supplementary to, but not in lieu of, the discussions in FSER Section 16.0, unless otherwise noted.

1.6 Material Referenced

DCD Tier 2, Revision 10, Section 1.6, was revised by modifying Table 1.6-1 and adding two new tables that clarify the status of documents referenced in Tier 2 of the DCD. DCD Tier 2, Revision 10, Table 1.6-1 lists GE and GEH topical reports and technical reports that are incorporated by reference into the ESBWR DCD, and Table 1.6-2 lists other technical reports that are incorporated by reference into the ESBWR DCD. These reports contain information that the NRC regards as requirements on the ESBWR DCD design, and which are considered to be matters resolved in subsequent licensing and enforcement actions involving plants referencing the ESBWR design certification under Paragraph VI, ISSUE RESOLUTION, of the ESBWR Design Certification Rule. Table 1.6-3 lists reports to which the DCD refers, but which only contain general reference material for the ESBWR DCD Tier 2. These reports do not contain any requirements on the ESBWR DCD design.

The additional reports the staff reviewed to prepare this supplemental FSER include:

1. GE Hitachi Nuclear Energy, "ESBWR Steam Dryer Acoustic Load Definition," NEDE-33312P, Revision 5, Class III, December 2013.

2. GE Hitachi Nuclear Energy, "ESBWR Steam Dryer - Plant Based Load Evaluation Methodology - PBLE01 Model Description," NEDE-33408P, Revision 5, Class III, December 2013.

3. GE Hitachi Nuclear Energy, "ESBWR Steam Dryer Structural Evaluation," NEDE-33313P, Revision 5, Class III, December 2013.

4. GE Hitachi Nuclear Energy, "ESBWR Spent Fuel Seismic Qualification," September 22, 2011.

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.8 Seismic Design

3.8.4.1 Regulatory Criteria

The staff reviewed the design of the Off Gas System (OGS) to determine whether it is in compliance with the General Design Criteria (GDC) 1 and 2 of Appendix A to 10 CFR Part 50 as they relate to structures, systems, and components important to safety being designed to quality standards commensurate with their importance to safety, and to withstand appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena. RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components (SSCs) Installed in Light-Water-Cooled Nuclear Power Plants,” specifies that SSCs which process radioactive wastes are evaluated for needed protection from external hazards, and provides an acceptable method of complying with GDC 1 and 2 for radioactive waste processing systems. The staff performed its review using the guidance provided in RG 1.143.

3.8.4.2 Summary of Technical Information

ESBWR DCD Table 3.2-1 shows that the OGS is located in the Turbine Building (TB). Section 11.3.2.3 of the DCD states that the OGS process equipment is housed in a reinforced concrete structure adjacent to the Main Turbine Condenser in the TB. The Charcoal Adsorber Tanks are located in an adjacent vault, and the refrigerant dryers and the OGS monitoring instrumentation are located in separate adjacent rooms. DCD Figure 11.3-1 shows that the Charcoal Beds and the Guard Beds are located in the Charcoal Vault. Per DCD Table 12.3-8, Room 4196 (Elevation -1400 millimeter [mm]) is identified as the Charcoal Adsorber Vessel Vault. DCD Section 11.3.2.7, “Seismic Design,” states that the OGS is in compliance with RG 1.143 for seismic design RW-IIa. DCD Section 11.3.7.1, “Basis and Assumptions,” states that the OGS is designed to be detonation resistant, and seismic per Table 3.2-1, and meets all criteria of RG 1.143. DCD Section 3.5.2, “Structures, Systems, and Components to be Protected from Externally Generated Missiles,” states that provisions are made to protect the Off-Gas Charcoal Bed Adsorbers against tornado missiles. DCD Section 3.7.2.8.1, “Turbine Building,” provides the design criteria for the TB, and Tier 1, Table 2.16.8-1, “ITAAC for the Turbine Building,” provides design commitment for the TB. The staff used the information included in the above DCD Sections in its evaluation.

3.8.4.3 Staff Evaluation

Based on the radiological release criteria provided in RG 1.143, the ESBWR radwaste system including the OGS is classified as RW-IIa (High Hazard). The OGS is not a safety-related system. However, the OGS processes and controls the release of gaseous wastes to the environs, and is important to safety. The staff evaluated the classification of RW-IIa for the OGS as well as the design of the system, and finds it acceptable, as described below, based on compliance with GDC 1 and 2.

According to the information included in the DC, the OGS components are located below grade, in the lower interior of the TB. Section 3.7.2.8 provides the design criteria for the TB, and states that the method of analysis of the TB is the same as a seismic Category I structure including the loading cases and acceptance criteria as shown in Tables 3.8-15, “Load Combinations, Load Factors and Acceptance Criteria for the Safety-Related Reinforced Concrete Structures,” and 3.8-16, “Load Combinations, Load Factors and Acceptance Criteria for the Safety-Related Steel Structures.” RG 1.143, Table 1 through Table 4, provides the codes and standards, loads, load combinations, and acceptance criteria to be used for design of SSCs in radwaste facilities. A staff review of the design criteria included in these Tables shows that the design criteria used for design of seismic Category I structures are more conservative than those in RG 1.143. Therefore, design of the TB using the same loading cases and acceptance criteria as seismic Category I structures would provide the OGS components the protection afforded by design in accordance with the RG 1.143, RW-IIa classification. However, the design commitment for the TB shown in the ITAAC in Tier 1, Table 2.16.8-1, states that the TB design does not include protection against tornado missiles. According to Table 2, RG 1.143, the structures housing radwaste systems classified as RW-IIa should be designed for tornado missiles based on a tornado wind velocity that is 60 percent of the tornado wind velocity per RG 1.76 used for design of seismic Category I structures.

Though the entire TB structure is not designed for protection against tornado missiles, DCD Section 3.5.2 states that provisions are made to protect the Off-Gas Charcoal Bed Adsorbers against tornado missiles. In its review the staff noted that the Charcoal Bed Adsorbers are located in Room 4196 (Table 12.3-8) in the TB, and the other OGS components are located in adjacent rooms. DCD Figures 1.2-12 through Figure 1.2-14 and Figure 1.2-20 provides the location plan and sectional views of the rooms. Room 4196 is located at elevation -1400 mm, which is 5.9 meters (19.4 feet) below grade, and is surrounded by at least 120 centimeter (47 inch) thick walls (Table 12.3-8). There are multiple reinforced concrete floors above the vault. The vault is located at least 11.5 meters away from the exterior walls of the TB. Other OGS components and monitoring instruments are located in adjacent rooms that are below grade. Therefore, the OGS components in the TB are protected from externally generated tornado missiles by at least one external wall, and an additional wall or floor, all of which are designed using the load combinations and acceptance criteria used for seismic Category I structures. Because the TB is designed to the same acceptance criteria as a seismic Category I structure and because the charcoal adsorber bed and other OGS components are located in the lower interior of the TB with surrounding walls and floors such that a tornado missile must penetrate through at least one external wall and one additional wall or floor before coming in contact with any OGS component, the staff concludes that there is reasonable assurance that housing of the OGS components inside the TB provides adequate protection for these components against design basis tornado missiles for radwaste processing systems per RG 1.143, RW-IIa.

3.8.4.4 Conclusion

Based on the foregoing, the staff concludes that housing of the OGS as protected, structurally, inside the TB satisfies all design criteria specified in RG 1.143 under the classification of RW-IIa for the radwaste system. Therefore, the staff finds the design acceptable.

3.9 Mechanical Systems and Components

3.9.5 Reactor Pressure Vessel Internals

This supplemental final safety evaluation report (supplemental FSER) documents the NRC staff's review of information included in the DCD, referenced engineering reports, and GEH responses to RAIs on the design and evaluation of the structural integrity of ESBWR reactor pressure vessel (RPV) internals.

Following the issuance of the FSER on March 9, 2011, the NRC staff identified issues applicable to the ESBWR steam dryer structural analysis based on information obtained during the NRC review of a license amendment request for a power uprate at an operating BWR nuclear power plant. Based on this information, the NRC staff issued RAIs to GEH in connection with the steam dryer structural analysis. As a result of the resolution of those issues, GEH revised the DCD to withdraw the licensing topical reports (LTRs) that addressed the ESBWR steam dryer structural evaluation, and to reference new engineering reports that describe the updated ESBWR steam dryer analysis methodology. The NRC staff has reviewed the revised DCD sections, the new GEH engineering reports, and RAI responses. In addition, the NRC staff conducted several public meetings with GEH to discuss the ESBWR steam dryer analysis. The staff also performed an audit of the GEH steam dryer analysis methodology at the GEH facility in Wilmington, NC, in March 2012, and performed a vendor inspection at that facility of the quality assurance program for GEH engineering methods in April 2012. The following LTRs were withdrawn by GEH:

- GEH, LTR NEDE-33313 and NEDE-33313P, "ESBWR Steam Dryer Structural Evaluation," all revisions

- GEH, LTR NEDE-33312 and NEDE-33312P, "ESBWR Steam Dryer Acoustic Load Definition," all revisions

- GEH, LTR NEDC-33408 and NEDC-33408P, "ESBWR Steam Dryer-Plant Based Load Evaluation Methodology," all revisions

- LTR NEDC-33408, Supplement 1 and NEDC-33408P, Supplement 1, "ESBWR Steam Dryer – Plant Based Load Evaluation Methodology Supplement 1," all revisions

As a result of the withdrawal of the LTRs on the ESBWR steam dryer by GEH, this supplemental FSER completely replaces the NRC staff review and conclusions in the applicable NRC safety evaluation reports for the GEH LTRs on the ESBWR steam dryer. The NRC safety evaluation reports on the original GEH topical reports for the ESBWR steam dryer are hereby withdrawn. This supplemental FSER also completely replaces the discussion of RAI 3.9-58 in Section 3.9.2.3.3, "Staff Evaluation," and Section 3.9.2.3.4, "Conclusion," of the March 9, 2011, FSER. In the SER for Reactor Internals Flow-Induced Vibration Program NEDE-33259P, page 16, Reference 1, to the audit response document was incorrectly referred to as NEDO-33312, "ESBWR Steam Dryer Acoustic Load Definition." This supplemental FSER corrects the reference to NEDO-33312 in the SER for NEDE-33259P. The new GEH engineering reports are addressed in this supplemental FSER.

This supplemental FSER replaces in its entirety Section 3.9.5, “Reactor Pressure Vessel Internals,” of the FSER issued on March 9, 2011. Information related to ESBWR RPV internals other than the steam dryer (such as core support structures) has been copied from the FSER and placed in this supplemental FSER to provide the description of the NRC staff review of all ESBWR RPV internals in one location.

GEH has submitted information in support of its DCD that it considers “proprietary” within the meaning of the definition provided in Title 10 of the Code of Federal Regulations (10 CFR) 2.390(b)(5), “Public inspections, exemptions, requests for withholding.” The applicant has requested that this information be withheld from public disclosure and the NRC staff agrees that the submitted information sought to be withheld includes proprietary commercial information and should be withheld from public disclosure. This chapter of the NRC staff’s evaluation includes proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information will appear as a blank space surrounded by “square brackets” as follows:

[[ ]]

The complete text of this chapter, including proprietary information, can be found at Agencywide Documents Access and Management System (ADAMS) Accession Number ML14155A342. This document can be accessed by those who have specific authorization to access the applicant’s proprietary information.

3.9.5.1 Regulatory Criteria

The following regulatory requirements provide the basis for the acceptance criteria for the staff’s review:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, General Design Criterion (GDC) 1, “Quality Standards and Records,” as it relates to designing RPV internals (reactor internals) to appropriate quality standards commensurate with the importance of the safety functions to be performed

- GDC 2, “Design Bases for Protection against Natural Phenomena,” and Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” to 10 CFR Part 50, as they relate to designing reactor internals to withstand the effects of earthquakes without loss of capability to perform their safety functions

- GDC 4, “Environmental and Dynamic Effects Design Bases,” as it relates to designing reactor internals to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including loss of coolant accidents (LOCAs)

- GDC 10, “Reactor Design,” as it relates to designing reactor internals with appropriate margin to ensure that adequate structural support of the reactor core is provided such that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences

- 10 CFR 50.55a, as it relates to designing, fabricating, testing, and inspecting reactor internals to appropriate quality standards commensurate with the importance of the safety functions to be performed

- 10 CFR 52.47, “Contents of Applications; Technical Information,” as it relates to the application for design certification specifying design information sufficiently detailed to permit the preparation of procurement specifications and construction and installation specifications by an applicant

The following regulatory guidance provides the basis for the acceptance criteria for the staff’s review:

- RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” as it relates to verifying the structural integrity of reactor internals for steady-state and transient flow-induced vibration (FIV) loading

- ASME *Boiler & Pressure Vessel Code* (BPV Code), Section III, Division 1, 2001 Edition and Addenda through 2003, consistent with ESBWR DCD Tier 2, Table 1.9-22

3.9.5.2 *Summary of Technical Information*

DCD Tier 2, Revision 10, Section 3.9.5, addresses the RPV internals. It also references as requirements the information in three GEH Engineering Reports on the ESBWR steam dryer structural evaluation methodology:

- GE Hitachi Nuclear Energy, “ESBWR Steam Dryer Acoustic Load Definition,” NEDE-33312P, Revision 5, Class III (Proprietary), December 2013.

- GE Hitachi Nuclear Energy, “ESBWR Steam Dryer - Plant Based Load Evaluation Methodology - PBLE01 Model Description,” NEDE-33408P, Revision 5, Class III (Proprietary), December 2013.

- GE Hitachi Nuclear Energy, “ESBWR Steam Dryer Structural Evaluation,” NEDE-33313P, Revision 5, Class III (Proprietary), December 2013.

As used in this supplemental FSER, RPV internals consist of all structures and mechanical components inside the reactor vessel, with the exception of the fuel system design, including reactor fuel assemblies and reactivity control elements, which are addressed in DCD Tier 2, Section 4.2.

RPV internals are constructed and tested to quality standards commensurate with the importance of the safety functions to be performed. In accordance with the applicable NRC regulations, RPV internals are designed with appropriate margins to withstand the effects of normal operation; anticipated operational occurrences; natural phenomena, such as earthquakes; and postulated accidents, including the design basis LOCA.

The NRC staff followed the guidance in Standard Review Plan (SRP) Section 3.9.5, “Reactor Pressure Vessel Internals,” Revision 3, dated March 2007, and RG 1.20, Revision 3, dated March 2007, during the evaluation of the RPV internals in reviewing the ESBWR design certification.

I. Steam Dryer Initial Design

The ESBWR steam dryer design builds on the successful operating experience of the Advanced Boiling-Water Reactor (ABWR) steam dryer. The ESBWR steam dryer design also draws experience from operating plant replacement steam dryer program fabrication, testing and performance. The ESBWR RPV has a larger inner diameter at the vessel flange than the ABWR, which allows dryer banks to be extended, thereby accommodating a higher steam flow. The ESBWR design certification applicant has developed a methodology for evaluating the effects of the acoustic response of the reactor and main steam system on the steam dryer as described in NEDE-33312P. Although the ESBWR steam dryer is designed to have a larger diameter and wider vane banks to accommodate a higher steam flow, the vane height, skirt length, outer hood setback from the main steam nozzle, and water submergence are similar to the ABWR steam dryer. In the detailed design of the steam dryer, the fatigue analysis performed for the ESBWR steam dryer uses a fatigue stress amplitude limit of 93.7 megapascals (MPa) (13,600 [pound-force per square inch (psi)], as described in Section 7.1 of NEDE-33313P. For additional conservatism in the predictive analysis, the analysis stress results performed during the detailed design process will also meet a minimum alternating stress ratio (MASR) of 2.0 between the analysis results and the fatigue acceptance limit. Design loads for the steam dryer are based on evaluation of the ASME BPV Code load combinations specified in the DCD. These load combinations consist of deadweight loads, static and fluctuating differential pressure loads (including turbulent and acoustic sources), seismic, thermal, and transient acoustic and fluid impact loads.

II. Steam Dryer As-Built Analysis

The as-built steam dryer will be assessed as described in Section 7.2 of NEDE-33313P, including reconciliation of changes between the as-designed and as-built steam dryers, adjustments to the structural finite element (FE) model, updated bias and uncertainty based on testing, and the resulting updated stresses and stress ratios. Steam dryer dynamic testing is performed with a sufficient number of excitation locations to ensure adequate coverage of the dryer. Uncertainties are addressed in the comparison of predicted mode shapes with those measured during the dryer dynamic testing (i.e., boundary conditions and dryer support). Differences are assessed between predicted resonance frequencies and those measured during the dryer dynamic testing to ensure worst-case coupling between peak excitation and peak response is captured. DCD Tier 1, Table 2.1.1-3, in ITAAC 16, requires the licensee to verify that the as-built steam dryer fatigue analysis provides at least an MASR of 2.0 to the fatigue stress amplitude limit of 93.7 MPa (13,600 psi). On-dryer instrumentation sensor specifications, sensor locations and correlations between sensors and peak stress locations on the upper and lower dryer are identified, as well as biases and uncertainties associated with the sensors and data acquisition system. Acceptance limits for each sensor with supporting calculations (spectra and time histories) are provided, with the limits derived from calculations using the minimum load case method as described in Section 9 of NEDE-33313P. Limit curves for power ascension are based on the worst case of both the design basis calculations that use the end-to-end Grand Gulf Nuclear Station (GGNS) steam dryer bias and uncertainty and those from the as-built steam dryer calculations that use the combined FE structural and Plant Based Load Evaluation (PBLE01) biases and uncertainties.

III. Power Ascension Monitoring and Inspections

A Steam Dryer Monitoring Plan (SDMP) will be developed to implement the Comprehensive Vibration Assessment Program for the steam dryer methodology consistent with RG 1.20. The SDMP will reflect industry experience with the performance of steam dryer power ascension testing. The SDMP includes criteria for comparison and evaluation of projected strain levels with data obtained from the on-dryer instrumentation, acceptance limits developed for selected on-dryer strain gage and accelerometer locations, tables of predicted steam dryer stresses at 100 percent power, strain amplitudes and power spectral densities (PSDs) at strain gage locations, predicted acceleration amplitudes and PSDs at acceleration locations, and maximum stresses and locations. The SDMP provides directions for establishing correlations between measured accelerations and strains and the corresponding maximum stresses; identification of steam dryer strain gage locations for which limit curves will be developed and criteria for selection of those locations; and the methodology for developing projected strain levels for the next power level and for full power. The SDMP specifies details of the installation and calibration of the steam dryer instrumentation with the instrumentation mounted and calibrated in accordance with the manufacturers' instructions to accurately measure the dynamic response. The SDMP defines specific assessment points (approximately 75, 85, and 95 percent power) during power ascension and activities to be accomplished during assessment points. After full power has been achieved, data at the full power level will be provided to the NRC within 72 hours, and a full stress analysis report and evaluation will be provided to the NRC within 90 days of reaching the full power level. For the confirmatory stress analysis, a structural assessment is performed to benchmark the FE model strain and acceleration predictions against the measured data. The dryer stresses are determined using the loads measured on the surface of the dryer and adjusted for end-to-end bias and uncertainties determined from the FE model benchmark. A fatigue stress amplitude limit of 93.7 MPa (13,600 psi) with an MASR of 1.0 is used as the acceptance limit for this confirmatory stress analysis. This confirmatory stress analysis will demonstrate that the steam dryer will maintain its structural integrity over its design life considering variations in plant parameters (such as reactor pressure and core flow rate). During the first two scheduled refueling outages after reaching full power conditions, a visual inspection will be conducted of all accessible areas and susceptible locations of the steam dryer in accordance with accepted industry guidance on steam dryer inspections. At the end of the second refueling outage following full power operation, an updated SDMP reflecting a long-term inspection plan based on plant-specific and industry operating experience will be implemented.

IV. Subsequent Plants

The ESBWR steam dryer is a prototype steam dryer under the guidance in RG 1.20. Because the ESBWR steam dryer is considered a prototype in the design certification, each subsequent ESBWR steam dryer will also be treated as a prototype in its design and analysis. Subsequent ESBWR steam dryers could only be treated as non-prototypes under the provisions of RG 1.20 through future amendment to the design certification or NRC approval of a plant-specific departure and exemption from design certification requirements.

3.9.5.2.1 Identification and Discussion of Structural and Functional Integrity of the Major Reactor Pressure Vessel Internals, including Core Support Structures

3.9.5.2.1.1 Safety Classification of Reactor Pressure Vessel Internals

DCD Tier 2, Section 3.9.5, identifies the following structures as core support (CS) structures:

- shroud
- shroud support
- core plate, including its hardware
- top guide, including its hardware
- orificed and peripheral fuel supports
- control rod guide tubes (CRGTs)
- nonpressure boundary portion of control rod drive (CRD) housings

The RPV internals include the following safety-related components:

- standby liquid control (SLC) system header, spargers, and piping
- in-core guide tubes (ICGTs) and stabilizers
- nonpressure boundary portion of in-core housings

In addition, the RPV internals include the following nonsafety-related components:

- chimney and partition
- chimney head and steam separator assembly
- steam dryer assembly
- feedwater (FW) spargers
- RPV vent assembly
- surveillance sample holders

3.9.5.2.1.2 Functional Description of Reactor Pressure Vessel Internals

The minimum floodable inner volume of the RPV includes the volume up to the level of the gravity-driven cooling system (GDCS) equalizing nozzles. One end of the reactor internals, where the shroud, chimney, steam separators, and guide tubes are located, is unrestricted and therefore free to expand.

The CS structures form partitions within the reactor vessel to sustain pressure differentials across the partitions, direct the flow of the coolant water, and locate and support the fuel assemblies. The shroud and chimney make up a stainless steel (SS) cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. The RPV shroud support is a ring supporting the core plate and a series of vertical support legs supporting the ring. The core plate provides lateral support and guidance for the CRGTs, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The entire CS structure is bolted to a support ring with 12 support legs that are welded to the bottom of the RPV. The top guide consists of a circular plate with square openings for fuel assemblies. Each opening provides a lateral support and guidance for four, or in some cases fewer, fuel assemblies. The top guide is mechanically attached to the top of the shroud. The chimney is bolted to the top surface of the top guide. Each peripheral fuel support is located at the outer edge of the active core, supports one fuel assembly, and contains an orifice to ensure proper coolant flow to the supported fuel assembly. Each orificed fuel support holds four fuel assemblies vertically and horizontally and has four orifices to provide proper coolant flow distribution to each of the four assemblies. Each orificed fuel support rests on top of a CRGT, and a control rod passes through a cruciform opening in the center of the support. The CRGTs are located inside the vessel and extend from the top of the CRD housings up through holes in the core plate. The CRD housing supports the bottom of the CRGT and transmits the weight of the CRGT, orificed fuel support, and fuel assemblies to the reactor vessel lower head.

The reactor vessel internals direct and control flow through the core and support both safety-related and nonsafety-related functions. The chimney is a cylindrical structure that is mounted on the top guide and supports the steam separator assembly. The chimney provides the driving head necessary to sustain the natural circulation flow. The chimney forms the annulus separating the upward flow of the steam/water mixture exiting the core from the FW and the subcooled recirculation flow returning downward from the steam separators. Inside the chimney are partitions that channel the flow of the steam/water mixture exiting the core into smaller chimney sections to limit cross flow and flow instabilities. The partitions do not extend to the top of the chimney, thereby forming a mixing chamber or a discharge plenum for the steam/water mixture before entering the steam separators. Individual SS axial-flow steam separators are supported on and attached to the top of standpipes that are welded into the chimney head. In each separator, the steam/water mixture rising through the standpipe passes vanes that impart a spin and establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the downcomer annulus.

The steam dryer assembly consists of multiple banks of dryer units mounted on a common structure, which is supported by brackets welded to the reactor vessel wall. The dryer assembly includes the dryer banks, drain collecting trough, drain duct, and a skirt that forms a water seal extending below the upper end of the separator. Reactor vessel internal stops limit the upward and radial movement of the dryer assembly if it is subjected to blowdown and seismic loads. These stops are arranged to permit differential thermal expansion of the dryer assembly with respect to the RPV.

The FW spargers deliver makeup water to the reactor during plant startup, power generation, and shutdown modes of operation. The FW spargers are SS headers located in the mixing plenum above the downcomer annulus. A separate sparger in two halves is fitted to each FW nozzle by a tee and is shaped to conform to the curve of the vessel wall. FW enters the center of the spargers and is discharged radially inward to mix the cooler FW with the downcomer flow from the steam separators and steam dryer.

Each of the two SLC system nozzles supplies four injection lines via SLC header and distribution lines. The injection lines have nozzles penetrating the shroud at four different elevations. The injection lines enable the sodium pentaborate solution to be injected around the periphery of the core.

The RPV vent assembly passes steam and noncondensable gases from the reactor head to the steamlines during startup and operation. The ICGT protect the in-core instrumentation from the flow of water in the bottom head plenum and provide a means of positioning fixed detectors in the core. The ends of the ICGTs are supported by the core plate and the in-core housing, and a latticework of tie bars connected to the core support ring provides additional lateral support.

The surveillance sample holders are welded baskets hanging from the brackets attached to the inside of the reactor vessel wall and extend to the midheight of the active core. The radial positions of the basket are such that the impact and tensile specimens, which are carried in the baskets, are exposed to the same environment and maximum neutron fluxes experienced by the reactor vessel itself.

3.9.5.2.1.3 Flow Induced Vibration Assessment Program

Appendix 3L to DCD Tier 2, Revision 10, outlines a comprehensive vibration assessment program for evaluating and ensuring the integrity of reactor internal components subject to steady-state and transient flow conditions. This program includes an analytical evaluation phase, a startup test phase, and an inspection phase, consistent with the guidelines of RG 1.20, and is intended to verify that no FIV problems exist for the as-built condition of the RPV internals.

The first part of the evaluation identifies components that are deemed susceptible to FIV and for which additional evaluation and potential instrumentation for startup testing may be necessary. The chimney partition, SLC internal piping, and the steam dryer have been identified as components for additional FIV analysis and startup test instrumentation.

The second part of the evaluation will establish finite element analyses and establish correlation functions, based on prior data, where available, to determine stress levels for those components deemed to require additional evaluation to confirm their adequacy and to confirm that stresses will be maintained below the fatigue stress limits of 68.95 MPa [10 kilopounds-force per square inch (ksi)] for all components, except the steam dryer (which is addressed separately). The analyses will include the determination of vibration frequencies and mode shapes, as necessary. The applicant has presented the second part of the evaluation for internal components other than the steam dryer in Appendix 3L to DCD Tier 2 and NEDE-33259P, “Reactor Internals Flow Induced Vibration Program.” The applicant has presented the results of these evaluations for the steam dryer in NEDE-33312P, NEDE-33313P, and NEDE-33408P. As outlined in Appendix 3L to DCD Tier 2, the applicant conducted analyses for the chimney partition, as described in NEDE-33259P, because the chimney partition is a component that has never been subjected to preoperational or initial startup testing. Appendix 3L to DCD Tier 2 also outlines the steam dryer evaluation program.

NEDE-33259P evaluates internal components, other than the steam dryer, to establish the need for further analysis and testing. Each of the other component designs and operating conditions are compared for similarity with those of the ABWR, four of which are operating. As a result of this comparison, in addition to the chimney partition and steam dryer, the shroud/chimney assembly, the chimney head/steam separator assembly, and the SLC lines were determined to require further analysis as part of the ESBWR FIV prototype test program. Because of their similarities to the operating ABWR reactors, further evaluation is not considered necessary for the remaining RPV internals components.

3.9.5.2.2 Design Criteria Used for Assessing the Adequacy of Core Support Structures

DCD Tier 2, Section 3.9.5.4, provides the following criteria for assessing the adequacy of CS structures:

- The design and construction of the CS structures are in accordance with the provisions of the ASME BPV Code, Section III, Subsection NG.

- The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals other than the CS structures satisfy ASME BPV Code, Section III, Subsection NG-3000, and must be constructed so as not to adversely affect the integrity of the CS structures, as stipulated in ASME BPV Code, Section III, Subarticle NG-1122.

3.9.5.2.3 Criteria Used for Assessing the Adequacy of Steam Dryer and Chimney Assemblies, Including the Information from Appendix 3L to DCD Tier 2

Appendix 3L to DCD Tier 2 describes potential FIV monitoring of reactor internals in an ESBWR prototype plant. The evaluation process identified both the chimney, a component new to the ESBWR design, and the steam dryer as structures that will be monitored during power ascension in the ESBWR prototype. The steam dryer was chosen based on industry experience where some steam dryers in operating BWR plants have experienced structural degradation as a result of fatigue failure under extended power uprate (EPU) conditions.

For normal operating conditions, Appendix 3L to DCD Tier 2 has identified FIV analysis and FIV test programs to demonstrate the adequacy of the components and to confirm that their stresses are bounded by fatigue limits in DCD Tier 2, Section 3.9.2.3.

3.9.5.2.4 Criteria Used for Assessing the Adequacy of Internal Structures Other Than Steam Dryer and Chimney Assemblies, Including the Information from NEDE-33259P

DCD Tier 2, Tables 3.9-4 through 3.9-7, identify the stress, deformation, and fatigue limit criteria of safety-related components from which appropriate criteria are selected for a specific component and loading condition. The applicant stated that the criteria are based on applicable codes and standards for similar equipment, manufacturing standards, or empirical methods, based on field experience and testing, and satisfy ASME BPV Code, Section III, Subsection NG-3000. The stated construction philosophy is to provide adequate clearances for components that must move during emergency and faulted conditions and not adversely affect the integrity of the CS structure, in accordance with ASME BPV Code, Section III, Paragraph NG-1122. For the other components designated as nonsafety-class internals, ASME BPV Code design provisions are followed where applicable. Otherwise, accepted industry or engineering practices are used.

As discussed in Section 3.9.5.2.1 of this supplemental FSER, Appendix 3L to DCD Tier 2 and NEDE-33259P describe a method for establishing component adequacy for FIV under normal operating conditions, with the ultimate goal of showing that the fatigue stresses in the components (except the steam dryer) are less than 68.95 MPa (10 ksi). The criterion used in NEDE-33259P to judge which components warrant additional evaluation and which components are considered acceptable and require no additional evaluation is to compare their design and operating conditions for similarity with those of the ABWR and operating BWRs. Resolution of RAI 3.9-75, S01 in Section 3.9.2 of this supplemental FSER provides further information, including a discussion of the classification of the ESBWR reactor internals as a prototype, in accordance with RG 1.20.

3.9.5.2.5 Design Basis Loading Events

DCD Tier 2, Section 3.9.5.3, states that CS structures and safety-related internal components must satisfy the safety design basis (DCD Tier 2, Section 3.9.5.4) for the following three load events:

1. RPV line break accident, which is a break in any one line between the reactor vessel nozzle and the isolation valve resulting in significant pressure differential across some of the structures within the reactor and reactor building vibration (RBV) caused by suppression pool dynamics

2. earthquakes that subject the CS structures and reactor internals to significant forces as a result of ground motion and consequent RBV

3. safety relief valve (SRV) or depressurization valve discharge resulting in RBV caused by suppression pool dynamics and structural feedback

3.9.5.2.5.1 Load Combinations and Stress Limits

DCD Tier 2, Section 3.9.1.4, discusses the evaluation methods and stress limits used for the faulted conditions. DCD Tier 2, Table 3.9-2, presents load combinations and acceptance criteria for CS structures.

The applicant used the TRACG computer code to determine pressure differences for reactor internals during the events under normal, upset, emergency, and faulted conditions. The code analyzes the transient conditions within the reactor vessel following anticipated operational occurrences, infrequent events, and accidents (e.g., LOCAs). To determine the maximum pressure differences across the reactor internals, the applicant performed a statistical uncertainty study. To determine the upper bound pressure difference, two standard deviations of the uncertainty were added to the normal pressure differences.

In DCD Tier 2, Section 3.7, the applicant described a dynamic analysis method used to determine the loads resulting from an earthquake and other building vibrations acting on the reactor vessel internals.

3.9.5.2.5.2 Flow Induced Vibration

For FIV, the normal operating pressure differential drives the coolant flow that impinges on and loads the reactor internal components in different ways. Table 3 of NEDE-33259P presents flow velocities and vortex shedding frequencies for ESBWR and ABWR components deemed similar. According to Appendix 3L to DCD Tier 2, the applicant has completed two-phase hydraulic flow testing simulating expected reactor flow conditions for the chimney partition, and the pressure loading function has been determined. Appendix 3L to DCD Tier 2 and NEDE-33312P, NEDE-33313P, and NEDE-33408P describe the FIV evaluation program for the ESBWR steam dryer.

3.9.5.2.6 Design Bases

DCD Tier 2, Section 3.9.5.4, states that the reactor internals, including CS structures, must meet the following safety-related design bases:

- The reactor nozzles and internals shall be so arranged as to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

- Deformation of internals shall be limited to ensure that the control rods and core standby cooling system can perform their safety-related functions.

- Mechanical design of applicable structures shall ensure that the above safety-related design bases are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

The reactor internals, including CS structures, shall be designed to the following power generation design bases:

- The internals shall provide the proper coolant distribution during all anticipated normal operating conditions to full-power operation of the core without fuel damage.

- The internals shall be arranged to facilitate refueling operations.

- The internals shall be designed to facilitate inspection.

The applicant stated that the design loading categories for the CS structures and safety class internals stress limits are consistent with ASME BPV Code, Section III, Subsection NG. The stress and fatigue limits for the CS structures are also consistent with ASME BPV Code, Section III, Subsection NG.

The applicant provided the stress, deformation, and fatigue criteria for safety-related reactor internals (except CS structures), which are based on the criteria established in applicable codes and standards for similar equipment, by manufacturers' standards, or by empirical methods based on field experience and testing. These criteria include the minimum safety factors provided for each of the four ASME BPV Code, Section III, service conditions (i.e., normal, upset, emergency, and faulted).

The applicant stated that the design criteria, loading conditions, and analyses that provide the basis for the design of the safety class reactor internals, other than the CS structures, satisfy ASME BPV Code, Article NG-3000, and are constructed so as not to adversely affect the integrity of the CS structures (ASME BPV Code, Section III, paragraph NG-1122).

Appendix 3L to DCD Tier 2 states that the primary design basis is for ESBWR internals to safely withstand expected FIV forces. As discussed in Section 3.9.5.2.1 of this report, dynamic stress analysis using finite element model (FEM) analysis has been or will be performed for all of the reactor internal components, including the steam dryer that will be instrumented during startup testing. NEDE-33259P includes these results for all of the vessel internals except the steam dryer; DCD Tier 2, Appendix 3L, includes some of the results for the chimney partition. ITAAC 8b of DCD Tier 1, Table 2.1.1-3, requires performance of a stress analysis for the as-built configuration of the steam dryer, chimney, chimney partitions, and related components to verify that the design limits of ASME BPV Code, Section III, Article NG-3000 have been satisfied.

The applicant determined the fundamental frequency of the chimney partition (approximately 54 hertz (Hz) to be much larger than the frequency of the maximum peak-to-peak pressure fluctuation (2 Hz). Therefore, the applicant performed an equivalent static analysis to show that the fatigue stress limits bounded the calculated stress.

3.9.5.3 *Staff Evaluation*

3.9.5.3.1 Identification and Discussion of the Structural and Functional Integrity of the Major Reactor Pressure Vessel Internals, Including Core Support Structures

As described in Section 3.9.5.2.1 of this supplemental FSER, the ESBWR design certification applicant identified the major safety-related reactor internal structures, including CS structures, for the ESBWR. In addition, the applicant identified the internal structures that are not safety-related. DCD Tier 2, Section 3.9.5, summarizes the functions of the internals. The NRC staff finds that the applicant has adequately discussed the physical arrangement of these components inside the vessel, which provides axial support and lateral retention of the internal assemblies and components. The applicant used the method described in Appendix 3L to DCD Tier 2 and NEDE-33259P for establishing component integrity for FIV under normal operating conditions to show that the fatigue stresses in the internal components are bounded by the fatigue limits in DCD Tier 2, Section 3.9.2.3.

As discussed in Appendix 3L to DCD Tier 2, related to FIV, the applicant indicated that many of the reactor internal components require additional analysis to demonstrate their design adequacy. Furthermore, FIV evaluation analyses are needed for components with significantly different features and loading conditions from valid prototype reactor internals, in accordance with RG 1.20 and SRP Section 3.9.5. Therefore, in RAI 3.9-132, the staff asked the applicant to provide detailed descriptions of the components, their boundary conditions, the load definitions, the design criteria, the bias errors and uncertainties (B/Us), and the evaluation analyses for the ESBWR shroud/chimney assembly, the chimney head/steam separator assembly, the SLC lines, the CRGTs and CRD housings, the in-core monitor guide tubes (ICMGTs) and housings, the chimney partition, and the steam dryer.

In response, the applicant stated that it would submit a revision of NEDE-33259P to account for ongoing design changes of the reactor internals, including additional analysis for most of the components identified in RAI 3.9-132. The applicant determined that no analyses were necessary for the CRGTs and CRD housings and the ICMGTs and housings because of their similarity to current ABWR designs, as discussed in NEDE-33259P, Revision 2. RAI 3.9-132 was tracked as an open item in the SER with open items.

The staff's review of the revised NEDE-33259P addresses the resolution of several other RAIs related to the RPV internals FIV program, in addition to resolution of RAI 3.9-132, and is documented in the staff's safety evaluation report on NEDE-33259P. After its review of Revision 2 of the LTR, the staff held an audit at the applicant's offices in Wilmington, NC, on August 25, 2009. During the technical discussions of FIV related issues, as documented in the GEH "Response to NRC Report of the August 25, 2009, and September 9, 2009, Regulatory Audit of Reactor Pressure Economic Simplified Boiling Water Reactor," dated October 8, 2009, the applicant indicated that, because design details of the chimney partition were still being evaluated, the detailed design of the chimney partition is not complete. The applicant committed to complete the final design of the chimney partition and FIV stress analyses as part of ITAAC 8b (listed in Table 2.1.1-3 of DCD Tier 1) in order to verify this design commitment has been implemented. The revised NEDE-33259P adequately addressed the staff's concerns and provided appropriate analysis. Therefore, RAI 3.9-132 and its associated open item are resolved. The staff also concludes that the addition of ITAAC 8b in DCD is sufficient to resolve the associated staff audit comment.

3.9.5.3.2 Criteria Used for Assessing the Adequacy of Core Support Structures

The staff finds that the criteria proposed by the applicant for assessing the adequacy of CS structures are acceptable because they utilize the requirements of ASME BPV Code, Section III, Division 1, Subsection NG.

3.9.5.3.3 Criteria Used for Assessing the Adequacy of Steam Dryer and Chimney Assemblies, Including the Information from Appendix 3L to DCD Tier 2

The NRC staff finds the use of flow testing and structural dynamic analysis appropriate for assessing the adequacy of the chimney assembly and steam dryer because the chimney is a new component to the ESBWR design, and some steam dryers in operating reactors have experienced structural degradation resulting from fatigue failure under EPU conditions. The staff also finds the use of a fatigue limitation as described in DCD Tier 2, Section 3.9.2.3 acceptable because it satisfies, or is more conservative than, the ASME BPV Code provisions.

The NRC staff issued numerous RAIs requesting information on the steam dryer design, analysis, construction, monitoring, and inspection. In this supplemental FSER, the staff discusses only RAIs and their responses necessary to describe the final NRC position on the process for evaluating the adequacy of the ESBWR steam dryer.

3.9.5.3.3.1 Summary of ESBWR Steam Dryer Design Approach, Analysis Methodology, and Startup Monitoring

The ESBWR design certification applicant describes its procedures for designing and assessing the structural integrity of ESBWR steam dryers in Section 3L.4 of Appendix 3L to DCD Tier 2. Section 3L.4.1 outlines the key design features of the ESBWR steam dryer, and how it will be based on successfully operating ABWR dryers. Section 3L.4.2 specifies the use of materials not conducive to corrosion and stress-corrosion cracking. Section 3L.4.3 lists the dryer loading combinations, based on the ASME BPV Code, considered in the structural integrity analysis. The loads include deadweight, seismic, thermal, transient acoustic and fluid impact loads, static pressure, and fluctuating differential pressure loads (including turbulent and acoustic sources).

DCD Tier 2, Section 3L.4.4, specifies how the fluctuating differential pressures acting on an ESBWR steam dryer are defined and references NEDE-33408P and NEDE-33312P. Section 3L.4.5 describes the steam dryer structural analysis procedure for both static and alternating stresses, referencing NEDE-33313P.

The acoustic loading and structural analysis procedures will be used during the ESBWR steam dryer detailed design phase, and later during verification phases once in-plant instrumented ESBWR dryer data are available during startup monitoring, as described in Section 3L.4.6. COL information items and ITAAC, described elsewhere in this supplemental FSER, will provide for appropriate interaction with the NRC by COL applicants and licensees to confirm that the as-built and as-operated steam dryer satisfies the applicable design specification criteria.

The ESBWR steam dryer design approach, modeling and analysis methodologies, and verification testing plans are covered in the DCD and referenced engineering reports. The detailed ESBWR dryer design is not yet specified, nor are the associated expected pressure loading and alternating stresses, which depend on the final design specification of the dryer. Therefore, the actual pressure loading and alternating stresses are not evaluated in this supplemental FSER. Detailed steam dryer design, pressure loading, and stress analysis results will be made available as part of COL action items or ITAAC, along with subsequent stress verification measurements and studies during and after plant startup.

The major overall design, analysis, and verification elements are:

- Complete the detailed design of the ESBWR steam dryer based on existing ABWR dryers, which have experienced no structural integrity problems during operation. Ensure the ESBWR plants have no flow-excited acoustic resonances in main steam line (MSL) valve standpipes (which caused dryer damage in the Quad Cities (QC) Units 1 and 2 nuclear power plants).

- Simulate the fluctuating differential pressure loading distribution over the ESBWR dryer using the PBLE01 methodology, which is based on instrumented dryer surface pressure measurements and acoustic modeling methods. Use worst-case loading based on measurements from steam dryers in operating BWRs as the ESBWR design specification.

- Apply the PBLE01 pressure loading to a structural FEM of the as-designed ESBWR dryer, and compute time histories of stresses within the dryer. Apply end-to-end bias errors and uncertainties from benchmarking of the overall analysis procedure, computed using measurements and simulations of the alternating strains in the Grand Gulf Nuclear Station (GGNS) replacement steam dryer (RSD) during the GGNS initial EPU power ascension. Also, compute stress amplifications near concentrations (such as welds) using well-established standards and mesh convergence studies.

- Ensure the adjusted dryer peak stresses are less than material fatigue limits for the dryer (13,600 psi or 93.7 MPa) with specified margin.

- Update the ESBWR dryer stress calculations to reflect as-built dimensions and properties. The as-built stress calculations will use updated bias errors and uncertainties based on dynamic measurements made on the as-built ESBWR dryer.

- Use the computed dryer stresses to specify optimal locations for on-dryer instrumentation for measuring surface pressures, as well as alternating strains and vibrations. Establish allowable power ascension limits for the measured levels based on worst-case computed stresses to date (maximum of the as-designed and as-built calculations).

- Measure strains, accelerations, and pressures on the actual ESBWR dryer during power ascension. Re-benchmark end-to-end bias errors and uncertainties at 75 percent of power using approximate scaling methods (such as F-factor and root mean square (RMS)) used successfully during previous EPU power ascensions, update on-dryer sensor limits, and project sensor levels to higher powers. If sensor limits are exceeded, reduce power and recompute dryer stresses using exact methods.

- Complete the design, construction, instrumentation, and power ascension process through the COL application and ITAAC process. Provide final ESBWR dryer stress calculations within 90 days of completion of power ascension.

- Treat all future ESBWR dryers as prototypes and follow the procedure above. The GGNS-based bias errors and uncertainties used for the initial ESBWR dryer design will be replaced by ESBWR-specific bias errors and uncertainties for subsequent designs.

To provide assurance that the ESBWR steam dryer will not experience fatigue cracking, the ESBWR design certification applicant has agreed to two additional conservative elements in the dryer analysis procedure: (1) including an additional factor of 2.0 in their minimum alternating stress ratio (MASR) for the as-designed and as-built dryer (i.e., completing the design details and manufacture of the steam dryer such that stresses remain less than one-half of the ASME BPV Code fatigue limit of 93.7 MPa (13.6 ksi)), and (2) not taking credit for positive (overprediction) bias errors in the detailed design process from their PBLE01 benchmarking based on the GGNS RSD. Also, in the event the bias errors and uncertainties are insufficient for ESBWR on-dryer strain and acceleration simulations to bound those measured during startup testing, the ESBWR design certification applicant specifies that the bias errors and uncertainties be increased so that all measurements are bounded by simulations, or quantitatively assess the impact of the strain and acceleration underpredictions on dryer fatigue life.

The following sections of this supplemental FSER describe in more detail: the steam dryer acoustic loading methodology, minimization of dryer loading caused by flow over standpipes in

MSLs, how specific acoustic loading for the ESBWR dryer will be derived and minimized, and the structural vibration and stress analysis approach, including stress concentration assessments near welds. The procedures are demonstrated and benchmarked by the ESBWR design certification applicant using the GGNS RSD.

In response to several RAIs, the ESBWR design certification applicant updated NEDE-33312P, NEDE-33313P, and NEDE-33408P to describe the process for ESBWR steam dryer evaluation. To ensure that the description in these documents is clear and consistent with RAI responses, as well as steam dryer evaluation processes recently reviewed and approved by the NRC staff, the NRC staff requested in RAI 3.9-299 that the applicant clarify the description of the steam dryer evaluation process in its engineering reports. In response to RAI 3.9-299, the applicant revised Section 1.0, "Introduction," in NEDE-33313P to provide an overall description of the ESBWR steam dryer evaluation process.

NEDE-33313P now states:

I. Design

1. Maintain key aspects of the ABWR steam dryer and steam plenum region geometries that have provided satisfactory performance with similar rated steam flow, reactor size, and steam outlet nozzle configuration.

   2.0 Evaluate the expected acoustic response of the reactor steam dome and main steam system in order to avoid or eliminate geometries that can result in large acoustic loads.

   3.0 Analyze the steam dryer design with independent sets of design loads that include high amplitude loads covering a wide frequency spectrum. These design loads are developed from instrumented plant steam dryer data using the PBLE01 methodology. End-to-end bias and uncertainty developed from GGNS steam dryer data is applied to the design basis, but without credit for positive bias that works in the direction of reduced fatigue margin. For ESBWR, the projected load definitions bound the ABWR test data when extrapolated [[

   .]]

   4.0 Demonstrate that the fatigue analysis results for the as-designed steam dryer maximum calculated alternating stress intensity meet or exceed an MASR of 2.0 to the allowable alternating stress intensity of 93.7 MPa (13,600 psi).

   5.0 Demonstrate that the primary stress results for the as-designed steam dryer meet the acceptance criteria for the normal, upset, emergency, and faulted load combinations.

II. As-Built Steam Dryer

1. Address any changes between the as-designed and as-built steam dryer.

2. Perform dynamic testing, "frequency response testing," of the fabricated steam dryer to compare the predicted versus measured frequency response. Define the FEM bias and uncertainty [[ ]] based on the results of the comparison. Recalculate stress using the FEM bias and uncertainty based on frequency response test.

3. Verify that the fatigue analysis results for the as-built steam dryer maximum calculated alternating stress intensity will meet or exceed an MASR of 2.0 to the allowable alternating stress intensity of 93.7 MPa (13,600 psi).

4. Verify that the primary stress results for the as-built steam dryer meet the acceptance criteria for the normal, upset, emergency, and faulted load combinations.

5. Identify on-dryer instrumentation sensor specifications, sensor locations, correlations between sensors and peak stress locations, and bias and uncertainties of sensors and data acquisition system.

6. Define steam dryer instrument acceptance limits that maintain peak alternating stress amplitude less than 93.7 MPa (13,600 psi) for the steam dryer. Limit curves for power ascension will be based on the worst-case of the design-basis calculations that use the end-to-end GGNS bias and uncertainty, and those from the as-built steam dryer calculation that use the combined FE structural and PBLE01 biases and uncertainties.

III. Power Ascension Monitoring and Inspection

1. Develop a steam dryer power ascension monitoring and inspection program that reflects industry experience with the performance of steam dryer power ascension testing.

2. Instrument and monitor the steam dryer during power ascension to measure steam dryer pressure loads as well as steam dryer strain and acceleration to assure that adequate steam dryer fatigue margin is maintained.

3. At approximately 75 percent power during the initial power ascension, perform the following:

   a. Record pressures, strains, and accelerations from the on-dryer mounted instrumentation. Evaluate the data and compare the measured dryer strains and accelerations to acceptance limits.

   b. Develop a PBLE01-based ESBWR FIV load definition based on selected on-dryer instruments. Using appropriate methods, such as F-factor and RMS, and the above PBLE01-based ESBWR load definition, predict the steam dryer strain and acceleration response at this condition.

   c. Compare the predicted steam dryer strain and acceleration against the measured data and determine frequency dependent end-to-end bias and uncertainty values. Adjust the predicted strain and acceleration responses using the frequency-dependent end-to-end bias errors and uncertainty values. If any of the measured sensor data exceed the adjusted predictions, then modify the bias errors and uncertainty values and limit curves and ensure measured sensor responses do not exceed the adjusted predictions, or quantitatively evaluate the impact on fatigue life.

   d. Define the steam dryer peak stress projections based on the revised results from step b with modified end-to-end bias and uncertainties from step c. Compute the steam dryer maximum stress and minimum stress ratio from the predictive analysis using up to a [[ ]] of load applications. Prepare cumulative stress plots for at least the [[ ]] most highly stressed locations on both the upper and lower dryer with the dominant stress component at each location used for the plots. The peak stress amplitude adjusted for the bias and uncertainty is maintained less than 93.7 MPa (13,600 psi).

e. Update limit curves based on the results from step d. Level 1 and Level 2 limit curves will be generated for all functioning strain gage and accelerometer locations on the steam dryer and will include bias errors and uncertainties as described in Section 7 of NEDE-33313P.

f. Trend the recorded data and project the stress, strain, and accelerometer sensor responses for the next assessment point and full power to demonstrate margin for continued power ascension.

4. Continue power ascension in no more than 5 percent power increments up to 100 percent power in accordance with the power ascension monitoring plan. Appropriate methods, such as F-factor and RMS, will be used to monitor dryer stresses at intermediate power levels during power ascension.

5. During power ascension, if flow-induced resonances are identified and the strains or vibrations increase above the pre-determined criteria, power ascension is stopped. The acceptability of the steam dryer for continued operation is evaluated [[
.]] The limit curves are then redefined based on the on-dryer data. The limit curve factor is revised [[
.]]

6. When full power is obtained, the steam dryer peak stress projections are determined based on the full power test data and adjusted for end-to-end benchmark bias and uncertainties and instrument uncertainties, to demonstrate that the steam dryer will maintain its structural integrity over its design life considering variations in plant parameters (such as reactor pressure and core flow rate).

7. Conduct steam dryer inspections during the first two refueling outages and develop a long-term steam dryer inspection program based on the results of those steam dryer inspections.

IV. Subsequent Plants

Power ascension testing of subsequent plants will follow the same FIV monitoring process using on-dryer instruments incorporating lessons learned from power ascension of previous ESBWR plants as applicable. The power ascension acceptance limits for subsequent plants are based on assuring that the stresses remain less than 93.7 MPa (13,600 psi). Limits are based on frequency domain curves developed from the initial unit test data factored by a limit curve factor. The limit curve factor is determined [[ ]]

The NRC staff finds that NEDE-33313P clarifies the ESBWR steam dryer evaluation methodology consistent with the RAI responses and steam dryer evaluation processes accepted in recent power uprate applications for operating BWR nuclear power plants. Therefore, RAI 3.9-299 is resolved.

3.9.5.3.3.2 Steam Dryer Acoustic Loading Methodology

The ESBWR steam dryer acoustic loading methodology PBLE01, described in NEDE-33408P, is an inverse source identification approach which measures the fluctuating pressures on a steam dryer and determines the equivalent excitation sources at the MSL openings. This approach uses Matlab scripts (which is a high-level language and interactive environment for numerical computation, visualization, and programming), and a three-dimensional acoustic finite element code (SYSNOISE) to map the fluctuating pressures in the steam dome volume. The excitation sources are positioned at the inlet nozzles of the four MSLs in the model for the particular dryer. The equivalent nozzle sources, which are fluctuating velocities oriented normally to the RPV walls, are applied to acoustic FEMs of the RPV steam volume to simulate the dynamic differential pressure loading over the entire dryer surface. The loading is computed for frequencies up to 250 Hz because industry data have indicated that significant peaks at higher frequencies are not present in the measured acoustic pressure loads on steam dryers in operating nuclear power plants. The PBLE01 loading includes any source which induces a fluctuating pressure load on instrumented dryer surfaces.

The acoustic finite element code, SYSNOISE, is well-established and has been used widely for more than 15 years. SYSNOISE uses the acoustic Helmholtz equation to compute the acoustic field over selected frequencies. For SYSNOISE results to be valid, the element sizes for an RPV acoustic finite element mesh must satisfy the common discretization error criterion of at least six elements per wavelength for frequencies up to 250 Hz. The steam material properties (speed of sound and density) are based on standard steam tables. The ESBWR design certification applicant also describes and uses standard procedures for estimating steam damping.

To determine dryer loading, individual frequency response functions (FRFs) between each MSL nozzle fluctuating velocity and the on-dryer pressure measurement locations are computed. The acoustic model uses different wet steam properties upstream (inside the dryer) and downstream (in the RPV dome and MSL inlets). Boundary conditions at the top of the steam separator tubes are assumed to be anechoic, representing the complete absorption of acoustic waves, as is typical for this type of analysis. An approximate impedance boundary condition, based on the speed of sound and density of bubbly water, is applied to the steam-water interface. These approximations are reasonable, and any inaccuracies are accounted for in the bias errors and uncertainties derived from benchmark comparisons.

A licensee will use pressure measurements at selected locations on the steam dryer to calculate the nozzle sources, or nozzle velocities, with the aid of the inverted matrices of FRFs. Measurements on the GGNS Replacement Steam Dryer at EPU conditions are used to demonstrate the procedure. Singularity factors (SFs) are computed to verify the stability of the FRF matrix inversion, and the ESBWR design certification applicant shows that singularities are eliminated by using a larger number of pressure measurement points on the dryer to calculate the source terms, overdetermining the system of equations. Based on the computed SF, the ESBWR design certification applicant reports that the use of at least eight pressure measurement locations on the dryer outer hoods to define the source terms is adequate.

The PBLE01 method is demonstrated using the GGNS instrumented RSD. All working pressure sensors on the RSD (more than the minimum of eight called for by the method) were used to estimate MSL nozzle velocities and subsequent on-dryer surface (on the outer surface) and differential (delta between internal and external surfaces) pressure loading. Simulated and measured dryer surface pressures are compared at EPU, as well as at Originally Licensed Thermal

Power (OLTP) conditions. Frequency dependent pressure loading bias errors and uncertainties are computed over the upper and lower dryer surfaces.

The differential pressures over the dryer surface are then applied to a structural FEM of the GGNS RSD. The structural modeling and analysis procedures are defined separately in NEDE-33313P. The resulting structural vibration simulations are used to estimate fluctuating strains at the instrumented locations on the actual RSD. The differences between the simulations and measurements are used to compute end-to-end bias errors and uncertainties over the upper and lower dryer. These end-to-end quantities encompass nearly all of the overall dryer stress analysis procedure, with the exception of structural mesh convergence studies to ensure stress concentration regions are resolved, and any weld factors that must be applied. Using end-to-end bias errors and uncertainties is preferred to using individual analysis component errors summed using square root sum of squares (SRSS) methods, because any excitation mechanisms not captured in the PBLE01 procedure, such as buffeting of the submerged portion of the dryer skirt by boiling-water pulsations, may be properly accounted for with strain and acceleration-based end-to-end bias errors.

The computed bias errors are the statistical means of the errors computed for the sensor groups spread over the upper and lower dryers. The uncertainties are one standard deviation of the individual errors. This is atypical, as most uncertainty standards and procedures use two standard deviations, which would capture 95 percent of the variability over all of the sensors. However, the ESBWR design certification applicant demonstrates the suitability of their uncertainties against GGNS data, as described below.

All of the benchmarking calculations are performed over short time subintervals extracted from a much longer time record acquired in the GGNS plant. Two time subintervals are chosen to maximize dryer loading for low frequency (LF) and high frequency (HF) signals. Strains and accelerations at the other time intervals are evaluated using approximate scaling methods (called RMS and F-factor by the applicant), and additional 'time interval bias' is computed and included in the upper and lower dryer frequency dependent bias errors.

The final end-to-end bias errors and uncertainties are tabulated in Appendix F of NEDE-33408P, and will be applied in the stress calculations for the detailed design of the ESBWR steam dryer. Although some of the bias errors from the GGNS benchmarking study are positive (indicating overly conservative simulations), the ESBWR steam dryer methodology will not take credit for them in the detailed design process, setting all positive bias errors to zero. This will ensure that any frequency shifting between the GGNS and ESBWR dryer forcing functions or structural response will not lead to nonconservative bias error adjustments.

The ESBWR design certification applicant demonstrates the application of the final bias errors and uncertainties for each GGNS RSD strain gage and accelerometer location at EPU conditions. Plots of adjusted strain and acceleration time histories and spectra are shown in Appendix G of NEDE-33408P. Each simulated spectrum reflects the upper envelope of calculations made by stretching and compressing the dryer loading time histories in increments of 2.5 percent, spanning a [[ ]] uncertainty range of the simulated dryer resonance frequencies. These worst-case calculations are compared to the measurements, and are mostly, but not always, bounding. A few sensors indicated peak responses that were up to 30 percent above the simulations. The NRC staff considered these GGNS underpredictions in evaluating the ESBWR steam dryer design process for bias and uncertainty assumptions.

In that the GGNS dryer data has significant peaks near flow-induced standpipe resonances at high frequencies (which will not occur in the ESBWR plants per ITAAC requirements), the ESBWR design certification applicant provides additional plots showing only the LF data, where the standpipe resonances are not present. These plots are more relevant to ESBWR, but also show a few sensors where simulations do not bound measurements (up to 20 percent nonconservatism).

The lack of conservatism in some of the GGNS dryer simulations is due to the uncertainties being based on a single standard deviation, which by definition is only expected to encompass 75 percent of the sensor data. However, the demonstration for the GGNS RSD does not mean that the procedure will not produce bounding simulations for the ESBWR steam dryer. The data available for GGNS RSD benchmarking was limited by the loss of two critical on-dryer pressure sensors near one of the MSLs. Partial GGNS benchmarking at OLTP (with all on-dryer sensors operational) of the surface pressures, shown in Section 3.2 and Appendix B of NEDE-33408P, shows improved agreement compared to EPU comparisons. Also, the as-designed and as-built ESBWR dryer stresses must satisfy a more restrictive MASR of 2.0. This higher factor of safety, which provides 100 percent margin (overprediction) that is much greater than the 20 to 30 percent underpredictions shown in Appendix G of NEDE-33408P, provides reasonable assurance that the applicant’s procedure will lead to bounding alternating stress calculations.

As discussed below in this supplemental FSER regarding ITAAC, an ESBWR licensee must confirm, through measurements made on an instrumented ESBWR dryer, that simulations bound measured data. At that point, the licensee need only maintain an MASR of 1.0, because actual prototype dryer data will be available, and the bias errors and uncertainties will be based on actual ESBWR, not GGNS, benchmarking. However, in the event that the as-built ESBWR steam dryer simulations do not bound measurements for all on-dryer strain gages and accelerometers, the COL licensee may either (a) adjust its ESBWR bias errors and/or uncertainties to ensure all measurements are indeed bounded, or (b) quantitatively assess the impact of the underpredictions on dryer fatigue life.

3.9.5.3.3.3 Steam Dryer Acoustic Loading Effects from Valve Standpipes and Main Steam Piping

In some operating BWR plants, low-order acoustic resonances within safety relief valves (SRV) and safety valve (SV) standpipes locked in to flow-induced shear layer instability modes over the standpipe openings, generating extremely powerful acoustic pulsations within the MSLs. These pulsations subsequently propagated along the MSL and into the RPV, impacting the steam dryers, and leading to fatigue cracking and the eventual generation of large loose metal parts within the RPV. The ESBWR design will provide assurance that such flow-induced resonance behavior will not occur in the MSLs of ESBWR plants at normal operating conditions because MSL and standpipe design will preclude such acoustic resonances during normal operation, as described below.

In Section 3.9.5.3, “Loading Conditions,” of the DCD Tier 2, the applicant states that:

> The safety relief valves (SRVs) and safety valves (SVs) standpipes and main steam branch lines in the ESBWR are specifically designed to preclude first and second shear layer wave acoustic resonance conditions from occurring and to avoid pressure loads on the steam dryer at plant normal operating conditions.

The ESBWR design certification applicant provides additional details in the DCD and NEDE-33313P on the design procedure to avoid flow-excited acoustic resonances, and how the final design will be confirmed through ITAAC 36 in Table 2.1.2-3 of DCD Tier 1.

The lock-in of shear layer instability modes over branch line openings with acoustic resonances within branch line cavities is well understood.[2] Well-established procedures described in these and other references will be used to ensure that flow speeds and geometric dimensions are specified so that shear instability and acoustic resonance frequencies do not coincide during normal operating conditions.

The ESBWR design certification applicant's assessments of the ESBWR piping and valve design show that the onset of the primary shear resonance, and the secondary (much weaker) shear resonance, would occur at steam flow rates higher than normal ESBWR operating conditions. Also, the ESBWR design certification applicant states that the ESBWR MSL design avoids any low-frequency flow-excited resonances. Assessments will be confirmed as part of the detailed design completion.

In that the ESBWR design certification applicant's approach for avoiding flow-excited acoustic resonances in the MSLs is based on well-established procedures and will be confirmed during the ITAAC process, the NRC staff finds the approach to be acceptable.

3.9.5.3.3.4 ESBWR Acoustic Steam Dryer Load Definition and Minimization

The ESBWR design certification applicant describes the procedure for generating an ESBWR steam dryer differential pressure load definition in NEDE-33312P. Since there is no analytic means of accurately defining the dryer loading, it will be based on measurements taken from operating BWR plants. The ESBWR design certification applicant also describes design procedures for minimizing the dryer loading based on flow and acoustic computational modeling.

The design approach for the ESBWR steam dryer consists of the following basic steps:

1) Maintain key design aspects of the ABWR steam dryers, which show satisfactory performance.

2) Evaluate the acoustic response of the ESBWR reactor dome to avoid geometries that can result in large acoustic loads.

3) Analyze the ESBWR dryer design under the design loads that are determined from in plant measurements of existing instrumented dryers, including an ABWR dryer and BWR/4 RSDs.

4) Instrument (using pressure transducers, accelerometers, and strain gages) and monitor the ESBWR dryer during power ascension to ensure that adequate dryer fatigue margin is maintained.

[2] See, e.g., Hambric, Mulcahy, Shah, Scarbrough, Wu, "Flow-Induced Vibration Effects on Nuclear Power Plant Components Due to Main Steam Line Valve Singing," NUREG/CP-0152, Volume 6 (2006), Ninth ASME/NRC Symposium on Pumps, Valves, and In-Service Testing; and Ziada and Shine, "Strouhal Numbers of Flow-Excited Acoustic Resonance of Closed Side Branches," *Journal of Fluids and Structures*, Vol. 13, No.1, pp. 127-142, 1999.

The ESBWR steam dryer has geometry similar to that of ABWR dryers, which have not experienced flow- or acoustically-induced vibration issues. ABWR in-plant measurements, along with measurements from other operating BWR plants, will be used to define conservative fluctuating pressure loads in finalizing the ESBWR dryer design.

The ESBWR and ABWR RPV diameter and MSL configurations are expected to be identical. The ESBWR and ABWR dryers will also have similar vane height, skirt length, and water level. The ESBWR dryer diameter is expected to be larger, with longer vane banks than in the ABWR. The setback between the upper hood region and the RPV walls (and therefore from the MSL inlets) is 33” for both the ABWR and ESBWR designs. The ESBWR steam flow rate is expected to be 15 percent higher than that of the ABWR, leading to increased dryer loading. The vessel head shapes are different, with a hemispherical head for the ABWR and a torispherical head for the ESBWR.

In completing the detailed design of the ESBWR steam dryer, computational fluid dynamics (CFD) Reynolds-averaged Navier-Stokes (RANS) analyses will be conducted of the ESBWR dryer and dome steam flow to assess differences between the ABWR and ESBWR RPV dome heads. Although the CFD RANS results will be considered only qualitatively, they will be useful for finalizing design details for the ESBWR dryer to minimize local flow-induced forces, particularly near the MSL nozzle inlets.

The ESBWR dryer and RPV region acoustic FEM will be constructed and used to address the acoustic effects of ESBWR geometry on the dryer loading. The procedures described in NEDE-33408P will be used for the acoustic modeling. The dryer loading is defined by specifying MSL inlet source levels, which will be used to finalize the dryer design to minimize loading and dryer response. Surface loading will be determined at dryer locations that correspond to surface pressure measurements made on ABWR dryers. The ABWR dryer surface pressure measurements, increased in amplitude to account for the anticipated 15 percent higher flow rate in the ESBWR, will be bounded by the final design ESBWR dryer loads.

The acoustic ESBWR dryer and RPV model will include MSLs, so as to finalize the ESBWR MSL and valve designs to minimize flow-induced acoustic pulsations and resulting dryer loads. The final ESBWR system design will ensure that no flow-induced resonances of valve standpipes, such as those that occurred in the QC plants, will appear in the ESBWR plant near normal operating conditions. As indicated in Table 3L-1 of the ESBWR DCD Tier 2, the average main steamline flow velocity in the ESBWR will be 47 meters per second (m/s) while the flow velocity in the ABWR in Japan is 46 m/s and in the QC plants is 62 m/s.

In-plant data from two other instrumented dryers will be used to finalize the ESBWR dryer design loads. The two datasets span the worst-case LF and HF loads observed in operating BWR plants. The data are compared with ABWR pressure measurements in NEDE-33312P, but only to confirm that the ABWR data, after being increased in amplitude to reflect the anticipated 15 percent higher ESBWR steam flow rate, are bounded. The steam flow rates in the two plants used to define ESBWR loads are greater than or equal to that projected for the ESBWR plant. In the example at a single location on the skirt provided by the ESBWR design certification applicant in Figure 4.1-1 of NEDE-33312P, the ESBWR projected loads are about 50 percent higher than loads measured in the ABWR that were also scaled upward to account for the higher ESBWR steam flow rate.

The proposed procedure for defining the ESBWR steam dryer loads is acceptable and conservative, because it is based on dryer measurements at worst-case loading conditions observed in two operating plants, and will also be shown to bound ABWR dryer surface pressure measurements (scaled up to account for the 15 percent increased flow in ESBWR). However, the actual loads and subsequent ESBWR dryer response have not yet been determined. The actual loads will be assessed during the startup monitoring process as discussed later in this FSER supplement.

The NRC staff finds that the ESBWR steam dryer acoustic load definition is acceptable in that it is a bounding approach, will use end-to-end B/U from an instrumented similar dryer, and will be confirmed with on-dryer measurements, as set forth in detail above.

3.9.5.3.3.5 ESBWR Steam Dryer Structural Analysis

3.9.5.3.3.5.1 Summary of ESBWR Steam Dryer Structural Analysis Process

The ESBWR design certification applicant describes its design analysis approach and design criteria for its steam dryer in NEDE-33313P. The main objectives of the analysis are (1) to evaluate the susceptibility of the dryer to high-cycle fatigue cracking caused by FIV during normal operation, and (2) to predict the stresses that would result from the specified ASME BPV Code load combinations.

The ESBWR steam dryer design is similar to the design of the dryers in operating BWRs in the United States and operating ABWRs in another country. The dryer includes steam drying vanes and perforated plates arranged in six parallel rows called dryer banks. The dryer consists of an upper and a lower support ring. The dryer banks are above the upper support ring, and the skirt extends from below the upper support ring to the lower support ring. The dryer is supported at the upper support ring by the RPV support brackets. The dryer is not directly connected to the chimney head or the steam separator assembly. The skirt projects downward to form a water seal around the array of steam separators. The dryer will be instrumented with pressure transducers, strain gages, and accelerometers that will facilitate estimating (1) the FIV loads on the dryer, and (2) end-to-end B/U for determining the dryer stresses.

The applicant has developed weld fatigue factors for estimating maximum peak stress intensities at the dryer weld locations and included these factors in NEDE-33313P. The applicant has identified the minimum size of the double-sided fillet welds used in fabricating the dryer.

The applicant specifies the use of the ANSYS finite element analysis (FEA) computer code, [[ ]] The model will be subject to the FIV loading time history and any loading scale factors as developed in NEDE-33312P. A [[ ]] will be performed to account for any errors in the calculation of the dryer's natural frequencies (assumed to be less than 10 percent). A dynamic analysis will be conducted with the ANSYS computer code direct integration or harmonic analysis with [[ ]]. The resulting stresses along with weld fatigue factors and appropriate B/U will be used to estimate the maximum peak stress intensity. If the estimated maximum stress intensity exceeds the fatigue limit, [[ ]]

The applicant provided a list of the load combinations for ASME BPV Code, Service Level A, B, C, and D conditions. The ASME load combination stress analysis will determine the maximum stress intensities, which will be compared to the stress limits in the ASME BPV Code, Section III, Subsection NG, with specified margin.

3.9.5.3.3.5.2 Staff Evaluation of ESBWR Steam Dryer Structural Evaluation

3.9.5.3.3.5.2.1 Comparison of ESBWR and ABWR Steam Dryers

The comparison of key parameters of the ABWR and ESBWR dryer designs is provided in Table 3L-1 of Appendix 3L to the DCD Tier 2. Table 3L-1 shows that the plate thicknesses for the ESBWR dryer hoods and skirt are increased over those used in the ABWR design. Although the steam flow rate is 15 percent higher for the ESBWR than for the ABWR, because the plate thickness of several ESBWR components is larger than that in the ABWR, the ESBWR and ABWR stresses should be comparable.

One important difference between the ABWR and the ESBWR is the presence of SRV resonance during normal operation. In the ABWR, a resonance onset for the second shear layer mode has been observed at about 200 Hz (Figure 3-16, NEDC-33601P, Rev. 0). Similar onset has been observed in a BWR/6 nuclear power plant. As discussed in Section 3.9.5.3.3.3 of this supplemental FSER, the ESBWR MSLs and SRVs are designed such that a resonant onset for either the first or second shear layer mode does not take place during normal operation.

3.9.5.3.3.5.2.2 Steam Dryer Fabrication

The design for a steam dryer that is expected to be subject to FIV is described in DCD Tier 2, Section 3L.2.3, "Design and Materials Evaluation." The DCD discusses streamlining structural discontinuities within the steam dryer, thus reducing stress risers that contribute to fatigue failure. The ESBWR steam dryer design [[ ]]. As stated in NEDE-33313P, the design of the ESBWR steam dryer is similar to that of the ABWR steam dryer with additional fabrication improvements developed for RSDs in operating BWRs. The ABWR steam dryers had already incorporated significant improvements over the BWR/6 design in [[ ]].

The ESBWR design may use fillet welds, but these welds are at locations [[ ]]; only double-sided fillet welds are used in the ESBWR design, with one exception noted below. In addition, to reduce the susceptibility to fatigue cracking, the ESBWR steam dryer design eliminates some welds in previously identified high stress locations by using [[ ]] in the design rather than welding separate plates together. In response to RAI 3.9-214, S01 the applicant stated that all the welds at high-stress locations will receive solution heat treatment provided the weldments do not present size, material, or distortion issues.

For the ESBWR steam dryer fabrication, the root pass and the final pass of any multiple-pass weld are examined for weld quality using [[ ]]. The in-between passes are visually examined for presence of any crack-like defect. The thickness of each weld pass would be smaller than the critical flaw size, so that no undetected flaw larger than the critical flaw size would be present.

The staff finds the information provided above by the applicant regarding the fabrication and inspection of the steam dryer welds to be acceptable for the following reasons: The use of [[ ]] will increase the dryer's resistance to high-cycle fatigue. In addition, [[ ]] of root and final passes and visual examination of in-between passes of the weld would reduce the possibility of fabrication flaws in the welds, and [[ ]] would reduce residual stresses in those welds; these actions would reduce the susceptibility to high-cycle fatigue. The staff finds that the applicant's response to RAI 3.9-214 S01 has provided additional clarifying information regarding the fabrication of the ESBWR steam dryer specified in DCD Tier 2, Section 3L.2.3, and Section 4.0, "Design Criteria," in NEDE-33313P. Therefore, RAI 3.9-214 S01 is resolved.

In response to RAI 3.9-214, the applicant stated that the ABWR steam dryers had already incorporated significant improvements over the BWR/6 design in [[ ]]. In RAI 3.9-214 S02(a), the staff requested that the applicant provide examples of these improvements. In response to RAI 3.9-214 S02, the applicant stated that some of these improvements include [[

]]. The staff finds that the applicant's responses to RAI 3.9-214 and RAI 3.9-214 S02(a) have provided additional clarifying information regarding the improvements to the ESBWR steam dryer design specified in DCD Tier 2, Section 3L.2.3, and Section 4.0 in NEDE-33313P that will reduce the stresses at the welds and improve the fatigue resistance of the ESBWR dryer. Therefore, RAI 3.9-214 and RAI 3.9-214 S02(a) are resolved.

In response to RAI 3.9-214 S02 (b), the applicant stated that if the stress at a given weld is higher than the acceptance criteria after applying the fatigue strength reduction factors (FSRFs), or if a maximum stress location falls on a weld, then that weld is considered a high-stress location and will be redesigned as discussed in its response to RAI 3.9-214, S01. The staff finds this response acceptable because the proposed approach for redesigning the high-stress weld would reduce the possibility of high-cycle fatigue cracking at the welds. Therefore, RAI 3.9-214 S02 (b) is closed.

3.9.5.3.3.5.2.3 Quality Factors and Fatigue Strength Reduction Factors for Steam Dryer Welds

The applicant specifies the use of full penetration welds and fillet welds for the ESBWR steam dryer fabrication; partial penetration girth welds are not used. Except for the attachment of lugs, fillet welds are double-sided welds. The steam dryer welds are either primary load-bearing (SD1) or secondary non-load-bearing (SD2).

All SD1 welds are full penetration groove welds. For these welds, the applicant specifies the use of ASME BPV Code, Section III, Subsection NG for weld classification, fatigue factor, quality factor, and PT inspection of the root and final weld passes.

For SD2 welds, the applicant specifies the use of Subsection NG for weld classification, but defines alternate weld quality and fatigue factors. A weld quality factor of 1.0 is applied for all SD2 welds. This is less conservative than the values listed in Subsection NG for [[ ]] of the root and final weld passes. The weld quality factor reduces the maximum allowable primary stresses, which affect the static stress analysis, but not the fatigue evaluation. Given that fatigue is the primary mechanism of concern for FIV-related damage to the steam dryer, this weld quality factor is therefore acceptable. The applicant specifies the performance of a [[ ]] of the root pass of the weld. Field experience has shown that [[ ]] of the root pass might not detect a rejectable defect at the root. As a result, Section 4.3 in NEDE-33313P specifies that representative weld samples using the same joint design and material types as specified for the ESBWR steam dryer are destructively tested. Further, metallurgical evaluation demonstrating an acceptable weld root is called for prior to weld procedure approval. These tests are relied upon to demonstrate that no defects are present at the root of production welds.

The applicant's FSRF for SD2 full penetration groove welds is specified to be 1.4 vs. 1.0 in Subsection NG. The staff accepts the use of 1.4 because it is more conservative.

For fillet welds, one of the following three methods may be used with an FEA to obtain the peak stress at a fillet weld:

1) For a shell element model, an [[ ]] as specified by the ASME BPV Code.

2) For a shell element model, an [[ ]].

3) For a solid element submodel, three FSRFs are applied: [[ ]

For all three methods, the FSRFs are applied to the converged stresses, with convergence bias applied where appropriate, as described in Section 3.9.5.3.3.5.2.4 below.

In response to RAI 3.9-285 and RAI 3.9-286 and their supplements, for evaluation of Methods 1 and 2 listed above, the applicant has considered a simple test problem, a thin vertical plate welded to a thick horizontal plate using a double-sided fillet weld. [[

]]

NEDE-33313P provides that the section properties of the double-sided fillet welds in the ESBWR steam dryer are as good as or better than the section properties of the thinner plate (vertical plate) that is being welded. Specifically, NEDE-33313P stipulates that the ESBWR steam dryer design will maintain [[.]] In other words, the weld throat is at least equal to half the thickness of the vertical plate, so for the double-sided fillet weld, the total throat size is at least equal to the thickness of the thinner plate. double-sided fillet weld in the ESBWR dryer will satisfy this size criterion. For such weld design, no thickness correction (discussed in Section 4.2 of NEDE-33313P) is needed for estimating the peak stress at the fillet weld.

In response to RAI 3.9-285 and its supplement, the applicant estimated peak stress at the weld in the test problem using [[

]] NEDE-33313P, Section 4.2 also specifies a limitation of [[ ]] in the ESBWR steam dryer to ensure that Method 2 provides a conservative result.

In summary, Methods 1 and 2, which use FEA, bound the results obtained by the traditional method (Method 2 with restriction on fillet weld size noted above), and, therefore, the use of an FSRF of [[ ]]. However, only FEA (and not the traditional method) are used for fatigue assessment of the dryer because of its complex design. Method 2 will be demonstrated to bound the Method 1 results for the limiting fillet welds in the dryer; otherwise, one of the other methods will be used in analyzing those welds.

For Method 3, in response to RAI 3.9-286 and its supplement, the applicant described that the numerical results for the test problem indicate that either the root location or the toe location produce the highest peak stress intensity, depending on the specific geometry of the plate configuration. The root location results are consistent with Methods 1 and 2 and bound the traditional method results for the test problem where the submodel stress on the weld throat has converged. The FSRFs for Method 3 will be demonstrated to bound Method 1 and the traditional method before using them to estimate peak stresses for the design-basis fatigue evaluation. The staff finds the use of Method 3 in estimating peak stresses at the fillet welds in the ESBWR dryer to be acceptable based on the results of the test problem and the provisions specified in NEDE-33313P.

The applicant specifies that the calculated peak stress intensity from the FEA will be treated as the alternating stress intensity, and the FSRFs will be applied to the peak stress intensity. The staff finds that this approach is acceptable for assessing fatigue damage because [[

]], which is conservative.

3.9.5.3.3.5.2.4 Finite Element Modeling

A three-dimensional FEM of the ESBWR steam dryer based on nominal dimensions will be created using the ANSYS finite element code. The ESBWR steam dryer modeling approach is consistent with those used for steam dryers at operating nuclear power plants, including the GGNS steam dryer that is used as the benchmarking basis for the end-to-end ESBWR steam dryer analysis process. This structural model will predominately contain [[ ]], but also include fluid, beam, solid, and mass-only elements where appropriate. The boundary conditions for the structural model will be at nodes at the support bracket locations. When shell elements are physically attached to solid elements, there is a mismatch between the nodal degrees of freedom of the element types. To avoid a hinge-like connection due to the lack of rotational degrees of freedom in the solid elements, the shell elements will be connected to the solid elements using a solid-shell interface model discussed later in this section.

This section discusses the following three FEM topics: (1) mesh refinement for achieving stress convergence, (2) use of submodels, and (3) solid-shell interface model.

<u>Mesh Refinement</u>: In NEDE-33313P, the applicant presents the use of mesh convergence studies to define the element sizes based on a relative stress change of [[ ]] from a previous iteration. Regions where this criterion is not met are modeled coarsely in the overall analysis, but then finer shell or solid submodels are used to obtain converged stresses based on displacement boundary conditions taken from the overall analysis.

However, the [[ ]] does not represent the true convergence error because the converged results have not been determined. As a result, further mesh refinement might be necessary. For example, the results for the refined meshes could be extrapolated to zero mesh size. The corresponding results would then represent the converged results and may be used to determine the convergence error. To address this issue, the applicant revised the mesh convergence process for defining the stress convergence error for the ESBWR steam dryer. First, the global FEM for the ESBWR steam dryer will be built using mesh sizes similar to those used for the steam dryers in operating BWRs evaluated for EPU operation. Mesh convergence studies were performed at high stress locations using [[ ]]. These studies revealed the following four types of convergence:

1) As the mesh is refined, the stress results changed less than 1 percent and a monotonic decreasing trend is evident, implying that the stresses in the global analysis have been already converged. For this case, the stress convergence error is zero.

2) The stresses are converged after successive mesh refinement. For this case, the stress convergence bias factor is equal to the converged stress result divided by the corresponding stress result from the global analysis.

3) The stresses are not converged as the mesh is successively refined. For this case, the stresses are extrapolated to zero mesh size and the corresponding stress results are considered the converged stresses and a bias factor is determined as before.

4) The mesh convergence process may not adequately resolve the stress intensity for areas having stress singularities or areas having tight corners; use of a submodel is considered and is discussed later in this section.

The staff finds the mesh refinement process to be acceptable because it reasonably determines the stress convergence error. In particular, the staff notes that the stress convergence error is location dependent. The error will be small for the locations having small strain gradients and large for the locations having large strain gradients. In that the instruments are installed at locations having small strain gradients, the corresponding stress convergence bias errors would be small. The high stress locations would generally have large strain gradients and, therefore, the corresponding stress convergence bias errors would be large. So, the end-to-end bias errors, which are determined using the measured instrument results, will not generally include the large stress convergence bias errors present at the high stress locations. Therefore, in addition to the end-to-end bias error, the high stresses will be adjusted for the corresponding stress convergence bias errors as indicated in Section 6.0 of NEDE-33313P.

Submodeling: The applicant specifies a submodeling procedure to further analyze the portions of the dryer where consecutive mesh refinements do not provide stress convergence; such portions include locations with high-stress intensity, stress singularity or geometric features such as tight corners. For an overall ESBWR dryer design, the applicant has demonstrated the submodeling procedure at two fillet weld locations. The submodeling procedure [[

]]. The applicant [[
]]. The applicant then refined the mesh of the submodel and [[
]]. The applicant continued to refine the mesh until one of the three types of stress convergence (described earlier in this section) was achieved, and estimated the corresponding stress convergence bias error and adjusted the stresses accordingly. The stress report lists the adjusted stress results. The staff finds the response acceptable because the applicant has refined the mesh of the submodel until convergence has been achieved.

During its review, the NRC staff requested that the applicant explain how the size of the submodel will be determined to ensure that the displacements and stresses at the cut boundaries remain unchanged as the finite element mesh is refined and as stress and strain are changed at the location of interest. The applicant indicated that [[

]]. The staff finds this response acceptable because the applicant's approach to determine the size of the submodel mesh ensures that the local changes caused by the refined mesh would not affect the stresses and displacements at the cut boundaries. Therefore, this concern is resolved.

Solid-Shell Interface Models: At a solid-shell interface, a thick component such as the upper support ring is welded to a relatively thin component such as the steam dryer skirt. The thick component is modeled with solid elements and the thin component with shell elements. Because the shell element nodes have 6 degrees of freedom and the solid element nodes have 3 degrees of freedom, a condition requiring kinematic compatibility between the shell and solid elements at the interface needs to be imposed. The applicant imposes such a condition by modeling these interfaces with an overlay or an embedded element, which is a shell element, laid over or embedded into the solid element model at the interface. The planar dimensions of the overlay element are selected to match the shell and solid elements' dimensions at the interface. The thickness of the overlay element is selected such that it gives the correct analytical results for tip deflection of a cantilever beam represented as a thin component at the interface and modeled using shell elements; the selected thickness of the overlay element is equal to the thickness of the beam. The staff has also learned from its experience with operating BWR EPU applications that the use of an overlay element in the analysis of steam dryers provides conservative results compared to the results from the submodel analysis of the solid-shell interface using only solid elements. The applicant also demonstrated that the shell overlay element method does not artificially reduce the local stress because of the added overlay shell elements. Therefore, the staff finds the use of overlay elements is acceptable because it provides conservative results.

3.9.5.3.3.5.2.5 Finite Element Model Bias Errors and Uncertainties

As discussed in NEDE-33313P, the applicant's design process will shift the frequency of the ESBWR steam dryer loading by [[ ]] to account for uncertainty and bias in the FEM resonance frequencies. The staff identified two concerns with this process: (1) [[ ]] might not be sufficient if the differences between the measured and predicted natural frequencies of the dryer are greater than [[ ]], and (2) such [[ ]] does not account for errors in the mean and peak frequency response amplitudes resulting from the uncertainty or bias in plate dimensions, boundary conditions (joints between plates and other members), pre-stresses within members, and friction between internal vanes and other components. To address these two concerns, the applicant specifies the performance of dynamic testing (also called a hammer test or shaker test) of the as-built ESBWR steam dryer. The results of the testing are used to compare the predicted and measured frequencies and frequency response functions. If significant discrepancies of more than [[ ]] occur in the frequency comparison, then those differences will be evaluated and if necessary addressed with adjustments made to the FEM or frequency shifting in the finite element stress analysis to ensure appropriate coupling between peak excitation and peak response is captured.

The staff finds the applicant's response acceptable because the benchmarking of the FEM against instrumented dryers would properly account for errors in the mean and peak frequency response amplitudes and the frequency differences greater than [[ ]].

The applicant indicates that during power ascension peak dryer stresses might be determined by using either an F-factor approach, similar to that used during the Vermont Yankee Nuclear Power Station (VY) EPU power ascension, or an RMS method. The applicant demonstrated previously that for the Susquehanna Steam Electric Station (SSES) RSD, the F-factor and RMS methods provide good estimates of the peak dryer stresses with the RMS method producing the most conservative (highest) stresses.

The time segment selection bias is based on a comparison of the peak stress computed for the chosen time segment with peak stresses from all other segments acquired during dryer in-plant measurements. The applicant estimates the peak stresses at the other time segments using the F-factor or RMS approaches. The applicant also describes its rationale for strain gage location and measurement uncertainty based on installation uncertainty and calibration uncertainties.

Based on its review of steam dryers at nuclear power plants receiving power uprate license amendments, the NRC staff requested that the ESBWR design certification applicant specify [[
]] bias errors and uncertainties for the ESBWR steam dryer. The applicant indicated that [[

]].

The process to address FEM bias errors and uncertainties has been specified in NEDE-33313P. The staff finds the process acceptable because [[ ]] are documented in NEDE-33313P.

3.9.5.3.3.5.2.6 Dynamic Stress Analysis and Fatigue Analysis Considerations

The applicant specified the use of either or both [[
]] to assess dryer stresses. In the [[

]].

The applicant describes the [[

]]. The applicant indicates that [[

]].

High-cycle fatigue cracking has occurred in some operating BWR steam dryers after extensive operating time (more than 10 to 20 years) at current licensed thermal power (before EPU operation). This implies that the frequency of the largest range of the alternating stress intensity experienced by the steam dryer is low [[ ]]. Therefore, the length of the pressure time history considered in the analysis should be long enough to assess the high-cycle fatigue damage during its intended lifetime of 60 years.

To characterize the high-cycle fatigue usage over a 60-year component life, the FIV loading used in the FEA of stress would consider peak stress intensities that occur at frequencies as low as approximately [[ ]]. The ESBWR steam dryer will experience [[ ]] during the design life of 60 years; the corresponding maximum allowable amplitude of the alternating stress intensity is about 106 MPa (15,400 psi), according to Figure I-9.2.2 in Appendix I to Division 1 of Section III of the ASME BPV Code (2001 Edition, 2003 Addenda). This value is conservative because the alternating stress intensity of 106 MPa (15,600 psi) is higher than the allowable fatigue limit of 93.7 MPa (13,600 psi) for the ESBWR steam dryer design. This conservatism is also indicated by the number of cycles ($1\text{x}10^{11}$ cycles determined from Figure I-9.2.2) for the fatigue limit of 93.7 MPa (13,600 psi), which is more than [[ ]] orders of magnitude higher than that of the [[ ]].

[[

]].

[[

]].

[[

]]. The NRC staff reviewed this approach and determined that it represents a reasonable method consistent with the ASME BPV Code provisions for stress evaluation because it accounts for stress underpredictions that may result with the use of these methods.

3.9.5.3.3.5.2.7 F-Factor and RMS Methods for the Calculation of Stresses Caused by Flow-Induced Vibration

The main objective of the F-factor method and the RMS method is to determine the change in the peak stress from the change in the measurements or PBLE01 loads. This can be accomplished because the FIV evaluation of a steam dryer follows a linear elastic analysis. These methods are computationally efficient and provide a nearly [[ ]] of the FIV stresses during power ascension and normal operation. These methods [[

]].

[[

]]. The [[ ]] method was validated during the EPU application for the VY operating plant.

[[

]].

The SSES licensee used the [[ ]] and [[ ]] methods during its EPU power ascension testing; and these two methods provided comparable results. The staff finds the use of the [[ ]] and [[ ]] methods acceptable for the ESBWR steam dryer because their technical bases are reasonable, and they were successfully used at VY and SSES to analyze the RSD performance during power ascension for the implementation of EPU operating conditions.

3.9.5.3.3.5.2.8 NRC Findings on ESBWR Steam Dryer Structural Evaluation

Based on its review of NEDE-33313P, as set forth above, and the applicant's responses to the RAIs, the NRC staff finds that the ESBWR steam dryer structural evaluation conforms to the guidance of RG 1.20 and the ASME BPV Code, Section III, Subsection NG, Article NG-3000, with justified exceptions, and satisfies the requirements of GDC 1, 2, and 4 of Appendix A to 10 CFR Part 50, and 10 CFR 50.55a. Accordingly, the staff finds that the ESBWR steam dryer is designed to adequately resist fatigue damage and to maintain structural integrity such that no loose parts are generated during its 60-year design life. This finding is based on the following factors:

- The ESBWR steam dryer design includes several features that increase its fatigue resistance: [[
]].

- PBLE01, used for estimating the pressure loading on the ESBWR steam dryer, is benchmarked with the pressure measurements on the GGNS RSD.

- Design loads have been conservatively derived from the acoustic pressure loads on steam dryers at a BWR/3 and a BWR/4 nuclear power plant, which include SRV resonance loads, using PBLE01. No SRV resonance loads will be present in the ESBWR operation because of enhancements to the ESBWR design. In addition, the ESBWR dryer design will satisfy an MASR equal to 2.0.

- End-to-end biases and uncertainties associated with the FEM method used for the steam dryer have been estimated from the analysis of the GGNS RSD. These bias and uncertainties will be applied to the ESBWR steam dryer design without taking credit for overprediction bias in completing the detailed design.

- Actual pressure loads acting on the steam dryer will be measured, and the corresponding fatigue stresses will be estimated during power ascension testing of the ESBWR steam dryer to further confirm the dryer's structural integrity.

- The ESBWR dryer will be instrumented with strain gages and accelerometers in addition to pressure sensors. The measured results will be used to confirm the fatigue resistance of the dryer at full power.

3.9.5.3.3.6 Steam Dryer Instrumentation for Startup Monitoring

Section 3L4.6, "Instrumentation and Startup Testing," of DCD Tier 2 states that the ESBWR steam dryer is instrumented with temporary vibration sensors to obtain FIV data during power operation. The DCD states that the primary function of the vibration measurement program is to confirm that the FIV load definition used in the structural evaluation is conservative with respect to the actual loading measured on the steam dryer during power operation, and to verify that the steam dryer can adequately withstand stresses from FIV forces for the design life of the steam dryer. The instrumentation and startup testing program for the ESBWR steam dryer follows RG 1.20.

Section 3L4.6 states that the steam dryer vibration sensors consist of strain gages, accelerometers, and dynamic pressure sensors appropriate for the application and environment. Table 3L-3, "Typical Vibration Sensors," provides a list of vibration sensors with their model numbers typical for steam dryer monitoring. The DCD states that the selection and total number of sensors is based on experience with other BWR steam dryers, and that the sensors are specifically designed to withstand the reactor environment.

Section 3L4.6 states that, prior to initial plant start-up, strain gages are resistance spot-welded directly to the steam dryer surface. Accelerometers are tack welded to pads that are permanently welded to the steam dryer surface. Surface mounted pressure sensors are welded underneath a specially designed dome cover plate to minimize flow disturbances that may affect the measurement.

Section 3L4.6 states that the data acquisition system consists of strain gages, pressure transducers, and accelerometers, as well as signal conditioning electronics. The locations of the sensors are selected to avoid pressure nodes in the acoustic harmonic response for frequencies that contribute most heavily to loading in the dryer components with the highest stress. The final pressure transmitter locations are evaluated using the PBLE01 model with multiple combinations of FRF sets corresponding to different transmitter locations. The resulting data are used to find locations that provide redundancy and minimize singularities over the frequency ranges of interest, with special consideration at frequencies critical to high stress locations in the dryer. The sensitivity of locations to dimensional tolerances is also considered. NEDE-33313P provides additional details on the steam dryer instrumentation.

Section 3L.4.6 specifies that the strain gages, accelerometers, and pressure transducers will be field calibrated prior to data collection and analysis. The DCD also states that this calibration will include the addition of natural strain gage factors based on the specific vendor supplied calibration sheets and their effects on the final stress tables. The DCD specifies that strain gage manufacturer installation procedures will be followed when installing the ESBWR steam dryer strain gages. In addition, care will be taken to assure surface preparation, welding energy, and weld strength recommendations are followed for each strain gage. Applicable lessons learned from the manufacturer's recommendations will also be incorporated into the welding procedure specification. Further, welder training will include pre-job briefs and discussion of the proper technique for applying the gages. In addition, the welders will practice on shims until peel tests are successfully completed. The DCD specifies that Quality Control personnel will be present to accept the welding process.

As discussed in Section 6.3.8, "Instrument Bias and Uncertainty," of NEDE-33313P, the instrumentation bias and uncertainty is accounted for in the design methodology when comparing predictions to measured values and also when establishing limits. The instrumentation bias and uncertainty addresses the overall accuracy of the total measurement system which includes the individual sensors (pressure transducers, strain gages, and accelerometers) as well as the signal conditioning devices that are used to convert the measured parameter to a physical measurement. For strain gages, additional factors are included in the overall measurement accuracy that reflect the bias and uncertainty introduced as a result of the installation process. The overall random signal conditioning devices uncertainty based on specified accuracies of the electronic devices is combined using the SRSS method. This uncertainty does not include drift of electronic devices with time, because the drift is not specified by the vendor. To account for drift, a conservative assumption taken from NRC-approved instrument setpoint methodologies is to assume that the drift for 6 months (or less) is equal to the instrument accuracy. The signal conditioning devices drift is then combined with the signal conditioning devices uncertainty and the sensor uncertainty using the SRSS method.

ITAAC 12, ITAAC 13, and ITAAC 14 in DCD Tier 1, Table 2.1.1-3, specify requirements to verify the installation of pressure sensors, strain gages, and accelerometers, respectively, on the as-built ESBWR steam dryer to monitor its performance during power ascension.

The NRC staff finds that the information provided in the DCD and NEDE-33313P on the steam dryer instrumentation to be acceptable based on the successful application of steam dryer instrumentation at other nuclear power plants. Specific instrumentation details will be made available for the as-built steam dryer to support completion of the ITAAC related to the ESBWR steam dryer. The NRC staff's evaluation of the ITAAC for steam dryer instrumentation is discussed later in this supplemental FSER.

3.9.5.3.3.7 ABWR/BWR Operating History and Experience with Replacement Steam Dryers Relevant to ESBWR Steam Dryer

In NEDE-33259P, the applicant describes testing of an ABWR plant in Japan and provides a table of selected FIV parameters measured in the ABWR and the estimates for the ESBWR.

Section 3L.4.1, "Steam Dryer Design and Performance," in DCD Tier 2 states that the prototype for the ESBWR steam dryer builds on the successful operating experience of the steam dryers at ABWR nuclear power plants in Japan. The ESBWR steam dryer will also draw on experience from RSD fabrication, testing, and performance at operating nuclear power plants in the United States. DCD Tier 2 states that the SRV and SV standpipes and MSL branch lines in the ESBWR will be specifically designed to preclude first and second shear layer wave acoustic resonances that could be a significant contributor to steam dryer loading at normal operating conditions.

Table 3L-1, "Comparison of Typical Major Steam Dryer Configuration Parameters," in DCD Tier 2 specifies design parameters related to the ESBWR steam dryer, the ABWR prototype steam dryer, and a BWR/3 RSD. For example, the average steamline flow velocity in the ESBWR will be similar to the velocity in the ABWR while much lower than the velocity in an operating BWR nuclear power plant that experienced steam dryer FIV problems.

NEDE-33312P indicates that a key aspect in the development of the ESBWR FIV load definition is to incorporate the ABWR steam dryer geometry. NEDE-33312P provides proprietary details in comparing the ESBWR steam dryer and the ABWR steam dryer. NEDE-33312P notes that there have been no identified FIV problems at the steam dryers in operating ABWR nuclear power plants in Japan.

In view of the above, the NRC staff finds that the design of the ESBWR steam dryer and its evaluation methodology will incorporate lessons learned from the operating experience with ABWRs in Japan and operating BWRs in the United States.

3.9.5.3.3.8 ESBWR Chimney Partitions Structural Integrity

DCD Tier 2, Section 3L.3 of Appendix 3L, describes how the applicant assessed the structural integrity of the chimney partition assembly using scale model testing (SMT), computational flow analyses, and FEM and stress analysis. The applicant computed a maximum stress of 41 MPa (5.95 ksi) using static analyses (based on its determination of a 2-Hz pressure fluctuation in the partition flow), which is less than the allowable 68.95 MPa (10 ksi) established by ASME design codes.

In RAI 3.9-140, the staff asked the applicant to provide the following information:

a) For flow conditions for which the two-phase pressure measurements were made on the chimney partition, provide the prototype conditions that they simulate, and describe the expected steam/water mixture flow rates and speeds through the chimney partitions. In addition, provide the magnitude and frequency content of the associated loads. Finally, discuss how the loading conditions resulting from flow in the mixing chamber at the top of the chimney were included in the two-phase load definition on the partitions.

b) Explain how the applicant's FEM considered fluid loading (resulting from exterior water and interior steam/water mixture) and the effects of the fluid loading on the model response, particularly for the 2-Hz pressure fluctuation. The applicant should also discuss the damping assumed in the chimney FEM, including the damping caused by the fluid loading.

c) Describe the structural attachments and constraints of the chimney partitions and the chimney, and justify the modeling of the boundary conditions in the FEM analysis.

In response to RAI 3.9-140(a), the applicant stated that the inlet flow conditions that were used in the test bound the actual flow conditions. The maximum load was measured in the 1/6-scale test at 7.5 kilopascal (kPa) [1.09 psi] (peak-to-peak/2) with 20 percent margin added, and the frequency was measured at 2 Hz. The applicant further stated that regarding the loading conditions resulting from the flow in the mixing chamber at the top of the chimney with respect to its effect on partitions, the test facility contained a tank that simulated the upper mixing chamber that was effective at collecting water as it exited the partitions. The staff concludes that the test setup described effectively models the two-phase flow and simulates the pressure conditions that occur in the mixing chamber. Therefore, the staff finds the applicant's response to RAI 3.9-140(a) acceptable.

In response to RAI 3.9-140(b), the applicant stated that the FEM applied a pressure load of 7.5 (kPa)[1.09 psi] uniformly on the plates and used the SRSS method for the sum of pressure loading between adjacent cells. This test focused on the chimney partitions and, as such, did not consider the effect of exterior water. The applicant further stated that, regarding the fluid loading, the eigenvalues of the partitions are 53.8 Hz (276 degrees Celsius (C) [529 degrees Fahrenheit (F)]) and 56.6 Hz (20 degrees C [68 degrees F]) without added mass, which is significantly higher than the 2 Hz dominant frequency of the fluid excitation. Therefore, dynamic effects were neglected and a static analysis was performed; no damping effects were considered. The staff agrees with the applicant's conclusions, because the relative stiffness of the chimney partition structure is sufficiently separated from the frequency of the FIV forcing function to support a static analysis approach.

In response to RAI 3.9-140(c), the applicant stated that the FEM analysis modeled the chimney partition cells as integral elastic bodies and assumed the outermost ends of the partitions to have fixed ends. The detailed design of the chimney partition structure will include structural support components at the outermost ends of the partitions to provide rigidity. The staff considered the applicant's response to RAI 3.9-140(c) acceptable because the analytical model provides a reasonable representation of the proposed design. However, in the audit at the applicant's offices in Wilmington, NC, on August 25, 2009, the staff determined that the detailed design of the chimney partition had not been completed. As a result, verification of the partition design and associated FIV stress analyses has been included as ITAAC 8b in Table 2.1.1-3 of DCD Tier 1, as discussed in Section 3.9.5.3.1 of this report and the SER for NEDE-33259P. Accordingly, the staff concludes that all aspects of RAI 3.9-140 are resolved.

3.9.5.3.3.9 ESBWR Reactor Pressure Vessel Internals Startup Monitoring Plans

Section 3L.5, DCD Tier 2, summarizes the program for preparing and performing the startup FIV monitoring for ESBWR reactor pressure vessel internals. The provisions of the startup test program are incorporated into the Initial Test Program detailed in Section 14.2 of DCD Tier 2. Table 3L-4, DCD Tier 2, specifies the reactor vessel internals (such as the steam dryer, chimney, and SLC line) to be monitored and the specific sensor type (such as strain gages, accelerometers, or pressure transducer) for each location. Section 3L.5 describes the data reduction and evaluation methods for the various reactor vessel internals.

In RAI 3.9-303, the NRC staff requested that the ESBWR design certification applicant revise the DCD or applicable engineering reports to describe the power ascension and monitoring program for the ESBWR steam dryer consistent with those programs for BWR operating nuclear power plants that had recently received EPU license amendments. In RAI 3.9-303, the staff specified key elements to be included in the ESBWR steam dryer power ascension monitoring and inspection program.

As discussed later in this supplemental FSER, the NRC staff finds that the DCD and applicable engineering reports have been revised to address the key provisions of the ESBWR steam dryer power ascension monitoring and inspection program specified in RAI 3.9-303. The staff has confirmed that the applicant has incorporated the planned changes in Revision 10 to the DCD, and NEDE-33313P, in response to RAI 3.9-303 as discussed in detail later in this supplemental FSER. Therefore, as discussed in more detail later in this supplemental FSER, RAI 3.9-303 is resolved. In the context of reviewing a COL application that references the ESBWR design, the staff will evaluate the specific provisions for incorporation into appropriate license conditions.

3.9.5.3.4 Criteria Used for Assessing the Adequacy of Internal Structures Other Than Steam Dryer and Chimney Assemblies, Including the Information from NEDE-33259P

The NRC staff recognizes that the criteria used by the ESBWR design certification applicant for assessing the adequacy of RPV internal structures other than the steam dryer and chimney assemblies are based on applicable codes and standards. The staff finds acceptable the criterion used for judging which components require additional evaluation and which components are considered acceptable and require no additional evaluation because of the similarity of the ABWR and ESBWR design and operating conditions. During its review of NEDE-33259P, the staff formulated additional RAIs to question potential FIV issues not addressed in the report.

In RAI 3.9-142, the staff asked the applicant to explain the fluctuating pressure expected to emanate from the various nozzles in the RPV adjacent to the chimney. This explanation was expected to include the reactor water cleanup/shutdown cooling (RWCU/SDC) nozzle, the isolation condenser (IC) return nozzle, and the GDCS nozzle near the chimney side walls, as shown in Figure 2 of NEDE-33259P.

In response, the applicant stated the following:

> Of the three systems that have nozzles and associated piping in the chimney region of the RPV, only the RWCU/SDC operates during normal plant operating conditions and has an external pump to drive flow. The other two systems are passive systems that do not operate during normal plant conditions and rely on hydraulic principles to create flow.
>
> For the RWCU/SDC system, the RPV nozzle is used to remove water from the RPV during normal plant conditions. The flow rate in this mode is a maximum of 2 percent of the feedwater flow, and is provided by a pump with comparatively low capacity. The vane passing frequency (VPF) from this pump will be similar to other pumps, but the amplitude will be very low. The BWR operating experience has been that only small sensing line components have been impacted by external pump vane passing frequencies.

> The Isolation Condenser (IC) system is only operated when containment isolation occurs and heat removal from the reactor system is required. When this system is opened, steam flow drives each of the closed loops and flow enters the RPV from the IC return line nozzle. Plant operation with this system in operation will be very limited, and with the large mass of the chimney structure no FIV issues will occur.
>
> For the Gravity Driven Cooling System (GDCS) lines, the only time these are placed in operation is during LOCA conditions when makeup water is required for the RPV. The flow from these nozzles is gravity driven from an elevated pool. The low associated flow rates and limited operating time, if such an event should ever occur, will not result in any vibration issues.

The staff finds that the operation of the ICS and GDCS will not result in vibration issues because these two systems are passive systems that do not operate during normal plant conditions and rely on hydraulic principles to create flow. In addition, plant operation with these systems engaged would be very limited.

However, the staff raised an issue regarding the pump-driven RWCU/SDC system that might produce FIV. Generally, the amplitudes of the pressure fluctuations resulting from VPFs from the pump are quite small. However, when the pulsation frequency coincides with the natural frequency of a component, the pressure pulsations can cause stresses of high magnitude even though the amplitude of the pressure fluctuations resulting from VPF is quite small. Small pressure fluctuations have been amplified in the steamlines of BWR plants and have caused pressure waves and vibrations that have damaged plant equipment, including steam dryers and SRVs. In RAI 3.9-142, S01, the staff asked the applicant to identify any vessel internal components that have natural frequencies that correspond to the pump VPFs. If such components were identified, the staff asked the applicant to submit analyses that clearly show that the stresses within those components are below the ASME Code fatigue limits.

In response, the applicant stated the following:

> Of the three systems that have nozzles and associated piping in the chimney region of the RPV, only the RWCU/SDC operates during normal plant operating conditions and has an external pump to drive flow. For the RWCU/SDC system, the RPV nozzle is used to remove water from the RPV during normal plant conditions. The flow rate in this mode is a maximum of 2 percent of the feedwater flow, and is provided by a pump with comparatively low capacity.
>
> The fluctuating pressure waves at the VPF produced by the RWCU/SDC pumps are not expected to affect the vessel internal components, or safety relief valves. Pressure waves at the VPF travels upstream and downstream from the pump. This pressure wave is attenuated due to flow path changes as it travels to the reactor. As the pressure wave enters the vessel, it is significantly attenuated because of the very significant increase in the flow area. The attenuation is expected to be related to the area ratio (vessel annulus area/nozzle area) squared. Thus, the small pressure fluctuations generated by the pumps is further reduced. In comparison to the current BWR forced-recirculation loops, which have much higher energy pumps and a shorter path of travel through piping and components the RWCU/SDC pumps produces much lower pressure induced vibration.

To ensure that resonance or near resonance conditions (between the vessel internals natural frequencies and the VPF) are not present, a comparison of the frequencies is made. The RWCU pump has 5 vanes and runs at 1780 revolutions per minute (rpm). This makes its VPF approximately 148 Hz. The Shutdown Cooling pump has 5 vanes running at 3550 rpm, resulting in a VPF of approximately 296 Hz.

The applicant also provided the lowest natural frequencies of the reactor components of interest near the vessel nozzle (SLC piping and the shroud, chimney, and separator). The applicant stated that because these lowest natural frequencies are far removed from the VPF, no resonance or near resonance conditions are present. It is possible for the higher modes of these components to be near the VPF. However, the responses for these higher modes are negligibly small, because (1) the response varies inversely as the frequency squared, and (2) the complex higher mode shapes result in very low generalized forces.

The staff finds this explanation acceptable, because the lower natural frequencies of the components of interest are sufficiently separated from the VPF to preclude application of potential forces associated with a resonance condition. Therefore, all aspects of RAI 3.9-142 S01 related to the clarification of the fluctuating pressure expected to emanate from the various nozzles in the RPV adjacent to the chimney, and vessel internal components that might have natural frequencies that correspond to the pump VPFs, are resolved.

In DCD Tier 2, Section 4.1.2.2, the applicant stated that individual fuel assemblies in groups of four rest on orifice fuel supports that are mounted on top of the CRGTs. Each guide tube, with its orifice fuel support, bears the weight of four fuel assemblies and is supported on a CRD housing penetration nozzle in the bottom of the reactor vessel. In RAI 3.9-143, the staff asked the applicant to clarify the load path and ensure that the weld at the nozzle is adequate to accommodate these loads. The staff asked the applicant to assess, in the event of weld failure, the adequacy of the CRGT and the CRD housing subjected to FIV and the ability to insert the control rod, considering the boundary conditions at the top of the CRGT and failed weld at the nozzle, and the CRGT base coupling connection with the CRD housing. The applicant responded to RAI 3.9-143 by stating the following:

> The CRD housing-to-CRD Stub Tube weld in the bottom head of the RPV carries the deadweight of four fuel assemblies, the orificed fuel support and the CRD guide tube. In addition, the weld carries the loads due to seismic and hydrodynamic accelerations as well as scram reaction loads, spring loads and vibratory loads. The load path is identical to that of earlier BWRs including the ABWR. A sketch of the CRD penetration was included in the applicant's response to RAI 4.5-19. The weld is analyzed, designed, manufactured and examined to be in full compliance with the requirements for ASME Code, Section III, Division 1, Class 1 pressure retaining components considering all the loads mentioned in the foregoing.
>
> The clearance between the CRD housing is controlled and kept as small as practicable for installation purposes. Thus, in the unlikely event of a complete weld failure, the transversal movement of the CRD Housing and the CRD Guide Tube is limited. FIV during this hypothetical condition would produce stresses in the CRD Guide Tube that are within the endurance limit as defined using the fatigue curve for austenitic SS, Figure I-9.2.1 of the ASME Code, Section III.
>
> A complete failure of the CRD housing-to-CRD Stub Tube weld is very unlikely. The existence of weld cracks in some older plants was discovered by leakage through

> the weld. The leakage started long in advance of any possibility of a complete weld failure. Also, the use of Columbium stabilized Alloy 82 weld material and Ni-Cr-Fe Alloy 600 stub tube material per ASME Code Case N-580-1 in the ESBWR has widely eliminated the concern for stress corrosion cracking in the weld and adjacent material.
>
> As mentioned in the foregoing, in the case of a complete weld failure, the transverse movement of the CRD Guide Tube is limited. The control rods and the control drive are designed to accommodate this misalignment during insertion of the control rods.

Based on its review, the staff requested in RAI 3.9-143 S01 that the applicant provide the following information to justify its conclusions:

1) Maximum transversal movement of the CRD housing and the CRGT (a) during normal operation and (b) under the condition with weld failure

2) Natural frequency of the worst system configuration with boundary conditions at the top of the CRGT, the CRGT base coupling connection with the CRD housing, and the failed weld at the bottom of the reactor vessel

3) Maximum cross-flow and longitudinal flow velocities along the system configuration identified in (2) above, and those at the CRGT-CRD housing coupling location

4) Results of the calculations for vortex shedding frequencies of the system configuration identified in (2) above, and the resulting maximum stress in the CRD

In response to RAI 3.9-143, S01, the applicant provided the following information:

> The reactor pressure vessel tube stub/CRD housing weld is part of the reactor coolant pressure boundary. As such it is designed, analyzed, fabricated, examined, and tested to ASME Section III, Subsection NB Class 1 requirements and is assigned the highest quality group classification A. This safety-related weld is designed and analyzed using seismic Category I loads and load combinations as shown in Tables 3.9.1 and 3.9.2 of the DCD Tier 2. This ensures the structural and functional integrity of the RPV and FMCRD [fine motion control rod drive]. The capability to insert the control rods is maintained under all plant operating events and dynamic loading events and load combinations as discussed in response to RAI 3.9-43. The material selection and fabrication process provide an extremely high probability of weld integrity as discussed in DCD Tier 2, Section 4.5. In conclusion there is an extremely low probability of leakage, of a rapidly propagating failure, and of gross rupture. If this weld were to fail (leak), it would be detected by the safety-related leak detection system as discussed in DCD, Tier 2, Subsection 5.2.5. The safety-related leak detection system indicates unidentified leakage through sump activity and sump level changes. The technical specifications specify limiting conditions of operation, required actions, surveillance requirements, and completion times to control the response as discussed in DCD, Tier 2, Chapter 16, Subsections 3.4.2 and 3.3.4.1. In the unlikely event of a gross weld rupture the radial clearance between the RPV tube stub and the CRD housing is very small (nominally 1/8 mm) which would minimize any transverse movement of a CRD housing. Frequency induced vibrations, stress, and flow are discussed in ESBWR Licensing Topical Report NEDE-33259.

The adequacy of the CRGT, CRD housing, and natural frequency and stress of the system configuration discussed in NEDE-33259P was based on the fixed end boundary condition at the penetration nozzle weld and not on the assumed complete weld failure. Therefore, the staff asked the applicant, in RAI 3.9-143 S02, to provide additional justification for its response to the four questions raised in RAI 3.9-143 S01. In response to RAI 3.9-143 S02, the applicant provided the following information:

> GEH no longer assumes the complete failure of the penetration nozzle weld. To ensure the structural integrity of the nozzle weld, it is analyzed, designed, fabricated, examined, and tested with the requirements of the ASME Code, Section III, Division 1, Class 1 pressure retaining components considering all the required loads mentioned in DCD Tier 2, Tables 3.9-1 and 3.9-2.
>
> For early BWR operating plants (BWR/2 plants and one BWR/3 overseas plant), stress corrosion cracking of furnace sensitized stainless steel CRD stub tubes that occurred were detected by leakage through the narrow annulus gap at the penetration. Subsequent plants used Ni-Cr-Fe Alloy 600 material, which has proven through many years of service to be crack resistant. For ESBWR, Columbium stabilized alloy 82 weld material and Ni-Cr-Fe Alloy 600 stub tube material per ASME Code Case N-580-1 has been selected to provide long term resistance to stress corrosion cracking. In the cases where leakage occurred, it was demonstrated, unlike typical nozzle designs where pipe separation can occur, the inherent features of the stub tube design provides a means to detect relatively small amounts of leakage that is readily detected, and significant structural margin remains such that there is no impact on the performance of the CRD. Therefore, the complete failure of the CRD penetration connection is not credible for design purposes, and does not need to be evaluated from a flow induced vibration perspective.

Additionally, to ensure the ability to insert the control rod, the applicant explained that the FMCRD is designed, fabricated, and tested as follows:

1) To quality standards commensurate with the importance of the safety-related functions to be performed in accordance with GDC 1 and 10 CFR 50.55a.

2) To withstand the effects of a safe-shutdown earthquake without loss of capability to perform its safety-related functions in accordance with GDC 2.

3) To assure the extremely low probability of leakage or gross rupture in accordance with GDC 14.

4) With appropriate margin to assure its reactivity control function under conditions of normal operation including anticipated operational occurrences in accordance with GDC 26.

5) With appropriate margin, and in conjunction with the emergency core cooling system, to be capable of controlling reactivity and cooling the core under postulated accident conditions in accordance with GDC 27.

6) To assure an extremely high probability of accomplishing its safety-related functions in the event of anticipated operational occurrences in accordance with GDC 29.

Based on the above, the staff finds the applicant's response to RAI 3.9-143 S02 acceptable in that the ESBWR design certification applicant has re-evaluated the penetration nozzle weld and determined that its complete failure is not credible. Based on this finding, the staff concludes that further FIV analyses of the CRGT and CRD housing and reconsideration of the ability to insert the control rod are not necessary. Also, in response to RAIs 3.9-143 and 3.9-143, S01, the applicant clarified the load path and provided the evidence to the staff that the weld at the nozzle is adequate to accommodate these loads. Therefore, all aspects of RAI 3.9-143 are resolved.

3.9.5.3.5 Loading Conditions

The NRC staff finds acceptable the loading conditions for which CS structures and safety-related internal components must satisfy the design basis (described in Section 3.9.5.2.6 of this report) because they include the significant loading events to which the structures and components are subjected. As indicated in Section 3.9.5.2.5 of this report, the applicant performed simulated flow tests for the chimney partition. As indicated in Section 3.9.5.2.5 of this report, the applicant has identified loading conditions for reactor internals. The applicant stated that it used the TRACG computer code to determine pressure differences for reactor internals during the events under different operating conditions.

Section 4.4 of the FSER contains the NRC staff's evaluation of the validation of TRACG for calculating the pressure differences for reactor internals during normal, upset, emergency, and faulted conditions.

The NRC staff describes its evaluation of the ESBWR steam dryer load definition in Section 3.9.5.3.3.3 of this supplemental FSER.

In that the natural circulation of the working fluid in the ESBWR is a new feature and only occurs when the fuel assemblies generate heat, the staff asked the applicant, in RAI 3.9-147, to justify that the flow velocities and their distribution over the reactor internals are verified for FIV analysis and testing, in accordance with SRP Section 3.9.2. In its response to RAI 3.9-147 dated November 22, 2006, the applicant explained how the working fluid flows in an ESBWR and highlighted positive aspects of the ESBWR design. The applicant stated that the flow paths are cleaner in an ESBWR, with fewer flow disturbances. In addition, the flow rates within the core region are slower than in a forced-circulation plant, leading to lower hydrodynamic excitation and resulting vibration. The NRC staff finds the applicant's explanation has clarified the natural circulation flow description for the working fluid in the ESBWR based on fluid dynamics with verification to be obtained during as-built testing. Therefore, RAI 3.9-147 is resolved.

3.9.5.3.6 Reactor Pressure Vessel Internals Design Bases

The NRC staff finds that the safety design basis and power generation design basis for reactor pressure vessel internals, as described in Section 3.9.5.2.6 of this report, are acceptable because they are based on the criteria established in applicable codes and standards for similar equipment, by manufacturer's standards, or by empirical methods based on field experience and testing. These criteria include the minimum safety factors provided for each of the four ASME BPV Code, Section III service conditions (Level A [normal], B [upset], C [emergency], and D [faulted]).

As indicated in Section 3.9.5.2.6 of this report, the applicant stated that, for the FIV of the chimney, the fundamental frequency of the chimney partition (approximately 54 Hz) was found to be much larger than the frequency of the maximum peak-to-peak pressure fluctuation (2 Hz). Therefore, the applicant performed an equivalent static analysis to show that the fatigue stress limits bounded the calculated stresses. Because the stresses are bounded, this approach is acceptable.

As indicated in Section 3.9.5.2.6 of this report, the applicant stated that the design and construction of the CS structures are consistent with ASME BPV Code, Section III, Division 1, Subsection NG. In RAI 3.9-148, the staff asked the applicant to identify the specific paragraphs of Subsection NG that are followed for the design and construction of the CS structures. In addition, in DCD Tier 2, Tables 3.9-4 through 3.9-7, the applicant provided the stress, deformation, and fatigue criteria for safety-related reactor internals (except CS structures), which are based on the criteria established in applicable codes and standards for similar equipment, by manufacturers' standards, or by empirical methods based on field experience and testing. Therefore, in RAI 3.9-148, the staff also asked the applicant to (1) identify the specific paragraphs of Subsection NG from which these criteria are derived or (2) if a basis other than the ASME BPV Code is used, identify and justify the other criteria (based on manufacturers' standards or empirical methods) that are used as the basis to develop the stress, deformation, and fatigue criteria for safety-related reactor internals.

In response to RAI 3.9-148, the applicant stated that the stress analysis of the reactor core support structures is performed in accordance with ASME BPV Code, Section III, Subsection NG, Subarticle NG-3200 for Service Conditions A, B, C, and D. In addition, the stress analysis uses ASME BPV Code, Section III, Appendix F, as applicable for Service Level D condition. An inelastic analysis method is used for a postulated blowout of a CRD housing caused by a weld failure, which is discussed in DCD Tier 2, Section 3.9.5.4. The staff finds the response related to core support structure acceptable, because the analytical process is consistent with the requirements of ASME BPV Code, Section III, Subsection NG, for the design of core support structures.

The applicant further stated that, for the stress analysis of reactor internal structures other than CS structures, it follows ASME BPV Code, Section III, Subsection NG, Subparagraph NG-1122(c), which states: "The Certificate Holder shall certify that the construction of all internal structures is such as not to affect adversely the integrity of the core support structures." In DCD Tier 2, Section 3.9.5, the applicant selected the safety factors for ASME BPV Code, Section III, Service Levels A, B, C, and D such that the calculated stress levels will meet the stress limits for CS structures given in ASME BPV Code, Section III, Subarticle NG-3200. The staff finds this explanation for the first three requirements (a, b, and c) of Table 3.9-5 acceptable, because the non-mandatory use of CS structure stress limits for design of internal structures is a conservative approach exceeding ASME requirements in accordance with ASME BPV Code, Section III, Subparagraph NG-1122(b).

The applicant further stated that the other criteria shown in Tables 3.9-4 through 3.9-7 are developed from Subsection NG of ASME BPV Code, Section III. In accordance with Subparagraph NG-3224.6, the deformation limit can be derived from the ultimate load determined by testing. The elastic limit, therefore, can be determined as a specified fraction of this load. In accordance with Subparagraphs NG-3228.4, NG-3224.1(e), and NG-3225, this fraction is 0.44, 0.6, and 0.88 for Service Levels A or B, C, and D, respectively. The staff finds this response acceptable.

The staff determined that the information presented in Table 3.9-5 did not address all its concerns; it requested the following information in RAI 3.9-148 S01:

1) Identify the specific paragraphs of Subsection NG for Requirement (d) as applied to Service Condition Levels A and B.

2) For Service Condition Level C, Requirement (d) provides a general limit of 0.6 times the ultimate strength (0.6 US), whereas Figure NG-3224-1 provides a smaller limit of 0.5 US. Please explain this difference.

3) The footnote (*) to equations e, f, g needs to be changed to read: "Equations e, f, g will not be used unless supporting data are provided to the NRC."

The following paragraphs discuss the staff's evaluation of the applicant's response to RAI 3.9-148 S01.

In response to RAI 3.9-148, S01, Item (1), the applicant stated that, according to Requirement (d) in DCD Tier 2, Table 3.9-5, for Service Levels A and B, the nominal primary stress evaluated using the elastic-plastic analysis is less than 0.4 times the US at temperature (i.e., EP ≤ 0.4 US).

The applicant explained that Subsection NG does not specifically refer to Requirement (d) for Service Levels A and B applicable to CS structures, and, therefore, to reactor internal structures. According to the applicant, however, this requirement for the reactor internal structures can be derived from Figure NG-3221-1, which specifies the primary stress in the CS structures to be less than 0.44 L_u for Levels A and B service conditions, where L_u is the ultimate load or the maximum load or load combination determined from the test on a prototype or model, as defined in NG-3228.4, and is the equivalent of US. Thus, Requirement (d) is comparable to the ASME BPV Code limit of 0.44 L_u for Service Levels A and B and satisfies the NG-1122(c) requirement that the reactor internal structures will not affect adversely the integrity of the CS structures. The staff finds this explanation acceptable, because the non-mandatory use of CS structure stress limits for design of internal structures is a conservative approach exceeding ASME requirements in accordance with ASME BPV Code, Section III, Subparagraph NG-1122(b), and, therefore, this part of the RAI is resolved.

For Service Condition D, Requirement (d) provides a general limit of 0.8 US, whereas ASME BPV Code, Section III, Appendix F, Subparagraph F-1341.2(b) provides a limit of 0.9 US. Therefore, Requirement (d) is acceptable for Service Level D.

In response to RAI 3.9-148, S01, Item (2), the applicant stated that, according to Requirement (d) in DCD Tier 2, Table 3.9-5, for Service Condition Level C, the nominal primary stress evaluated using elastic-plastic analysis is less than 0.6 times the US at temperature (i.e., EP ≤ 0.6 US).

The applicant explained that Subsection NG does not specifically refer to Requirement (d) for Service Level C applicable to CS structures, and, therefore, to reactor internal structures. But, according to the applicant, this requirement for the reactor internal structures can be derived from Figure NG-3224-1, which specifies the primary stress in the CS structures to be less than 0.6 L_e for Level C service conditions, where L_e is the ultimate load or the maximum load or load combination determined from the test on a prototype or model as defined in NG-3224.1(e) and is the equivalent of US. Thus, Requirement (d) is comparable to the ASME BPV Code limit of 0.6 L_e for Service Level C and satisfies the NG-1122(c) requirement that the reactor internal structures will not affect adversely the integrity of the CS structures. The staff finds this explanation acceptable, because the non-mandatory use of CS structure stress limits for design of internal structures is a conservative approach exceeding ASME requirements in accordance with ASME BPV Code, Section III, Subparagraph NG-1122(b), and, therefore, this part of the RAI is resolved.

In response to RAI 3.9-148, S01, Item (3), the applicant agreed to revise the footnote to equations e, f, and g as suggested by the staff. The applicant incorporated the revised footnote in DCD

Tier 2, Table 3.9-5. The staff finds this response acceptable. Based on the staff's evaluation, all aspects of RAI 3.9-148 are resolved.

3.9.5.3.6.1 Deformation Limits for Reactor Pressure Vessel Internals

DCD Tier 2, Table 3.9-4, provides deformation limits for safety class reactor internal structures. In RAI 3.9-149, the staff asked the applicant to provide the technical basis for the general limit listed in the table. In response, the applicant stated that, according to Appendix I to ASME BPV Code, Section II, Part D, the allowable stress intensity value, S_m, for austenitic SS is 90 percent of the minimum yield strength at temperature. The applicant has selected the minimum strain, ε, just before yielding of irradiated SS to represent the strain corresponding to minimum yield strength at temperature. The applicant stated that the magnitude of the minimum strain, ε, is based on experimental data from the industry.

To determine deformation limits, the applicant specified a general limit of $\leq 0.9/SF_{min}$, where SF_{min} is the minimum safety factor as defined in DCD Tier 2, Section 3.9.5.4.

According to DCD Tier 2, Section 3.9.5.4, SF_{min}, for Service Levels A to D vary from 2.25 to 1.125. The NRC staff finds the approach acceptable because the deformation limit is based on 90 percent of minimum yield strength and is therefore conservative. This is the only method set forth in the DCD for defining deformation limits in terms of minimum yield strain that the NRC approves.

In an earlier revision to the ESBWR DCD, the applicant had proposed another method that could have been used with experimental data pursuant to a departure from the approved method for defining deformation limits, as follows. The applicant stated that, when experimental data from the actual material are used, the general deformation limit $1.00/SF_{min}$ may be used instead of $0.9/SF_{min}$. In RAI 3.9-149, S01, the staff requested the following additional information for review:

a) The applicant should provide a reference for the industry data for irradiated SS as mentioned in its response. In addition, the applicant should summarize these industry data, especially the neutron fluence and irradiation temperature for the irradiated steel considered here. The applicant should also provide the end-of-the-life neutron fluence for the vessel internals that will be subject to deformation limits.

b) The applicant should provide the technical basis for the safety factors defined in DCD, Tier 2, Section 3.9.5.4.

c) The applicant should explain the increase in the general deformation limit from $0.9/SF_{min}$ to $1.0/SF_{min}$ when experimental data from the actual material are used. The applicant should also identify any codes or standards that support such an increase in the general deformation limit.

In response to RAI 3.9-149, S01, the applicant indicated that, if a COL applicant planned to perform any of the reactor internal structures qualification by the experimental data method, it would provide all of the supporting data to the staff for approval. In Revision 10 to the ESBWR DCD Tier 2, the applicant modified Table 3.9-4 to remove the provision allowing the use of a general deformation limit of $1.00/SF_{min}$. The staff finds the provision in Table 3.9-4 in Revision 10 of the ESBWR DCD Tier 2 for the use of a general deformation limit of less than or equal to $0.9/SF_{min}$ to be conservative and, therefore, acceptable. RAI 3.9-149, S01, is resolved.

3.9.5.3.6.2 Reactor Pressure Vessel Internals Vibration Tests

Since no preoperational FIV testing of the ESBWR will occur because it operates in a natural recirculation mode (as noted in DCD Tier 2, Section 3.9.2.4), the staff asked the applicant, in RAI 3.9-150, to discuss how the FEM computed natural vibration modes (vibration predictions) of the reactor internal components will be correlated with test data, as specified in SRP Section 3.9.5 and SRP Section 3.9.2, Item 4. In response, the applicant explained that, before startup testing, FEMs of the reactor internal components will be constructed and analyzed for their natural frequencies and mode shapes. Dynamic acceptance criteria for all accelerometers and strain gages to be placed on the components will be developed based on the FEM results. In addition, impact tests will be conducted before startup on instrumented components with an open reactor vessel at ambient conditions. The test results will be used to guide FEM revisions if they are deemed necessary.

The applicant's response states that impact tests will be performed for the first ESBWR. In RAI 3.9-150, S01, the staff asked the applicant to address in the DCD impact tests for the first and subsequent ESBWR plants. In response, the applicant explained that the objective of the first ESBWR reactor internals (except steam dryer) hammer tests is to identify the natural frequencies, mode shapes, and modal damping of the components of interest. The natural frequencies and mode shapes will be compared with those calculated using FEMs. If the calculated natural frequencies and mode shapes differ significantly from those obtained from the hammer test, then the FEMs will be refined such that the natural frequencies and mode shapes are in better agreement. The hammer test results will also serve as verification that the FEMs represent the ESBWR components realistically. For ESBWR plants subsequent to the first one, no hammer tests are planned because it is expected that the design of the RPV internal structures in subsequent ESBWRs will be identical to that of the first ESBWR. The staff finds the response acceptable because the applicant will validate its FEMs using the hammer test results. Therefore, all aspects of RAI 3.9-150 are resolved.

Section 5.1.3 in NEDE-33313P indicates that frequency response testing for the ESBWR steam dryer may involve hammer or shaker testing. NEDE-33313P indicates that excitation will be applied at multiple regions of the as-built ESBWR steam dryer. For each test, NEDE-33313P specifies that the input force, accelerations, transfer functions, and coherence at all accelerometers will be measured. The transfer functions for each measurement location are then calculated. NEDE-33313P specifies that the differences between the as-built steam dryer frequency response test results and the FE model predictions will be evaluated. If significant discrepancies are identified, adjustments will be made to the FE model or the FE stress analysis to address the dynamic testing results. The NRC staff finds the planned dynamic testing of the as-built ESBWR steam dryer to be acceptable based on experience with dynamic testing of RSDs for operating BWR nuclear power plants.

3.9.5.3.6.3 Potential Effects of Environmental Degradation over a 60-Year Design Life

In DCD Tier 2, Section 3.9.5, the applicant states that the ESBWR reactor plant design life is based on 60 years of plant operation. In RAI 3.9-245, the staff requested that the applicant describe the environmental conditions inside the reactor vessel and explain how the design of the reactor vessel internals accounts for potential degradation from environmental effects. The staff also asked the applicant to discuss potential degradation caused by environmental effects, such as intergranular and irradiation-assisted stress-corrosion cracking of SS and Inconel components, thermal embrittlement of cast SS components, and fatigue.

In response to RAI 3.9-245, the applicant stated that the susceptibility of the reactor internal components to intergranular stress-corrosion cracking (IGSCC), irradiation-assisted stress-corrosion cracking (IASCC) and thermal aging would be low during the ESBWR design life of 60 years because the normal operating conditions inside the reactor vessel are consistent with previous BWR designs. In addition, the vessel internals, including steam dryer, are made of IGSCC-resistant materials. The use of cast SS materials is limited to Grade CF3 material for which thermal embrittlement is not a potential degradation concern. DCD Tier 2, Sections 4.5.2.1 and 5.2.3.2.2, address IASCC considerations. However, in DCD Tier 2, Sections 4.5.2 and 5.2.3.2, the applicant did not address the issue of radiation-induced loss of fracture toughness of the internal materials and stress relaxation of the bolts used to fasten the reactor internal components.

In RAI 3.9-245, S01, the staff requested that the applicant explain whether the radiation-induced loss of fracture toughness of the internals materials and stress relaxation of the bolts would challenge the integrity of ESBWR reactor internals during the design life of 60 years. In response to RAI 3.9-245 S01 regarding the loss of fracture toughness, the applicant stated that the internal components being used in the ESBWR are bounded by the experience and levels of irradiation of current operating BWRs. As stated in DCD Tier 2, Sections 4.5.2 and 5.2.3.2.2, the ESBWR design incorporates materials and fabrication processes, as well as design features, to minimize welds and the potential for cracking. Therefore, radiation-induced loss of fracture toughness of the RPV internal structure materials and the stress relaxation of the bolts will not challenge the structural integrity of ESBWR reactor internals during its design life. The staff finds this response acceptable, because (a) the ESBWR RPV internals incorporate design features which tend to reduce degradation from irradiation, and (b) the irradiation levels during the 60-year design life of the threaded fasteners in the RPV internals are within acceptable levels, as further discussed below.

Regarding stress relaxation of the bolts, the applicant stated that DCD Tier 2, Section 3.9.3.9, addressed radiation effects for threaded fasteners. In addition, the design process for the reactor internal components includes the effects of stress relaxation from irradiation on threaded fasteners. However, the staff did not find this information in DCD Tier 2, Section 3.9.3.9, or in DCD Chapters 4 and 5. Consequently, during an audit held in the applicant's offices on August 25, 2009, the staff requested the following information:[3]

1) Locations of threaded fasteners used for the ESBWR RPV internals. What are the materials?

2) Provide a revised drawing of the connection between the chimney, shroud, and top guide.

3) What is the estimated end-of-life fluence for these fasteners?

4) What may be the maximum radiation-induced stress relaxation? Will it cause loosening of the threaded fasteners?

5) Are these fasteners susceptible to IASCC during the 60 years of service life?

6) What may be the loss of fracture toughness at the end of the 60-year service life? Will it challenge the structural integrity of the fasteners?

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[3] Audit Comment 17. NRC, "Report of the August 25, 2009 NRC Staff Audit on ESBWR RPV Internals," issued September 15, 2009, (ADAMS Accession No. ML0925704291).

In response, the applicant stated that the threaded fasteners for the core plate and top guide are the only fasteners that are located such that the effects of neutron radiation exposure are potentially significant. The material for these fasteners is Type XM-19 SS. The conservative estimates of axially averaged fast neutron fluence (E > 1 million electron volts) at peak azimuth for the ESBWR top guide studs and core plate studs at the end of 54 effective-full power years (EFPYs) are, respectively, 2.3×10^{19} neutrons per square centimeter (n/cm^2) and 1.0×10^{20} n/cm^2, and the corresponding stress relaxation are 8 percent and 22 percent. In that the core plate stud receives a larger fluence than the top guide stud, it is limiting. The design analysis of these fasteners ensures that sufficient preload is applied to prevent liftoff after accounting for thermal and irradiation-induced relaxation over the design life. Additional margin is applied to these end-of-life load relaxation factors to ensure that loosening does not occur from vibration or other potential relaxation mechanisms.

In addition, the applicant stated that, because the IASCC threshold for SS is 5×10^{20} n/cm^2, IASCC is not considered a plausible mechanism for these fasteners. Similarly, loss of fracture toughness is not a concern for these fasteners because the threshold for any significant loss of fracture toughness is 2×10^{20} n/cm^2. The staff finds the response only partially acceptable. The applicant stated that the average axial fluence would not exceed the threshold for IASCC and loss of fracture toughness, but the response does not ensure that the peak values of the axial fluence for these fasteners would not exceed the thresholds. Thus, in RAI 3.9-245 S02, the staff requested the information about the peak fluence values for these fasteners. The staff also requested a comparison of fast neutron fluences for these fasteners in the ESBWR, ABWR, and operating reactors and an evaluation of the need for a surveillance program to monitor the fast neutron fluence for the studs to ensure that it remains below the threshold values for IASCC and loss of fracture toughness.

In response to RAI 3.9-245 S02, the applicant stated that the peak fast neutron fluence for the top guide fasteners is conservatively estimated to be 8.9×10^{19} n/cm^2 at the end of 54 EFPYs, which is lower than the thresholds for IASCC and fracture toughness loss. Therefore, the top guide fasteners are not susceptible to IASCC and loss of fracture toughness during the 60-year design life.

The peak fast neutron fluence for the core plate fasteners is conservatively estimated to be 3.1×10^{20} n/cm^2 at the end of 54 EFPYs, which is lower than the thresholds for IASCC. Although the peak fluence is higher than the threshold for loss of fracture toughness, it is within the fluence range in which fully ductile fracture methods can be used for evaluating austenitic SS. In the vicinity of this fluence level, Type XM-19 SS retains significant tensile elongation and, therefore, any loss of fracture toughness with small fluence increases is likely to be small. Fracture toughness property measurements for material irradiated at higher fluence have confirmed high toughness as discussed in the EPRI report, Boiling Water Reactor Vessel and Internals Project (BWRVIP)-66, "BWR Vessel and Internals Project, Review of Test Data for Irradiated Stainless steel Components (BWRVIP-66)." Therefore, the top guide fasteners are not susceptible to IASCC and significant loss of fracture toughness during the 60-year design life. The staff finds the response acceptable, because the neutron fluence data for the RPV internals threaded fasteners demonstrate that the irradiation levels are below the threshold above which material degradation would be of concern. The applicant has revised DCD Tier 2, Section 5.3.2.1 and Table 5.3-4, to include this information about the fluence levels for the fasteners. The staff finds these revisions in Chapter 5 acceptable because ESBWR RPV fluence analysis results are also presented.

The applicant stated that it does not plan to include any surveillance program to address the concerns for IASCC and loss of fracture toughness because the peak fluence value at the ESBWR core plate stud is expected to be similar to that of ABWR and BWR/6 plants having comparable power ratings. The staff finds this response acceptable, because operating experience from reactors with comparable fluence levels do not indicate the need for a specific surveillance program for these RPV internals components. Therefore, all aspects of RAI 3.9-245 are resolved.

3.9.5.3.7 Combined License Information, ITAAC, and Model License Conditions

In the DCD, the ESBWR design certification applicant specified COL Information Items to be satisfied by the COL applicant referencing the ESBWR design certification, ITAAC to be met by the COL licensee related to reactor internals (including the steam dryer), and model license conditions related to the steam dryer monitoring plan (SDMP) that can be proposed by an applicant for a COL. The NRC staff reviewed the DCD and applicable engineering reports for the acceptability of these items as related to certification of the ESBWR design. The NRC staff issued numerous RAIs on the COL Information Items, ITAAC, and SDMP. In response to those RAIs, the ESBWR design certification applicant modified Section 10.0 in NEDE-33313P.

Section 10.1 in NEDE-33313P states that the ESBWR steam dryer is a prototype steam dryer under the guidance of RG 1.20. Because the ESBWR steam dryer is considered a prototype in the design certification, Section 10.1 states that each subsequent ESBWR steam dryer will also be considered a prototype. Section 10.1 indicates that subsequent ESBWR steam dryers would be considered a non-prototype under RG 1.20 only if the design certification is amended or future COL applicants or licensees seek NRC approval of a departure or exemption from the design certification requirements on a plant-specific basis.

Section 10.1.1 in NEDE-33313P states that a COL applicant will address COL Information Item 3.9.9-1-A for a prototype dryer. Section 10.1.1 in NEDE-33313P specifies that a COL applicant will prepare an as-designed ESBWR steam dryer analysis report. If a COL applicant for an initial ESBWR steam dryer design does not have some of the specified items prior to issuance of the COL, NEDE-33313P indicates that the COL applicant should follow the process in RG 1.206 to provide sufficient information for licensing and propose appropriate post-licensing commitments (e.g., ITAAC) to confirm the acceptability of the steam dryer. For the initial ESBWR steam dryer, NEDE-33313P indicates that an example application of the ESBWR steam dryer methodology has been provided for the design, analysis, and testing of an RSD at GGNS.

According to NEDE-33313P, the elements that are to be included in a Steam Dryer Design Analysis Report are as follows:

a. Describe the as-designed ESBWR dryer, dryer loading, and dryer stress analysis results.

b. Reference previously approved methodology in the DCD and Engineering Reports NEDE-33408P, NEDE-33312P, and NEDE-33313P.

c. Describe application of the bias and uncertainty as documented in the approved methodology.

d. Describe how the alternating peak stress intensities at the high stress locations were calculated (i.e., Method 1, Method 2, or Method 3 for weld locations); and tabulate the predicted alternating peak stress intensities.

e. Demonstrate final minimum alternating stress ratios (MASRs) greater than or equal to 2.0.

f. Include spectra and cumulative stress plots for the top five stress locations on the upper dryer and the top five stress locations on the lower dryer.

g. Describe a dryer dynamic test plan including sensor and drive locations sufficient to extract important resonances, with regional frequency response functions sufficiently resolved to establish regional bias and uncertainty for frequencies up to [[ ]].

h. Incorporate lessons learned from power ascension of previous ESBWR plants, as applicable.

Section 10.1.2 in NEDE-33313P states that ITAAC 16 in DCD Tier 1, Table 2.1.1-3 is intended to verify that the as-built steam dryer fatigue analysis provides at least a minimum MASR of 2.0 to the allowable alternating stress intensity of 93.7 MPa (13,600 psi). In particular, Section 10.1.2 states that the following elements are to be included in a Steam Dryer As-Built Analysis Report:

a) Describe changes between the as-designed and as-built steam dryers, including adjustments to the structural FE model, updated bias and uncertainty based on testing, and updated stresses and stress ratios.

b) Demonstrate that the as-built ESBWR steam dryer with the assumed pressure loading satisfies the methodology to calculate the resulting dryer alternating stress with at least an MASR of 2.0 as described in the DCD and its engineering reports.

c) For the dryer dynamic testing, specify the minimum number of excitation locations to ensure adequate coverage of the dryer, and that enough resonances are extracted so that comparisons may be made to simulations up to [[ ]]. Specify how the dryer will be subdivided into sensor groups/regions, whether multiple excitation locations will be specified within a group/region, and how the different regional errors for different excitation locations will be addressed.

d) Address the uncertainties in the comparison of predicted mode shapes with those measured during the dryer dynamic testing (i.e., boundary conditions and dryer support).

e) Address differences of greater than [[ ]] between predicted resonance frequencies and those measured during the dryer dynamic testing to ensure worst case coupling between peak excitation and peak response is captured.

f) Identify on-dryer instrumentation sensor specifications, sensor locations (including at least [[ ]] and at least [[ ]]), and correlations between sensors and peak stress locations on the upper and lower dryer.

g) Identify all biases and uncertainties associated with the sensors and data acquisition system.

h) Provide the acceptance limits for each sensor with supporting calculations (spectra and time histories). The limits should extend to 1 kHz based on the potential for HF excitation tones. Explain how the limits are derived from calculations using the minimum load case method described in Section 9 of NEDE-33313P. Limit curves for power ascension will be based on the worst case of both the design-basis calculations that use the end-to-end GGNS bias and uncertainty, and those from the as-built steam dryer calculations that use the combined FE structural and PBLE01 biases and uncertainties.

i) Confirm that redundant pressure sensors will be located adjacent to each MSL inlet.

j) Describe the ESBWR steam dryer power ascension monitoring and inspection program.

Section 10.1.3 in NEDE-33313P states that a structural assessment will be performed to benchmark the FEM strain and acceleration predictions against the measured data. The dryer stresses will be determined using the ESBWR on-dryer based measurement FIV load definition and adjusted for end-to-end B/U determined from the FEM strain and acceleration benchmark. A fatigue limit stress amplitude of 93.7 MPa (13,600 psi) with an MASR of 1.0 will be used as the acceptance limit for this confirmatory stress analysis using actual ESBWR on-dryer data.

Section 10.2 in NEDE-33313P supplements the DCD description of the comprehensive vibration program elements as to how the program will be implemented. Section 10.2 also describes actions to be completed by the COL licensee related to the power ascension monitoring and inspection program. Section 10.2 provides specific actions by the COL applicant and later the COL licensee to satisfy the provisions of COL Information Item 3.9.9-1-A listed in DCD Tier 2, Section 3.9.9.

For the prototype ESBWR steam dryer, item 2 of COL Information Item 3.9.9-1-A specifies that the COL applicant will:

a) provide a milestone of no later than 90 days before startup to prepare and provide to the NRC a Steam Dryer Monitoring Plan as described in NEDE-33313P, Section 10;

b) submit or reference a steam dryer predicted analysis (for the plant-specific or a sample steam dryer) that concludes the steam dryer will not exceed stress limits with applicable bias and uncertainties and the MASR of 2.0;

c) describe startup program (with proposed license conditions) that includes appropriate notification points during power ascension, and submittal of the completed analysis of steam dryer data within 90 days following completion of the power ascension testing and monitoring of the steam dryer; and

d) specify periodic steam dryer inspections during refueling outages (Subsection 3.9.2.4).

To satisfy item (a) in COL Information Item 3.9.9-1-A for the steam dryer, Section 10.2 in NEDE 33313P states that:

> A Steam Dryer Monitoring Plan (SDMP) for each ESBWR steam dryer will be prepared and provided to the NRC no later than 90 days before startup of the applicable ESBWR unit. The SDMP will reflect industry experience with the performance of steam dryer power ascension testing. The SDMP shall include the following, which shall be augmented or modified as appropriate to address industry experience:
>
> - Criteria for comparison and evaluation of projected strain levels with data obtained from the on-dryer instrumentation
> - Acceptance limits developed for selected on-dryer strain gage and accelerometer locations
> - Tables of predicted steam dryer stresses at 100 percent power; strain amplitudes and power spectral densities (PSDs) at strain gage locations;

predicted acceleration amplitudes and PSDs at acceleration locations; and maximum stresses and locations

- Directions for establishing correlations between measured accelerations and strains and the corresponding maximum stresses
- Identification of steam dryer strain gage locations for which limit curves will be developed, and criteria for selection of those locations
- Methodology for developing projected strain levels for the next power level and for full power
- Specific assessment points during power ascension. While completing assessment, power will remain steady to determine if any actions need to be taken or if power may ascend to the next level.
- Activities to be accomplished during assessment points
- Details of the installation and calibration of the steam dryer instrumentation with the instrumentation mounted and calibrated in accordance with the manufacturers' instructions to accurately measure the dynamic response

To satisfy item (b) in COL Information Item 3.9.9-1-A for the steam dryer, Section 10.2 in NEDE-33313P states that the COL applicant will include a reference to the demonstration of the ESBWR steam dryer structural integrity process that is described in NEDE-33408P. Alternatively, Section 10.2 states that the COL applicant could submit or reference an ESBWR steam dryer that has been subject to the predicted analysis process and successful startup ascension testing.

To satisfy item (c) in COL Information Item 3.9.9-1-A for the steam dryer, Section 10.2 in NEDE-33313P provides key elements for developing license conditions for implementing the startup monitoring program. Section 10.2 states that the model license conditions are key elements of the power ascension test procedures applicable to the steam dryer, which are described further in DCD Section 3L.5, but does not bind either a COL applicant or the NRC staff. A COL applicant may propose the model condition or a different condition in its application, and the NRC is free to exercise its discretion to include a license condition governing the startup test program as applied to the steam dryer in a COL that references the ESBWR design. The key elements specified in Section 10.2 are as follows:

1) Power Ascension Test (PAT) procedures for the steam dryer testing will be provided to NRC inspectors no later than 10 days before start-up. The PAT procedures will include the following:

   a) Level 1 and Level 2 acceptance limits for on-dryer strain gages, and on-dryer accelerometers to be used up to 100 percent power

   b) specific hold points and their duration during 100 percent power ascension

   c) activities to be accomplished during hold points

   d) plant parameters to be monitored

   e) actions to be taken if acceptance criteria are not satisfied

f) verification of the completion of commitments and planned actions

2) The initial hold point during the first startup of each ESBWR plant will be at no more than 75 percent of full power. At this hold point:

a) Record pressures, strains, and accelerations from the on-dryer mounted instrumentation. Evaluate the data and compare the measured dryer strains and accelerations to acceptance limits.

b) Develop a PBLE01-based ESBWR FIV load definition based on selected on-dryer instruments. Using appropriate methods, such as F-factor and RMS, and the above PBLE01-based ESBWR load definition, predict the steam dryer strain and acceleration response at this condition.

c) Compare the predicted steam dryer strain and acceleration against the measured data and determine frequency dependent end-to-end bias and uncertainty values. Adjust the predicted strain and acceleration responses using the frequency-dependent end-to-end bias and uncertainty values. If any of the measured sensor data exceed the adjusted predictions, then either modify the bias errors and uncertainty values and limit curves and ensure measured sensor responses do not exceed the adjusted predictions, or quantitatively evaluate the impact on fatigue life.

d) Define the steam dryer peak stress projections based on the revised results from step b with modified end-to-end bias and uncertainties from step c. Compute the steam dryer maximum stress and minimum stress ratio from the predictive analysis using up to a [[ ]] of load applications. Prepare cumulative stress plots for at least the [[ ]] most highly stressed locations on both the upper and lower dryer with the dominant stress component at each location used for the plots. The peak stress amplitude adjusted for the bias and uncertainty is maintained less than 93.7 MPa (13,600 psi).

e) Update limit curves based on the results from step d. Level 1 and Level 2 limit curves will be generated for all functioning strain gage and accelerometer locations on the steam dryer and will include bias errors and uncertainties as described in the applicable engineering reports.

f) Trend the recorded data and project the stress, strain, and accelerometer sensor responses for the next assessment point and full power to demonstrate margin for continued power ascension.

g) Make available to the NRC the ESBWR steam dryer analysis summary, updated stress analysis results (including end-to-end bias and uncertainty), limit curves, and data projections for higher power levels.

3) Subsequent hold points will be in approximately 5 percent power level increments where pressures, strains, and accelerations will be recorded and

evaluated. Data trending and a projection of strain levels will be generated for the next hold point and full power. Data trending analysis during power ascension must assess whether the limits would be violated at higher power levels. Data trending results and revised limit curves will be made available to the NRC at each hold point.

4) Power ascension monitoring shall address expected increases in loading and fatigue damage due to variable plant conditions throughout the life of the dryer.

5) During initial power ascension, if flow-induced resonances are identified and the strains or vibrations increase above the pre-determined criteria, power ascension is stopped. The acceptability of the steam dryer for continued operation is evaluated [[ ]]. The limit curves are then redefined based on the on-dryer data. The limit curve factor is revised [[ ]]. If a Level 1 limit curve is exceeded, power will be reduced to a previous power level where Level 1 was not exceeded and a stress analysis will be performed to develop new limit curves. During initial power ascension, should a Level 2 limit curve be exceeded, or if the trending indicates that a Level 1 limit may be challenged prior to reaching the next hold point, the acceptance limits will be evaluated, and revised if appropriate.

6) End-to-end bias and uncertainties shall be determined by comparing the predicted and measured strain or acceleration on the steam dryer at each hold point to confirm the conservatism of the predicted dryer stress field. Adjust the predicted strain and acceleration responses using the frequency-dependent end-to-end bias errors and uncertainty values. If any of the measured sensor data exceed the adjusted predictions, then either modify the bias errors and uncertainty values and limit curves and ensure measured sensor responses do not exceed the adjusted predictions, or quantitatively evaluate the impact on fatigue life.

7) At the initial hold point and the hold points at approximately 85 and 95 percent power, power ascension will not proceed for at least 72 hours after making the steam dryer data analysis and results available to the NRC, unless notified by the NRC that power ascension may proceed earlier. [The NRC staff notes that the last clause of this provision will be deleted and the provision modified in an actual license condition.]

8) During the Power Maneuvering in the Feedwater Temperature Operating Domain testing, pressures, strains, and accelerations will be recorded from the on-dryer mounted instrumentation across the expected range of normal steady state plant operating conditions. An evaluation of the dryer structural response over the range of steady state plant operating conditions will be included in the stress analysis report described in Item 9 below.

9) After full power has been achieved, data at the full power level will be provided to the NRC within 72 hours, and a full stress analysis report and evaluation will be provided to the NRC within 90 days of reaching the full power level. The report will include the minimum stress ratio and the final dryer load definition using steam dryer instrumentation, and associated bias

errors and uncertainties, to demonstrate that the steam dryer will maintain its structural integrity over its design life considering variations in plant parameters (such as reactor pressure and core flow rate).

To satisfy item (d) in COL Information Item 3.9.9-1-A for the steam dryer, Section 10.2 in NEDE-33313P states that a periodic steam dryer inspection program will be implemented with the following key elements:

1) During the first two scheduled refueling outages after reaching full power conditions, a visual inspection will be conducted of all accessible areas and susceptible locations of the steam dryer in accordance with accepted industry guidance on steam dryer inspections. The results of these baseline inspections will be provided to the NRC within 60 days following startup after each outage.

2) At the end of the second refueling outage following full power operation, an updated SDMP reflecting a long-term inspection plan based on plant-specific and industry operating experience will be provided to the NRC within 180 days following startup from the second refueling outage.

The NRC staff has evaluated Section 10.0 in NEDE-33313P for the actions to be accomplished by the COL applicant and COL licensee to satisfy COL Information Item 3.9.9-1-A, applicable ITAAC, and post-startup activities, as applicable. The staff addresses these actions in the following discussion.

In its response to RAIs 3.9-289, -290, and -291, the ESBWR design certification applicant proposed COL Information Item 3.9.9-1-A to request the COL applicant to classify its reactor per the guidance in RG 1.20 and provide a milestone for submitting a description of the inspection and measurement programs to be performed and the results of the program. The NRC staff determined that this proposed COL Information Item did not address the complete set of guidance in RG 1.20 for preventing potential adverse flow effects on the ESBWR steam dryer. In RAI 3.9-291 S02, the NRC staff requested that the ESBWR design certification applicant revise the COL Information Item to specify that the COL applicant will implement the recommendations in RG 1.20 for a comprehensive vibration assessment program. In response to RAI 3.9-291 S02, the ESBWR design certification applicant revised COL Information Item 3.9.9-1-A to provide more specificity on the actions to be taken by the COL applicant. However, the staff determined that COL Information Item 3.9.9-1-A needed to be further clarified regarding specific actions to be taken by the COL applicant.

In RAI 3.9-301, the NRC staff requested that the ESBWR design certification applicant clarify the items that the COL applicant should provide in an “as-designed” ESBWR steam dryer analysis report. If the analysis cannot be completed prior to COL issuance, the NRC staff noted that the COL applicant should follow the process in RG 1.206 to provide sufficient information for licensing and propose appropriate post-licensing commitments. Such commitments could be verified by ITAAC, license conditions, or FSAR changes to confirm the acceptability of the steam dryer as installed.

In response to RAI 3.9-301, the ESBWR design certification applicant revised COL Information Item 3.9.9-1-A to read as follows:

The COL applicant will:

(1) for the reactor internals, other than steam dryer, classify its reactor per the guidance in RG 1.20 and provide a milestone for submitting a description of the inspection and measurement programs to be performed (including measurement locations and analysis predictions) and the results of the vibration analysis, measurement and test program.

(2) for the steam dryer, which is classified as a prototype per the guidance in RG 1.20, (a) provide a milestone of no later than 90 days before startup to prepare and provide to the NRC a Steam Dryer Monitoring Plan as described in NEDE-33313P, Section 10; (b) submit or reference a steam dryer predicted analysis (for the plant-specific or a sample steam dryer) that concludes the steam dryer will not exceed stress limits with applicable bias and uncertainties and the minimum alternating stress ratio (MASR) of 2.0; (c) describe startup program (with proposed license conditions) that includes appropriate notification points during power ascension, and submittal of the completed analysis of steam dryer data within 90 days following completion of the power ascension testing and monitoring of the steam dryer; and (d) specify periodic steam dryer inspections during refueling outages (Subsection 3.9.2.4).

The reactor internals vibration analysis, measurement and inspection information that the first COL applicant and subsequent COL applicants need to provide, at the time of application, related to reactor vessel internals, including the CS structures (except the steam dryer), is explained in NEDE-33259P.

In response to RAI 3.9-301, the ESBWR design certification applicant revised NEDE-33313P to include Section 10.1.1, which states that a COL applicant will address COL Information Item 3.9.9-1-A for a prototype dryer. NEDE-33313P specifies that a COL applicant will prepare an as-designed ESBWR steam dryer analysis report. If a COL applicant does not have some of the specified items prior to issuance of the COL, NEDE-33313P indicates that the COL applicant should follow the process in RG 1.206 to provide sufficient information for licensing and propose appropriate post-licensing commitments (e.g., ITAAC) to confirm the acceptability of the final steam dryer as installed. The NRC staff finds the elements for the as-designed ESBWR steam dryer analysis report in Section 10.1.1 of NEDE-33313P to be acceptable.

As discussed above, the ESBWR design certification applicant revised Section 10.2 in NEDE-33313P in response to RAIs 3.9-289, 290, 291, 291 S02, and 301 to provide specific provisions for the COL applicant to address items (a) through (d) of COL Information Item 3.9.9-1-A. Each item in Section 10.2 in NEDE-33313P specifies the actions to be completed by the COL applicant (and later the licensee through its SDMP) to successfully implement COL Information Item 3.9.9-1-A for the design, fabrication, analysis, power ascension monitoring, and periodic inspection of the ESBWR steam dryer. The staff finds that the changes to NEDE-33313P have specified an acceptable approach to address COL Information Item 3.9.9-1-A. Therefore, RAIs 3.9-289, 290, 291, 291 S02, and 301 are resolved.

DCD Tier 1, Table 2.1.1-3, specifies in ITAAC 8.b that the RPV internal structures (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) will meet the provisions of ASME BPV Code, Subsection NG-3000, except for the weld quality and fatigue factors for secondary structural non-load bearing welds.

In RAI 3.9-302, the staff requested that the ESBWR design certification applicant include its description of the process for a licensee to satisfy ITAAC 8.b in DCD Tier 1, Table 2.1.1-3 in the DCD or the applicable engineering report consistent with the responses to RAIs 3.9-299 to 303. In response to RAI 3.9-302, the ESBWR design certification applicant included Appendix A, in NEDE-33313P to describe the process for satisfying this ITAAC. For example, Appendix A to NEDE-33313P states that the ASME BPV Code design report to be prepared for ITAAC 8b will include sufficient detail to show that the applicable stress limitations are satisfied in ASME BPV Code, Section III, Article NG-3000, as applicable to the steam dryer, when the component is subject to the loading conditions specified in the Design Specification. The appendix also states that a licensee will conduct an inspection of the fabricated, as-built steam dryer prior to its installation into the RPV to compare the as-built steam dryer to the ASME Code design report, as well as supporting documentation for the design report, which will include documents such as the structural evaluation, construction drawings, deviations, repairs, procurement documentation with receipt inspection records, and fabrication records. If any discrepancies are identified, the appendix states that those will be dispositioned prior to the licensee issuing an ITAAC closure notification letter. Once ITAAC-related activities are completed, the licensee would process a closure notification letter to the NRC in accordance with NRC-endorsed guidance and the ITAAC Closure Plan. The NRC staff finds the description in Appendix A to NEDE-33313P to be acceptable for the licensee to satisfy ITAAC 8.b in DCD Tier 1, Table 2.1.1-3. Therefore, RAI 3.9-302 is resolved.

DCD Tier 1, Table 2.1.1-3 specifies in ITAAC 12, 13, and 14 requirements for the installation of instrumentation on the as-built steam dryer to monitor its performance during power ascension. In particular, ITAAC 12 states that the number and locations of pressure sensors installed on the steam dryer for startup testing ensure accurate pressure predictions at critical locations. ITAAC 13 states that the number and locations of strain gages and accelerometers installed on the steam dryer for startup testing are capable of monitoring the most highly stressed components, considering accessibility and avoiding discontinuities in the components. ITAAC 14 states that the number and locations of accelerometers installed on the steam dryer for startup testing are capable of identifying potential rocking and of measuring the accelerations resulting from support and vessel movements. The NRC staff finds that these ITAAC will verify the installation of the monitoring instrumentation on the as-built steam dryer consistent with the ESBWR steam dryer methodology described the DCD and its incorporated engineering reports.

DCD Tier 1, Table 2.1.2-3, specifies in ITAAC 36 that the MSL and SRV/SV branch piping geometry will preclude first and second shear layer wave acoustic resonance conditions from occurring and avoids excessive pressure loads on the steam dryer at plant normal operating conditions.

In RAI 3.9-302, the NRC staff requested that the ESBWR design certification applicant include the process for the licensee to satisfy ITAAC 36 in DCD Tier 1, Table 2.1.2-3 in the DCD or the applicable engineering report consistent with the responses to RAIs 3.9-299 to 303. In response to RAI 3.9-302, the ESBWR design certification applicant included Appendix B in NEDE-33313P to describe the process to satisfy this ITAAC. For example, Appendix B to NEDE-33313P states that the licensee will include documented evidence of an analysis in the ITAAC closure package to ensure that the as-built piping and the piping branch-connected SRVs and SVs are designed to preclude first and second shear layer wave acoustic resonance conditions from occurring. The acceptance criteria for the piping and valve as-built dimensions will be contained in an acoustic resonance calculation that provides documented evidence that first and second shear layer wave resonances will not occur with final design dimensions. Once the main steam piping is installed, final documentation will be added to the list of documents in the ITAAC closure package of supporting information that demonstrate acceptance criteria are met. The COL licensee would then process a closure notification letter to the NRC in accordance with NRC endorsed guidance and the ITAAC Management Plan. The NRC staff finds the description in Appendix B to NEDE-33313P to be acceptable for the COL licensee to satisfy ITAAC 36 in DCD Tier 1, Table 2.1.2-3.

In its response to RAIs 3.9-291 and -291 S05, the ESBWR design certification applicant stated that ITAAC 16 will be included in DCD Tier 1, Table 2.1.1-3, to specify that a report of the fatigue analysis of the as-built steam dryer exists and demonstrates that the maximum calculated alternating stress intensity provides at least a minimum MASR of 2.0 to the allowable alternating stress intensity of 93.7 MPa (13,600 psi). The staff finds the applicant's response to RAIs 3.9-291 and -291 S05 with the plan to include the described ITAAC 16 in ESBWR DCD Tier 1, Table 2.1.1-3, to be acceptable. However, the NRC staff requested in RAI 3.9-302 that the ESBWR design certification applicant revise the DCD or applicable engineering report to describe what must be included in the report to satisfy ITAAC 16.

In response to RAI 3.9-302, the ESBWR design certification applicant revised NEDE-33313P to include Section 10.1.2 for information to satisfy ITAAC 16 in DCD Tier 1, Table 2.1.1-3 in verifying that the as-built steam dryer fatigue analysis provides at least a minimum MASR of 2.0 to the allowable alternating stress intensity of 93.7 MPa (13,600 psi). The NRC staff finds that the information described in Section 10.1.2 of NEDE-33313P for the COL licensee to satisfy ITAAC 16 in DCD Tier 1, Table 2.1.1-3 in verifying that the as-built steam dryer fatigue analysis provides at least a minimum MASR of 2.0 to the allowable alternating stress intensity of 93.7 MPa (13,600 psi) to be acceptable. The staff has confirmed that Revision 10 to the DCD, and NEDE-33313P, include the planned changes related to ITAAC 16. Therefore, RAIs 3.9-291, 291 S05, and 302 are resolved.

In RAI 3.9-293 S01, the NRC staff requested that the ESBWR design certification applicant clarify the DCD and applicable engineering reports that the steam dryer methodology proposed for the ESBWR design certification is the same for both the prototype and subsequent ESBWR plants. In response to RAI 3.9-293 S01, the ESBWR design certification applicant stated that each ESBWR steam dryer is considered a prototype under the proposed revisions to the ESBWR licensing basis described in the steam dryer RAI responses and as clarified in the DCD and associated engineering report. After at least one successful prototype has been through the startup testing and vibration assessment program, the ESBWR design certification applicant stated that a subsequent ESBWR COL applicant or COL licensee may (but need not) elect to seek NRC approval of a departure and exemption from the DCD items associated with the prototype versus non-prototype steam dryer. However, because such an approach would not be consistent with the licensing basis for the ESBWR design certification, the ESBWR design certification applicant stated that such a departure/exemption would involve a Tier 2* change and potentially a Tier 1 change to the ITAAC, both of which would require prior NRC review and approval. The ESBWR design certification applicant stated that any such proposal for a departure from the DCD by a COL applicant or COL licensee would follow NRC guidance in RG 1.20 for establishing the process for a nonprototype steam dryer. The ESBWR design certification applicant stated that a non-prototype steam dryer is no longer addressed in the DCD and a departure to use a non-prototype approach would, therefore, be subject to NRC approval in future licensing actions. The NRC staff finds the application of the ESBWR steam dryer methodology to prototype and nonprototype steam dryers to be acceptable as described in this supplemental FSER. The staff has confirmed that Revision 10 to DCD Tier 2, Section 3.9.2.3, and Section 1.0 in NEDE-33313P, specify the provisions for the prototype and subsequent ESBWR steam dryers consistent with the response to RAI 3.9-293 S01. Therefore, RAI 3.9-293 S01 is resolved.

In RAI 3.9-303, the NRC staff requested that the ESBWR design certification applicant revise the DCD or applicable engineering reports to describe the power ascension and monitoring program for the ESBWR steam dryer consistent with those programs for BWR operating nuclear power plants that had recently received EPU license amendments. In RAI 3.9-303, the staff suggested key elements of the ESBWR steam dryer power ascension monitoring and inspection program. As described above, the ESBWR design certification applicant has revised Section 10.0 in NEDE-33313P to specify key elements of the ESBWR steam dryer power ascension monitoring and inspection program. The NRC staff finds Section 10.0 in NEDE-33313P to provide key elements of the ESBWR steam dryer power ascension monitoring and inspection program acceptable for the ESBWR design certification. As indicated in the key elements of the ESBWR steam dryer power ascension monitoring and inspection program, the licensee will evaluate the steam dryer data when full power is achieved and will take appropriate action if necessary (such as reducing power or assessing the inspection interval) if the steam dryer data or stress analysis reveals that the static or fatigue limits might be exceeded. The NRC staff will develop specific license conditions for the ESBWR steam dryer power ascension monitoring and inspection program for each COL license. The staff has confirmed that NEDE-33313P includes the planned changes in response to RAI 3.9-303. Therefore, RAI 3.9-303 is resolved.

3.9.5.3.8 Tier 2* Information

In response to RAI 3.9-292, the ESBWR design certification applicant described changes to its ESBWR steam dryer licensing documents in response to several RAIs. In RAI 3.9-292 S03, the NRC staff requested that the ESBWR design certification applicant provide a final version of the proposed revision to the DCD and other licensing documents to incorporate all changes related to the ESBWR steam dryer review. The staff requested that the ESBWR design certification applicant identify the ESBWR steam dryer design and analysis information in the DCD and applicable engineering reports that is considered Tier 2* information. In response to RAI 3.9-292, S03, the ESBWR design certification application provided the planned changes up to that time to the DCD and engineering reports, and identified Tier 2* information. As part of this response, the ESBWR design certification applicant identified specific portions of the DCD to be Tier 2* information and also identified NEDE-33312P, NEDE-33313P, and NEDE-33408P as Tier 2* in their entirety. The NRC staff finds designation of the specified portions of the DCD and the complete NEDE-33312P, NEDE-33313P, and NEDE-33408P as Tier 2* to be acceptable. The staff has confirmed that Revision 10 to the DCD includes the planned designation of Tier 2* material, including NEDE-33312P, NEDE-33313P, and NEDE-33408P. Therefore, RAIs 3.9-292 and 292 S03 are resolved.

3.9.5.4 *Conclusions*

For the reasons set forth above, the NRC staff concludes that the DCD and engineering reports incorporated by reference provides sufficient information to support the adequacy of the design basis for the ESBWR reactor vessel CS structure and internal structures (reactor internals). Accordingly, the staff concludes that DCD Tier 2, Revision 10, Section 3.9.5 and the design process for reactor internals is acceptable and meets the requirements of GDC 1, 2, 4, and 10; 10 CFR 50.55a; and 10 CFR Part 52. This conclusion is based on the following findings:

1) The DCD meets the requirements of GDC 1, 10 CFR 50.55a, and 10 CFR Part 52 because the reactor internals are designed to quality standards commensurate with the importance of the safety functions performed. The design procedures and criteria for the safety-related reactor internals comply with the provisions of ASME BPV Code, Section III,

Subsection NG. The applicant has adequately addressed in their design process the potential adverse flow effects on the reactor internals, including the steam dryer, up to full licensed power conditions.

2) The DCD meets the requirements of GDC 2, 4, and 10 because the components important to safety are designed to withstand the effects of normal operation, maintenance, testing, and postulated accidents (including LOCAs) with sufficient margin to maintain their capability to perform safety functions. By incorporating the full provisions of ASME BPV Code, Section III, Subsection NG, for construction of the safety-related reactor CS structures, the applicant will apply a design process for the reactor internals with appropriate margin to ensure adequate structural support of the reactor core during normal operation, including the effects of anticipated operational occurrences.

   The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals provide reasonable assurance that during normal operating and postulated accident conditions the consequent deflections and stresses imposed on these structures and components will not exceed allowable stresses and deformation limits for the materials of construction. Limitation of stresses and deformations under such loading combinations is an acceptable basis for the design of these structures and components to withstand the most adverse loading events postulated to occur during service lifetime without loss of structural integrity or impairment of function.

3) The ESBWR steam dryer loading procedure is conservative and acceptable. The methodology is technically reasonable, in that it is based on actual on-dryer measurements, and includes bias errors and uncertainties computed from end-to-end benchmarking of an instrumented similar dryer in the GGNS plant. The bias errors and uncertainties are supplemented with an additional MASR safety factor of 2.0. The structural analysis of the ESBWR steam dryer will satisfy the provisions of the ASME BPV Code, Section III, Subsection NG, with justified exceptions. Finally, all ESBWR dryers will be treated as prototypes and instrumented for monitoring during plant power ascension. The NRC staff therefore concludes that the GEH steam dryer design approach, analysis methodology, and verification procedure will provide adequate protection to the public health and safety insofar as the ESBWR steam dryer will not experience fatigue cracking and not generate loose parts in ESBWR nuclear power plants.

4) The ESBWR design certification applicant specified the classification of the reactor internals as a prototype, in accordance with the guidance of RG 1.20, with a milestone for submitting the vibration assessment program commensurate with a prototype classification, including instrumentation and measurement procedures, inspection procedures, and correlation with analytical results. DCD Tier 2, Section 3.9.9, includes these provisions in COL Information Item 3.9.9-1-A.

8.0 ELECTRIC POWER

8.2 Offsite Power System

8.2.1 Regulatory Criteria

The following regulatory requirements provide the basis for the acceptance criteria for the staff's review:

- GDC 17, as it relates to the preferred power system (1) capacity and capability to permit functioning of SSCs important to safety, (2) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies, (3) physical independence, and (4) availability.

8.2.2 Summary of Technical Information

GEH's response to RAI 8.1-22, dated July 30, 2013, provided design details on how to detect a single-phase open phase condition (with and without a ground) on the high voltage side of a transformer connecting GDC 17 offsite power circuits to the transmission system (i.e., concerns cited in NRC Bulletin 2012-01) and initiate an alarm in the main control room (MCR). In addition, a supplemental response dated August 28, 2013 (supplement 1) modified the original DCD markups, and a second supplemental response dated November 20, 2013 (supplement 2) further modified the DCD markups to clarify that the monitoring of transformers would detect and alarm for open circuits in one or more phases.

As a result, the following information was added:

- In DCD Tier 1, Revision 10, Section 2.13: Design descriptions for unit auxiliary transformer (UAT) and reserve auxiliary transformer (RAT) monitoring systems.

- In DCD Tier 1, Revision 10, Table 2.13.1-2: ITAAC item 14a (Analysis) and 14b (Testing).

- In DCD Tier 1, Revision 10, Table 1C-2: Bulletin 2012-01 was considered as a part of operating experience review.

- In DCD Tier 2, Revision 10, Section 8.2.1.2.2: design details on detecting offsite circuit open phase conditions and initiating an alarm in the MCR upon detection.

8.2.3 Staff Evaluation

GDC 17 requires that "[a]n onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents." Therefore, the ESBWR electric power system (both offsite and onsite ac and dc power system) must meet the above requirements, and needs to address the design vulnerability identified in Bulletin 2012-01 to permit functioning of structures, systems, and components important to safety.

The staff determined that passive reactor designs should provide automatic detection of an offsite power system open phase circuit condition with and without a high impedance ground fault condition on the high voltage side of the main power transformer under all loading and operating configurations. In addition, an alarm should be provided in the MCR so that operators may take manual action if the standby diesel generators or the second offsite power line is not automatically connected to the plant investment protection (PIP) buses. This ensures that ac power, with adequate capacity and capability, is available to the important to safety equipment, including safety-related battery chargers, to meet their intended safety function in accordance with GDC 17 requirements.

The staff reviewed GEH RAI responses dated July 30, 2013, August 28, 2013, and November 20, 2013, that state open phase conditions (i.e., whether one, two, or three phases), with and without accompanying ground faults, on the high voltage side of GDC 17 credited offsite power can be detected and alarmed in the MCR. In its responses, GEH stated that the ESBWR design includes plans and requirements for detection and alarms of open phase conditions. GEH further elaborated that the plant design monitors and alarms all three phases of offsite and onsite power for abnormal voltages by using digital protective relays in the distributed control and information system (DCIS). To ensure GEH's compliance with GDC 17 with respect to the open phase conditions discussed in Bulletin 2012, the applicant added the following in DCD, Revision 10:

> <u>Tier 1, Section 2.13.1</u>
>
> (14) a. Monitoring of the normal and alternate power feeds on the high voltage side of the Unit Auxiliary Transformer (UAT) and Reserve Auxiliary Transformer (RAT) using the potential and current transformers of the digital protective relaying used for transformer protection is provided to detect open phase conditions, whether one, two, or three phases, with or without accompanying ground faults.
>
> (14) b. All three phases of all the UAT or RAT shall be monitored for undervoltage, open phase, and ground faults by the specific transformer protective relay. When an undervoltage, open phase or ground fault is detected in any combination of one, two or three phases by the designated UAT or RAT protective relay, the protective relay shall send an alarm via the DCIS alarm system to the MCR.
>
> <u>Tier 1, Table 2.13.1</u>
>
> ITAAC 2, item 14a - an analysis to determine the relay location and set points and item 14b - a test to verify proper functionality.
>
> <u>Tier 2, Section 8.2.1.2.2</u>
>
> Section 8.2.1.2.2 provided details on how the ESBWR will monitor the normal and alternate preferred power supply feeds using the digital protective relays for transformer protection, including detection of open phase conditions, whether one, two, or three phases, with or without a ground fault.

The open phase conditions for passive plants (like AP1000 and ESBWR) does not affect operation of safety related SSCs. However, detection of an open phase condition (i.e., one, two, or three phases) and alarming in the MCR is necessary to address the design vulnerability identified in Bulletin 2012-01 and thus permit the functioning of important to safety equipment in accordance with the requirements of GDC 17. According to the revised design details provided in Section 8.2.1.2.2, the potential and current transformers on the high voltage side of the UAT and RAT used by the digital protective relays will detect open phase conditions, with or without a ground fault. Upon detection of an open circuit condition, the protective relay scheme initiates an alarm in the MCR through the DCIS monitoring system. Operator manual actions are then taken to connect the standby diesel generators or the second offsite power source. In addition, the applicant stated that operator actions are addressed in procedures and described in DCD Tier 2, Revision 10, Section 13.5. DCD Tier 1, Revision 10, Section 2.13.1, ITAAC, provides the acceptance criteria for an analysis to verify site-specific relay set points for detection of open phase conditions and tests to verify the functionality of the protective equipment, as installed.

Based on the information discussed above, the staff finds that the ESBWR design complies with the requirements set forth in GDC 17 for having two offsite power circuits and two ac power sources, and conforms to the guidance in Bulletin 2012-01. In addition, the offsite circuits are monitored and alarmed in the MCR to detect open phase conditions, so that operator manual action can make power available for the functioning of structures, systems, and components important to safety. Also, operator actions are addressed in procedures and described in the DCD Tier 2, Revision 10, Section 13.5.

8.2.4 Conclusions

As set forth above, the staff reviewed the features that can detect and send an alarm for open phase conditions to minimize the probability of losing electric power from the offsite power circuit to the plant electrical system. Based on the information discussed above, the staff concludes that the ESBWR design includes features that meet the requirements in GDC 17 in regard to open phase faults on the offsite power system, with or without accompanying ground faults. Therefore, the staff concludes that no design vulnerability identified in Bulletin 2012-01 exists in the ESBWR electric power system.

9.0 AUXILIARY SYSTEMS

9.1.2 New and Spent Fuel Storage

9.1.2.1 Regulatory Criteria

The staff performed its review of the applicant's seismic design of fuel assemblies to be stored in ESBWR FSRs in accordance with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (hereafter referred to as the SRP), Section 3.8.4, "Other Seismic Category I Structures," Appendix D, "Guidance on Spent Fuel Pool Racks," Revision 2, issued March 2007. The staff reviewed the analysis to determine whether the design is in compliance with GDC 2 of Appendix A to 10 CFR Part 50 as it relates to SSCs important to safety being designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of earthquakes.

9.1.2.2 Summary of Technical Information

In DCD Section 9.1.2.4, the applicant indicates that the FSRs are designed to protect the fuel assemblies from excessive physical damage. DCD Section 9.1.7 lists as a reference NEDO-33373, which analyzes the seismic response of the rack system, and which the NRC approved as described in a Safety Evaluation Report (SER) dated October 20, 2010. Since the FSR design is adequate, as set forth in the October 2010 SER, the fuel assemblies are protected from excessive damage in the event of an SSE. To further confirm the structural integrity of the fuel in the FSRs in the event of an SSE, the NRC staff conducted an audit on August 5, 2011, and continued it on September 8, 2011, as documented in an audit report. The purpose of the audit was for the staff to review additional information provided by the GEH to confirm that the consequent loads on the fuel assembly during seismic excitation would not lead to fuel damage. The staff audited the DCD calculations used to confirm that fuel assembly integrity in the spent fuel pool and the buffer pool is maintained during a seismic event. During the audit, the staff reviewed additional information provided by GEH to confirm that the consequent loads on the fuel assembly that result from the design-basis seismic event would not lead to fuel damage. Subsequent to the September 8, 2011 audit, GEH submitted the GEH SFSQ Report which summarizes the evaluation of the seismic adequacy of the fuel assemblies in the FSRs.

GEH indicated in the SFSQ Report that given the similar boundary conditions to which the fuel assemblies are subjected in both the reactor core and in the FSRs for both the spent fuel and buffer pools, it can be demonstrated that the seismic response of the fuel assemblies in the FSRs as a result of the ESBWR seismic design-basis loads is bounded by the seismic response to which the in-core fuel assemblies were designed. GEH believes that this approach is appropriate because the seismic adequacy of the in-core fuel assemblies was established in an NRC staff-approved GE licensing topical report, namely, NEDE-21175-3-P-A, "BWR Fuel Assembly Evaluation of Combined Safe Shutdown (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (Amendment No. 3)," issued November 20, 1984, which is proprietary. NEDE-21175-3-P-A demonstrates fuel integrity for various GEH fuel designs for fuel in the reactor core under combined loss-of-coolant accident and safe-shutdown earthquake loads. GEH concludes that there is assurance of the seismic adequacy of the fuel assemblies in the FSRs if they are demonstrated to be subjected to combined loss-of-coolant accident and safe-shutdown earthquake loads lower than those to which fuel in the core are subjected.

According to the GEH SFSQ Report, the seismic response of the in-core fuel assemblies is limited by the channel plate in the horizontal direction and by the upper tie plate in the vertical direction. GEH indicates that for the horizontal response, the in-core fuel assemblies were modeled as pin-supported beams (simply supported beam model) in the seismic analysis as described in NEDE-21175-3-P-A. Therefore, the peak acceleration to induce the maximum bending moment at the midpoint of the channel occurs at the midpoint, which is the case for a simply supported beam.

GEH indicated that although the physical configuration, including boundary conditions for the fuel assemblies in the FSRs, closely resembles that of the in-core conditions, the seismic analysis of the FSRs modeled the fuel assemblies in the FSRs with gap elements to closely simulate their contacts with FSR cell plates, as described in NEDO-33373. Upon examining the results of the FSR seismic analysis, GEH concluded that the maximum seismic response in the horizontal direction corresponds to the peak horizontal acceleration occurring at the midpoint of the fuel assembly, a vibration mode similar to that of a simply supported beam. GEH concluded that this result is consistent with the dynamic characteristics analyzed for the in-core fuel assemblies for the horizontal motions. A comparison of the respective peak horizontal accelerations from both in-core and FSR seismic analyses indicates that the in-core fuel assemblies are subjected to the peak horizontal acceleration, which is substantially higher than that imposed on the fuel assemblies in the FSRs. Therefore, GEH concluded that the in-core fuel seismic response bounds that of the fuel assemblies in the FSRs for horizontal motions.

GEH also made a similar comparison for the vertical responses between the in-core and fuel assemblies in the FSRs, as follows: The vertical response of the fuel assemblies corresponds to the axial vibration mode for both in-core fuel and fuel assemblies in the FSRs. The resulting comparison indicates that the in-core fuel seismic response bounds that of fuel assemblies in the spent fuel pool with sufficient margin. However, the maximum vertical accelerations calculated for the buffer pool exceed the maximum vertical acceleration for which the in-core fuels were designed. This vertical acceleration value for the buffer pool is larger when compared with the spent fuel pool because the buffer pool is located at a higher elevation in the reactor building and, therefore, is subject to higher seismic motions than the spent fuel pool, which is located at grade. The in-core fuels were seismically designed for the core temperature of 288 degrees Celsius. If the temperature is lower, the material (metals) strength will increase, therefore increasing the seismic capacity of the fuel assemblies, which is the case for the buffer pool because the abnormal temperature for the buffer pool is limited to 60 degrees Celsius. By crediting the increased material strength in the fuel channel box because of the lower water temperatures seen in the buffer pool, the increased seismic strength of the fuels sufficiently compensates for the exceedance in the vertical acceleration demands in the buffer pool. Therefore, GEH's evaluation confirmed the seismic adequacy of fuel assemblies in the FSRs in the buffer pool.

GEH also investigated the rattling effect on the fuel assemblies in the FSRs as a result of the opening and closing of gaps between fuel assemblies and FSR cell plates induced by horizontal seismic motions. GEH reasoned that since the natural frequency of the fuel assemblies is calculated at about 7–8 hertz, a relatively LF compared to the frequency of about 33 hertz for the peak horizontal acceleration input motion responsible for the rattling effect, the fuel rattling inside the FSR cells has a negligible effect on the fuel's seismic adequacy.

Finally, the GEH SFSQ Report indicates that the temperature effect on the fuel material properties would increase the seismic capacity for the fuel assemblies in the FSRs in comparison with the properties of in-core fuels. GEH indicates that because the in-core fuels are subjected to much higher temperatures than the fuel assemblies in the FSRs, the material strength for the fuel assemblies in the FSRs increases. GEH also recognized that irradiated fuels at the lower temperature would have an increased material brittleness. GEH states, however, that since the seismic qualification of the in-core fuels maintains the fuel stress level at less than 70 percent of the material's ultimate strength, the stress on the fuel materials for the channel plate and upper tie plate is within the elastic limit of the material. Therefore, GEH concludes that the irradiation has no effect on the seismic performance of the fuel assemblies in the FSRs.

Based on the above discussion, the GEH SFSQ Report concludes that the fuel assemblies in the FSRs in both the spent fuel pool and the buffer pool maintain structural integrity when subjected to the design-basis seismic loads.

9.1.2.3 Staff Evaluation

In its safety evaluation of NEDO-33373, which the DCD references in Section 9.1.7, the NRC staff concluded that the FSRs in both the spent fuel pool and the buffer pool are structurally adequate to withstand the design-basis seismic loads established for the ESBWR standard design. The applicant has provided additional information in a supplemental evaluation to demonstrate that the fuel assemblies stored inside the FSRs also are capable of withstanding the design-basis seismic loads. This section of the report provides the staff's review and evaluation to determine the technical adequacy of the applicant's evaluation. The staff carried out its technical review in accordance with Appendix D to SRP Section 3.8.4 to ensure compliance with GDC 2.

For reasons to be discussed below, the applicant demonstrated in the GEH SFSQ Report that the seismic response of the fuel assemblies in the FSRs is bounded in part; by the seismic demands for which the in-core fuel assemblies were designed, as follows:

- The maximum horizontal seismic demand of fuel assemblies in the FSRs corresponds to a vibration mode that closely resembles that of simply supported beams and therefore is consistent with the vibration mode to which the in-core fuel assembly was seismically designed.

- The seismic response of the fuel assemblies in the spent fuel pool is bounded by the seismic response of the in-core fuel assemblies.

- The seismic response of fuel assemblies in the FSRs in the buffer pool is bounded by the seismic response of the in-core fuel assemblies in the horizontal direction.

- Although the vertical seismic load on the fuel assemblies in the FSRs in the buffer pool exceeds that on the vertical load on the in-core fuel assemblies, the increased material strength of the fuel assemblies in the FSRs in the buffer pool over that of the in-core fuel assemblies, as a result of the lower temperature present in the buffer pool, ensures that the fuel assemblies in the FSRs in the buffer pool are adequate for the seismic loads.

- Rattling because of the gaps in the FSR cell plates and the top of the fuel assemblies has minimal impact on the structural integrity of the fuel structures.

- The material strength of fuel assemblies in the FSRs is increased, compared to that of the in-core fuel assemblies, because the temperatures in the spent fuel and buffer pools containing the FSRs are substantially lower than the in-core temperature.

Based on this information, the applicant concluded that the fuel assemblies in the FSRs in both the spent fuel pool and the buffer pool maintain structural integrity when subject to the design-basis seismic loads, and the fuel assemblies in the FSRs are structurally adequate to withstand the ESBWR design-basis seismic loads. The staff's evaluation of this information is provided below.

Horizontal Seismic Response of Fuel Assemblies in the Fuel Storage Racks

The GEH fuel is structurally enclosed within a channel plate and capped with lower and upper tie plates on the ends. The channel plate provides lateral stiffness for the fuel. Both the channel and tie plates are made of Zircaloy material. The fuel assemblies in the FSRs rest on the FSR cell base plate, which is shaped to prevent lateral movements but allows the fuel assemblies to rotate at the lower tie plate against the FSR cell base plate. Therefore, the staff concludes that the FSR fuel assemblies are supported at the base as a pinned constraint identical to the base support for the in-core fuel assemblies. However, a gap of about 10 millimeters is present between the upper tie plate and the FSR cell plate, which is larger than the gap identified for the in-core fuel assembly. To demonstrate that the maximum horizontal seismic demand in the fuel assembly occurs if the gap at the top of the fuel assembly is closed and behaves as a pinned support, GEH used the results of the seismic analysis for the FSRs, as documented in NEDO-33373, to identify the maximum induced horizontal accelerations along the fuel assembly and the corresponding response profiles. Although the highest acceleration response occurs at the top, the corresponding structural response profile (in terms of lateral deflections) indicates that the fuel assembly moves in a way that is similar to a rigid body; therefore, the fuel would be stressed less than if it deformed. The next highest acceleration examined occurs at the midpoint of the fuel assembly, and the corresponding response profile shows the largest induced deflection, which closely resembles the vibration of a simply supported beam. This means that the upper tie plate comes in contact with the FSR cell plate and remains in that position when the maximum horizontal seismic demand is induced in the fuel assembly. This is consistent with the constraint condition analyzed for the in-core fuel.

Based on the above discussion, the staff concludes that fuel assemblies in the FSRs, analyzed as simply supported beams when responding to horizontal seismic loads, are adequately characterized with the data extracted from the staff-approved NEDO-33373 and therefore are acceptable.

Bounding of Seismic Demand of Fuel Assemblies in the Fuel Storage Racks by In-Core Fuel Assemblies

The in-core fuel was seismically designed (NEDE-21175-3-P-A) for the maximum horizontal acceleration at the midpoint of the channel plate. The GEH SFSQ Report compared maximum accelerations for the fuel assemblies in the FSRs in both the spent fuel and buffer pools with those loads for which the in-core fuel assemblies were designed. This comparison indicates that the horizontal seismic load on the fuel assemblies in the FSRs is substantially less than the seismic demand used to design the in-core fuel assemblies. Similar comparisons of the vertical maximum accelerations between the spent fuel pool FSRs and the in-core fuel assemblies also indicate substantial margin of the seismic demand on the fuel assemblies in the FSRs in the spent fuel pool as compared to the seismic demands for which the in-core fuel assemblies are designed.

The staff verified that the seismic design demands of the in-core fuel bound the seismic demands of the fuel assemblies in the FSRs in the spent fuel pool. Since NEDE-21175-3-P-A established the seismic adequacy of in-core fuel assemblies, and the applicant has demonstrated that the loading on the assemblies in the FSRs is less than the loading on the assemblies in the core, the staff therefore concludes that the fuel assemblies in the FSRs in the spent fuel pool are also seismically adequate.

For the buffer pool, the staff verified that the fuel seismic demands are bounded by the in-core fuel seismic demand design criteria in the horizontal direction. However, the vertical response of the fuel in the buffer pool exceeds the vertical acceleration for which the in-core fuel was designed because the buffer pool is located on the upper floor elevation of the reactor building and is thereby subject to higher seismic motion. This increased vertical loading on the fuel in the buffer pool, however, is more than compensated for by the increased material strength of the fuel under the cooler conditions found in the buffer pool as compared with the in-core conditions. Accordingly, the staff concludes that the fuel assemblies in the buffer pool are seismically adequate.

<u>Rattling Effect on Fuel Assemblies in the Fuel Storage Racks</u>

The applicant calculated the natural frequency of the ESBWR fuel assemblies to be in a range of about 7-8 hertz. In NEDO-33373, the applicant also used the result of the seismic analysis for the FSRs to identify the maximum acceleration time history response responsible for the rattling at the top of the fuel assemblies in the FSRs. This maximum acceleration corresponds to single narrow pulse with a frequency of about 33 hertz. The fuel assemblies have a natural frequency of 7-8 hertz and therefore cannot be excited by this high-frequency rattling pulse input. This is further substantiated by the examination of the corresponding response profile provided in the GEH SFSQ Report. This response profile indicates a deformation profile for the fuel assembly in response to the rattling pulse input motion that was close to that of a rigid body. Accordingly, the staff concludes that the rattling effect does not structurally damage the fuel assemblies.

<u>Material Strength of Fuel Assemblies in the Fuel Storage Racks</u>

Both the channel and the tie plates are constructed of Zircaloy material. The Zircaloy material strength decreases as the temperature increases. Since the in-core temperature is much higher than that of the spent fuel and buffer pools, the Zircaloy material exhibits higher material strength in the spent fuel and buffer pools. Therefore, the fuel assemblies in the FSRs in the spent fuel and buffer pools will have higher material strength than the in-core fuel assemblies. The staff concludes that it is appropriate to determine the seismic structural capacity of the fuel assemblies based on the effect of temperature on the material strength.

However, when the irradiated fuel is removed from the core and its temperature is subsequently reduced, the material exhibits higher brittleness (i.e., reduced ductility). The reduced material ductility would impact the structural performance of the material if the stress level is near the material yield point or exceeds that point. According to GEH SFSQ Report, the GEH in-core fuel is seismically designed to maintain a stress level less than 70 percent of the material’s ultimate strength, which effectively keeps the stress level in the fuel assemblies within the elastic limit of the material. Additionally, the increased brittleness caused by the irradiation would only be noticeable under shear stress. The forces experienced by the channel box in the FSRs in the spent fuel and buffer pools are compressive in nature. Therefore, the embrittlement caused by irradiation does not impact the seismic performance of the fuel assemblies in the FSRs in the spent fuel and buffer pools, and the maintenance of fuel integrity is demonstrated because the maximum loads are below the calculated acceleration limits. On this basis, the staff concludes that the thermal mechanical properties used by the applicant in demonstrating the seismic capacity of the fuel assemblies in the FSRs are adequate and therefore acceptable.

9.1.2.4 Conclusions

As set forth in the October 2010 SER on NEDO-33373, the design of the FSRs is acceptable. Therefore, the NRC staff concludes that the FSRs are adequate to protect the fuel assemblies stored in them in the event of an SSE. Further, the staff has determined that information in the GEH SFSQ Report provided by the applicant on September 23, 2011, confirms the finding that the fuel assemblies housed in FSRs in the spent fuel and buffer pools are in compliance with GDC 2 of Appendix A to 10 CFR Part 50. This confirmation is based on the following:

- The applicant, through a combined qualitative and quantitative evaluation (described in Section 9.1.2.2 of this report), has demonstrated that fuel assemblies in the FSRs in the spent fuel and buffer pools are structurally adequate to withstand the loads resulting from the SSE that has been established as the CSDRS for the ESBWR certified design.

- The applicant has demonstrated that the thermal-mechanical properties increase the seismic performance of fuel assemblies in the FSRs in the buffer pool, as opposed to the in-core fuels, whose seismic design was the baseline for the evaluation, thereby ensuring the seismic adequacy of the fuel assemblies in the buffer pool.

- The applicant has demonstrated that the rattling effect, due to seismic interaction, would not adversely affect the seismic performance of fuel assemblies in the spent fuel and buffer pools.

Therefore, the staff concludes that the applicant confirmed that the fuel assemblies in the SFP and buffer pool are structurally adequate to withstand the loads transmitted to the fuel during an SSE event, and therefore acceptable.

14.0 VERIFICATION PROGRAMS

14.3 Inspections, Tests, Analyses, and Acceptance Criteria

This supplemental FSER documents the NRC staff's review of changes made to the DCD Tier 1 after March 9, 2011. The previous DCD Tier 1 definition of "ASME Code requirements" did not specifically include alternatives to the Code pursuant to 10 CFR 50.55a(a)(3). Because the definition was not explicit in this regard, a concern was raised regarding whether a COL holder referencing the ESBWR DCD might need an exemption to use an alternative to the Code under 10 CFR 50.55a(a)(3). The NRC has previously stated explicitly that an exemption would not be needed in these cases (as noted in the statements of consideration on the revision to 10 CFR Part 52, 72 *Federal Register* 49438). To remove all doubt with respect to this concern, on August 28, 2013 GEH submitted a revision to the DCD Tier 1, Section 1.1.1 definition to state that "ASME Code" means the ASME Code or any NRC-approved alternative under 50.55a(a)(3). Because this change does not affect previous NRC safety findings in the FSER or change ESBWR's compliance with Code requirements, the staff finds these changes to the definition of ASME Code acceptable. The NRC staff confirmed that the proposed wording was incorporated in DCD Revision 10 and considers this issue resolved.

14.3.3 Piping Systems and Components

Following the issuance of the FSER on March 9, 2011, while confirming the inspectability and consistency of design certification ITAAC, a concern was raised that ESBWR ASME Code component design verification ITAAC, as written in Revision 9 of the DCD, might be viewed as requiring design verification of as-designed ASME Code components, rather than as-built ASME Code components, which is the underlying purpose of these ITAAC. Verifying interim ASME Code design reports at the design stage would result in an unnecessary regulatory burden with no benefit to safety, and the underlying purpose of the ITAAC in question was not to require such an activity. On August 28, 2013, GEH submitted markups to ten ITAAC sets that verify ASME Code components to make explicit that the activities needed to satisfy the ITAAC are performed at the as-built stage. The changes to each of the ITAAC sets involves deleting redundant design description and design verification ITAAC, and rewording the Acceptance Criteria of the as-built reconciliation ITAAC to address compliance of as-built component design to the requirements of ASME Code Section III. The net effect of this change combines the commitments of the design verification ITAAC and the as-built reconciliation ITAAC into a single equivalent ITAAC. The resulting ITAAC Acceptance Criteria are clear that the design verification and as-built reconciliation are to be performed using the ASME Code Design Reports, including reconciliation reports, at the as-built stage. This clarification ensures efficient ITAAC closure and reduces potential confusion. Because the resulting requirements imposed by the single revised ITAAC are equivalent to the requirements imposed by the two previous ITAAC, this change does not affect previous NRC safety findings in the FSER, thus the staff finds these changes to the ASME Code component ITAAC acceptable. The NRC staff confirmed that the proposed wording was incorporated in DCD Revision 10 and considers this issue resolved.

14.3.6 Electrical Systems (GEH information related to Bulletin 2012-01)

For DCD Tier 1, Revision 10, ITAAC information for the design vulnerability identified in NRC Bulletin 2012-01 is discussed in Section14.3.6. In RAI 8.1-22, the staff requested the applicant to provide ITAAC information (Chapter 2, Sections 2.13 and 4.2) in accordance with 10 CFR 52.47, to determine how it meets the electrical system vulnerability discussed in NRC Bulletin 2012-01 with respect to satisfying GDC 17 requirements. The applicant added the following in DCD Tier 1, Revision 10:

> Section 2.13.1
>
> (14) a. Monitoring of the normal and alternate power feeds on the high voltage side of the UAT and RAT using the potential and current transformers of the digital protective relaying used for transformer protection is provided to detect open phase conditions, whether one, two, or three phases, with or without accompanying ground faults.
>
> (14) b. All three phases of all the UAT or RAT shall be monitored for undervoltage, open phase, and ground faults by the specific transformer protective relay. When an undervoltage, open phase or ground fault is detected in any combination of one, two or three phases by the designated UAT or RAT protective relay, the protective relay shall send an alarm via the DCIS alarm system to the Main Control Room.
>
> Table 2.13.1-2
>
> ITAAC 2, item 14a - an analysis to determine the relay location and set points and item 14b - a test to verify proper functionality.

The staff has reviewed the ITAAC tables in Section 2.13.1. Based on the information discussed above, the staff concludes the design description and design commitment to be consistent with acceptance criteria established in ITAAC table.

16.0 TECHNICAL SPECIFICATIONS

16.1 Introduction and Regulatory Criteria

The NRC staff received a letter dated August 28, 2013 from GE Hitachi Nuclear Energy that referenced a table listing proposed minor editorial changes to the generic technical specifications and associated bases in Chapters 16 and 16B of the ESBWR DCD Tier 2, Revision 10. The NRC staff verified that the proposed changes were correctly implemented in Revision 10 that was submitted on December 11, 2013. The staff finds these changes acceptable because they are strictly editorial in nature and do not affect the technical content of the previously approved generic technical specifications and associated bases.

Table 16.1 – Changes to ESBWR DCD Tier 2, Chapters 16 and 16B, Revision 10

Item	Location	Description of Editorial Change
1 .	Entire Chapter (16)	Header changed from “Rev. 09” to “Rev. 10”.
2 .	Table of Contents (TOC)-Chapter 16	Updated TOC information consistent with changes to the associated LCOs.
3 .	Page 3.3.7.1-4	Corrected horizontal alignment of Functions 2, 3, and 4 in Table 3.3.7.1-1.
4 .	Page 3.3.7.2-1	Corrected horizontal alignment of Completion Times for Required Actions B.1.2, B.1.3, and B.2.
5 .	Entire Chapter (16B)	Header changed from “Rev. 09” to “Rev. 10”.
6 .	Table of Contents (TOC)-Chapter 16B	Updated TOC information consistent with changes to the associated LCOs.
7 .	Page B 3.1.2-2	Replaced "satisfies" with "satisfy" in the last sentence of the Applicable Safety Analyses section.
8 .	Page B 3.1.3-3	In the Actions section, changed “A.1, A.2, A.3, and A.4” to “[A.1, A.2, A.3, and A.4][A.1, A.2, and A.3]” and added left margin annotation (LMA) for “COL 16.0-1-A, 3.1.3-1.”
9 .	Page B 3.1.3-4	In the last paragraph, changed “A.2” to “A.[3][2]” and added left margin annotation (LMA) for “COL 16.0-1-A, 3.1.3-1.”
10 .	Page B 3.1.3-5	In the second full paragraph, changed first occurrence of “A.3” to “A.[3][2],” second occurrence to “A.[4][3],” and added left margin annotation (LMA) for “COL 16.0-1-A, 3.1.3-1.”
11 .	Page B 3.1.5-1	Replaced "satisfies" with "satisfy" in the last sentence of the Applicable Safety Analyses section.
12 .	Page B 3.1.7-9	In the first paragraph, third sentence, changed “…ensure that that…” to “…ensure that…”

Item	Location	Description of Editorial Change
13.	Page B 3.3.1.3-4	In the References section, aligned “None.” with the Surveillance Requirements section indentation.
14.	Page B 3.3.1.5-1	In the second sentence of the Applicable Safety Analyses section, removed the line break between “trip” and “signal.”
15.	Page B 3.3.1.5-5	Removed indentation of “REQUIREMENTS” in the flush-left title of the Surveillance Requirements section.
16.	Page B 3.3.1.6-6	In the References section, aligned “None.” with the Surveillance Requirements section indentation.
17.	Page B 3.3.5.2-1	Inserted a comma after “LCO” in the section heading.
18.	Page B 3.3.5.3-5	Inserted a comma after “LCO” in the section heading.
19.	Page B 3.3.5.4-1	Inserted a comma after “LCO” in the section heading.
20.	Page B 3.3.6.3-8	In the middle of the first paragraph, inserted a line break before “Reactor Vessel Water Level…”
21.	Page B 3.3.7.1-6	Inserted a comma after “LCO” in the section heading.
22.	Page B 3.3.7.2-1	Inserted a comma after “LCO” in the section heading.
23.	Page B 3.3.8.1-4	Changed “AND” to “and” in the section heading.
24.	Page B 3.3.8.1-5	Changed “AND” to “and” in the section heading.
25.	Page B 3.4.4-1	In the fifth paragraph, first sentence, of the Background section, inserted a space after “CFR.”
26.	Page B 3.5.3-5	In the last sentence, deleted space before the period.
27.	Page B 3.5.5-1	In the first sentence in the Applicable Safety Analysis section, changed “shutdown not” to “shutdown is not”.
28.	Page B 3.6.2.2-2	Inserted “interval” at the end of the last sentence in the Actions section.
29.	Page B 3.7.6-1	In the last sentence, removed the page break in the middle of the sentence and relocated text from the top of the following page.
30.	Page B 3.7.6-2	Moved first 1.5 lines of text to the bottom of the previous page.
31.	Page B 3.7.6-2	Replaced “satisfies” with “satisfy” in the last sentence of the Applicable Safety Analyses section.
32.	Page B 3.8.1-5	Removed erroneous “).” from the first paragraph for Required Action B.1 in the Actions section.
33.	Page B 3.9.1-2	Replaced “satisfies” with “satisfy” in the last sentence of the Applicable Safety Analyses section.

3 4 .	Page B 3.9.2-2	Inserted “The” at the beginning of the sentence in the Applicable Safety Analyses section.

23.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS) has completed its review of the supplemental FSER for certification of the ESBWR standard design. In particular, the ACRS Subcommittee reviewed the staff's evaluation of the revised analysis procedure for the structural and functional integrity of the ESBWR steam dryer and met with representatives from the NRC staff on March 5, 2014, to discuss the ESBWR steam dryer design.

During the 613th meeting of the ACRS, the full Committee met with representatives from the NRC staff on April 10, 2014, to discuss the ESBWR steam dryer design. The full Committee considered the ESBWR steam dryer design, and issued its letter report to the NRC Chairman on April 17, 2014. That letter report is included as Appendix F to this report.

In its letter report dated April 17, 2014, the ACRS concluded that the ESBWR steam dryer design is adequate and there is reasonable assurance that it can be built and operated without undue risk to the health and safety of the public.

24.0 CONCLUSIONS

The NRC staff has reviewed GEH's changes to the ESBWR design documentation and other principal review matters (see Section 1.5 of this report). On the basis of the evaluation described in the ESBWR FSER and this report, the NRC staff concludes that the ESBWR design documentation (up to and including Revision 10 to the DCD) is acceptable and that GEH's application for design certification meets the requirements of 10 CFR Part 52, Subpart B, that are applicable and technically relevant to the ESBWR standard plant design.

On the same basis, the staff also concludes that issuance of a revised standard design approval, in accordance with 10 CFR Part 52, Subpart E, will not be inimical to either the common defense and security or the health and safety of the public. A revised standard design approval, issued on the basis of the FSER and this report, does not constitute a commitment to issue a permit, design certification, or license, and does not in any way affect the authority of the Commission, the Atomic Safety and Licensing Board Panel, and other presiding officers in any proceeding under 10 CFR Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders."

APPENDIX A. Chronology

This appendix of the supplemental final safety analysis report (FSER) contains a chronological listing of routine licensing correspondence between the staff of the U.S. Nuclear Regulatory Commission (NRC) and GE – Hitachi Nuclear Energy (GEH) regarding the review of the ESBWR passive plant design under Project No. 717 and Docket No. 52-010. This supplement to the appendix lists the additional correspondence between the NRC and GEH during the time period between FSER issuance (March 2011) and completion of this supplemental FSER.

Revisions to ESBWR Design Control Document

Revision	Date
10	April 1, 2014
10	December 11, 2013

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14248A648	Transmittal of NEDO-33260 Revision 5, Related to ESBWR Design Certification Application - Chapter 17. 34 Page(s)	01-08-2010	"Letter", "Quality Assurance Program", "Report, Technical"	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14248A662	Transmittal of NEDO-33289 Revision 2, Related to ESBWR Design Certification Application - Chapter 17. 19 Page(s)	01-08-2010	"Letter", "Quality Assurance Program", "Report, Technical Report"	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14248A297	Transmittal of NEDO-33181 Revision 6, Related to ESBWR Design Certification Application - Chapter 17. 21 Page(s)	01-08-2010	"Letter", "Quality Assurance Program", "Report, Technical"	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML092570792	Request For Withholding Information From Public Disclosure (MFN-09-418). 6 Page(s)	25-Feb-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/NGE1	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML090130412	Request For Withholding Information From Public Disclosure (MFN 08-920, Supplement 3). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML090540811	Request For Withholding Information From Public Disclosure (MFN 09-067). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML091120425	Request For Withholding Information From Public Disclosure (MFN 09-129). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/NGE1	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092230524	Request For Withholding Information From Public Disclosure (MFN 09-358). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092230516	Request For Withholding Information From Public Disclosure (MFN 09-427). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/NGE1	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092860187	Request For Withholding Information From Public Disclosure (MFN 09-485). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092880018	Request For Withholding Information From Public Disclosure (MFN 09-520). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML100280008	Request For Withholding Information From Public Disclosure (MFN 09-770). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100540604	Request For Withholding Information From Public Disclosure (MFN 09-773, Revision 1). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100110062	Request For Withholding Information From Public Disclosure (MFN 09-775). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100620323	Request For Withholding Information From Public Disclosure (MFN 09-775, Revision 1). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML101760290	Request For Withholding Information From Public Disclosure (MFN 09-775, Revision 2). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100540998	Request For Withholding Information From Public Disclosure (MFN 09-789, Revision 1). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML100541225	Request For Withholding Information From Public Disclosure (MFN 09-790, Revision 1). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100270597	Request For Withholding Information From Public Disclosure (MFN 09-792). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100610248	Request For Withholding Information From Public Disclosure (MFN 09-792, Revision 1). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100560440	Request For Withholding Information From Public Disclosure (MFN 10-005). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100680498	Request For Withholding Information From Public Disclosure (MFN 10-043). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML103420335	Request For Withholding Information From Public Disclosure (MFN 10-267). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML110040232	Request For Withholding Information From Public Disclosure (MFN 10-356). 6 Page(s)	1-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML081430534	Request For Withholding Information From Public Disclosure (MFN 08-119). 7 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML081430522	Request For Withholding Information From Public Disclosure (MFN 08-169). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML090120883	Request For Withholding Information From Public Disclosure (MFN 8-169, Supplement 2). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML083450029	Request For Withholding Information From Public Disclosure (MFN 08-742). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML090070472	Request For Withholding Information From Public Disclosure (MFN 08-882). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML090120554	Request For Withholding Information From Public Disclosure (MFN 08-898). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML091470377	Request For Withholding Information From Public Disclosure (MFN 09-187). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML091890791	Request For Withholding Information From Public Disclosure (MFN 09-327). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092710376	Request For Withholding Information From Public Disclosure (MFN 09-503). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML093450784	Request For Withholding Information From Public Disclosure (MFN 09-531). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100110030	Request For Withholding Information From Public Disclosure (MFN 09-773). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100110034	Request For Withholding Information From Public Disclosure (MFN 09-778). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100110065	Request For Withholding Information From Public Disclosure (MFN 09-785). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100110297	Request For Withholding Information From Public Disclosure (MFN 09-788). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML100191964	Request For Withholding Information From Public Disclosure (MFN 09-789). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100191981	Request For Withholding Information From Public Disclosure (MFN 09-790). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100330036	Request For Withholding Information From Public Disclosure (MFN 09-803). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML101460330	Request For Withholding Information From Public Disclosure (MFN 10-141). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML103420338	Request For Withholding Information From Public Disclosure (MFN 10-285). 6 Page(s)	2-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092570777	Request For Withholding Information From Public Disclosure (MFN-09-411). 6 Page(s)	3-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092570850	Request For Withholding Information From Public Disclosure (MFN 09-454). 6 Page(s)	3-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML092710362	Request For Withholding Information From Public Disclosure (MFN 09-499). 6 Page(s)	3-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092710370	Request For Withholding Information From Public Disclosure (MFN 09-499, Revision 1). 6 Page(s)	3-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092710426	Request For Withholding Information From Public Disclosure (MFN 09-522). 6 Page(s)	3-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/NGE1	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092710498	Request For Withholding Information From Public Disclosure (MFN 09-545). 6 Page(s)	3-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092710519	Request For Withholding Information From Public Disclosure (MFN-09-550). 6 Page(s)	3-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100290652	Request For Withholding Information From Public Disclosure (MFN 09-595). 6 Page(s)	3-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML081430493	Request For Withholding Information From Public Disclosure (MFN 07-162, Supplement 1). 7 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML081510621	Request for Withholding Information from Public Disclosure (MFN-07-321, Supplement 1). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML081690326	Request for Withholding Information from Public Disclosure (MFN-07-603). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML081510649	Request for Withholding Information from Public Disclosure (MFN-08-168). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML083050588	Request For Withholding Information From Public Disclosure (MFN 08-647). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML083090898	Request For Withholding Information From Public Disclosure (MFN 08-661). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML083090893	Request For Withholding Information From Public Disclosure (MFN 08-672). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML090060903	Request for Withholding Information from Public Disclosure (MFN-08-912). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNR L/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML090400734	Request For Withholding Information From Public Disclosure (MFN 09-017). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML090410169	Request for Withholding Information from Public Disclosure (MFN-09-026). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML090760123	Request For Withholding Information From Public Disclosure (MFN 09-087). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML091120548	Request For Withholding Information From Public Disclosure (MFN 09-223). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML091590363	Request For Withholding Information From Public Disclosure (MFN-09-246). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML091610081	Request For Withholding Information From Public Disclosure (MFN 09-299). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML091880199	Request For Withholding Information From Public Disclosure (MFN 09-310). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML091890748	Request For Withholding Information From Public Disclosure (MFN 09-310, Supplement 1). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100050261	Letter Request for Withholding Information From Public Disclosure (MFN-09-330). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092180782	Request For Withholding Information From Public Disclosure (MFN 09-346). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092190769	Request For Withholding Information From Public Disclosure (MFN-09-402). 6 Page(s)	7-Mar-2011	Letter Proprietary Information Review	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML092860182	Request For Withholding Information From Public Disclosure (MFN-09-482). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML093450564	Request For Withholding Information From Public Disclosure (MFN-09-561). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML093450539	Request For Withholding Information From Public Disclosure (MFN-09-570). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML100320805	Request For Withholding Information From Public Disclosure (MFN 09-589). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100050286	Request For Withholding Information From Public Disclosure (MFN 09-714). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100050327	Request For Withholding Information From Public Disclosure (MFN 09-763). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100050388	Request For Withholding Information From Public Disclosure (MFN-09-764). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML090540823	Request For Withholding Information From Public Disclosure (MFN-09-073). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100540665	Request For Withholding Information From Public Disclosure (MFN 09-778, Revision 1). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100610031	Request For Withholding Information From Public Disclosure (MFN 10-029). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML100610032	Request For Withholding Information From Public Disclosure (MFN 10-030). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100620367	Request For Withholding Information From Public Disclosure (MFN-10-031). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100610036	Request For Withholding Information From Public Disclosure (MFN-10-032). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML100610360	Request For Withholding Information From Public Disclosure (MFN 10-033). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML101760300	Request For Withholding Information From Public Disclosure (MFN-10-160). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML103420368	Request For Withholding Information From Public Disclosure (MFN 10-298). 6 Page(s)	7-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML091880081	Request For Withholding Information From Public Disclosure (MFN 09-267). 6 Page(s)	9-Mar-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML103470431	Chapter 1 - Introduction and General Discussions. 34 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110030027	Chapter 2 - Site Characteristics. 62 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110040021	Chapter 3 - Design of Structures, Components, Equipment, and Systems. 53 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML103470435	Chapter 4 - Reactor. 101 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110030065	Chapter 5 - Reactor Coolant System and Connected Systems. 80 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML103470439	Chapter 6 - Engineered Safety Features. 101 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110030049	Chapter 7 - Instrumentation and Control Systems. 259 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML110030061	Chapter 8 - Electric Power. 47 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110100319	Chapter 9 - Auxiliary Systems. 179 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110050010	Chapter 10 - Steam and Power Conversion System. 47 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML103470442	Chapter 11 - Radioactive Waste Management. 62 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML103470446	Chapter 12 - Radiation Protection. 49 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML103470450	Chapter 13 - Conduct of Operations. 20 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110030062	Chapter 14 - Verification Programs. 130 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML110030032	Chapter 15 - Transient and Accident Analyses. 130 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110030064	Chapter 16 - Technical Specifications. 144 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110030024	Chapter 17 - Quality Assurance. 27 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110030033	Chapter 18 - Human Factors Engineering. 140 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110130034	Chapter 19 - Probabilistic Risk Assessment and Severe Accidents Evaluation. 85 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML103470578	Chapter 20 - Generic Issues. 41 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML103470582	Chapter 21 - Testing and Computer Code Evaluation. 74 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML103470589	Chapter 22 - Regulatory Treatment of Nonsafety Systems. 57 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML11045A000	Chapter 23 - Review by the Advisory Committee on Reactor Safeguards. 1 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML11045A001	Chapter 24 - Conclusions. 1 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110590697	Appendix A - Chronology. 714 Page(s)	9-Mar-2011	Final Safety Analysis Report (FSAR)	NRC/NRO		05200010
ML110590064	Appendix B - References. 101 Page(s)	9-Mar-2011	Final Safety Analysis Report (FSAR)	NRC/NRO		05200010
ML110530181	Appendix C - Acronyms. 17 Page(s)	9-Mar-2011	Final Safety Analysis Report (FSAR)	NRC/NRO		05200010
ML110530186	Appendix D - Principal Contributors. 10 Page(s)	9-Mar-2011	Final Safety Analysis Report (FSAR)	NRC/NRO		05200010

ML110450574	Appendix E - Index of NRC's Requests for Additional Information. 5 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML110030070	Appendix F - Report by the Advisory Committee on Reactor Safeguards. 12 Page(s)	9-Mar-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML110540310	Final Design Approval for Economic Simplified Boiling Water Reactor. 8 Page(s)	9-Mar-2011	Letter	NRC/NRO	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML110050215	Final Safety Evaluation Report For The Economic Simplified Boiling Water Reactor Design. 5 Page(s)	9-Mar-2011	Letter	NRC/NRO/DNRL	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML110680396	E-mail from D. Galvin to T. Enfinger re: Transmittal Letter Requesting GEH Proprietary Review of the Final SER. 1 Page(s)	9-Mar-2011	E-Mail	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML110540403	Federal Register Notice, Notice of Issuance of Final Design Approval Pursuant to Subpart E of 10 CFR Part 52, GE Hitachi Nuclear Energy Economic Simplified Boiling Water Reactor Standard Design. 2 Page(s)	9-Mar-2011	Federal Register Notice	NRC/NRO/DNRL		05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML110830156	Press Release-11-056: NRC Seeks Comment on Proposed Rule to Certify GE-Hitachi ESBWR Reactor Design. 2 Page(s)	24-Mar-2011	Press Release	NRC/OPA		05200010
ML110880315	2011/03/25-Comment (2) of Farouk Baxter on Proposed Rule PR-52 regarding ESBWR Design Certification. 1 Page(s)	25-Mar-2011	Rulemaking-Comment	- No Known Affiliation	NRC/SECY/RAS	05200010
ML110960626	SER Chapter 8.0 - Electric Power. 26 Page(s)	25-Apr-2011	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/BWR		05200010
ML11124A103	2011/05/02- Comment of Jerald Head on Behalf of GE-Hitachi Nuclear Energy Opposing Petition to Suspend All Pending Reactor Licensing Decisions & Related Rulemaking Decisions Pending Investigation of Lessons Learned from Fukushima Daiichi Nuclear 3 Page(s)	2-May-2011	Rulemaking-Comment	GE-Hitachi Nuclear Energy Americas, LLC	NRC/SECY/RAS	05200010
ML111040155	MELCOR Design Basis Accident Containment Audit Calculations for the Economic Simplified Boiling Water Reactor Plant (Final). 2 Page(s)	3-May-2011	Memoranda	NRC/NRO/DSRA/SBCV	NRC/NRO/DSRA/SBCV	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML11187A303	2011/05/11-Comment (4) of Anonymous on Proposed Rules PR 52 regarding ESBWR Design Certification. 2 Page(s)	11-May-2011	Rulemaking-Comment	- No Known Affiliation	NRC/SECY/RAS	05200010
ML11188A056	2011/05/11-Comment (62) of Anonymous on Proposed Rulemaking PR 52 regarding AP1000 Design Certification Amendment. 2 Page(s)	11-May-2011	Rulemaking-Comment	- No Known Affiliation	NRC/SECY/RAS	05200010
ML111710485	Enclosure 5, Industry Handout Regarding 4b Calculational Results ESBWR for 6/1/11 Public Workshop 37 Page(s)	1-Jun-2011	Slides and Viewgraphs	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DSR A	05200010
ML11158A088	2011/06/06-Comment (3) of Patricia T. Birnie, on Behalf of GE Stockholders' Alliance, on Proposed Rule PR-52 regarding ESBWR Design Certification Amendment. 1 Page(s)	6-Jun-2011	Rulemaking-Comment	GE Stockholders' Alliance	NRC/SECY/RAS	05200010
ML11195A027	Minutes of Advisory Committee on Reactor Safeguards ESBWR Subcommittee Meeting, May 26, 2011, Rockville, Maryland (OPEN). 52 Page(s)	14-Jul-2011	Meeting Minutes Memoranda Meeting Agenda Meeting Briefing Package/Handouts Slides and Viewgraphs	NRC/ACRS	NRC/ACRS	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML112150585	08/30/11 - Notice of Meeting with Industry to Discuss Proposed Revisions to the Enforcement Policy on Construction-Related Topics. 11 Page(s)	16-Aug-2011	Meeting Notice Meeting Agenda	NRC/NRO/DNRL/NRGA	NRC/NRO/DNRL/NRGA	05200006 05200010
ML112290967	Enclosure 6: Industry Handout Regarding Application of 50.69 to the ESBWR for August 9, 2011 Public Workshop. 21 Page(s)	17-Aug-2011	Slides and Viewgraphs Meeting Briefing Package/Handouts	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML112420161	MFN 11-193 - 10 CFR 50.46 Annual Report for the ESBWR Standard Plant Design. 3 Page(s)	30-Aug-2011	Annual Operating Report Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML112500453	09/08/2011-Audit of The Economic Simplified Boiling Water Reactor Design Certification Spent Fuel Pool and Buffer Pool Seismic Load Calculations. 4 Page(s)	8-Sep-2011	Memoranda Operating Plan	NRC/NRO/DNRL/BWR	NRC/NRO/DNRL/BWR	05200010
ML11269A093	Clarifications Requested by NRC Staff on Economic Simplified Boiling Water Reactor Fuel Design. 23 Page(s)	23-Sep-2011	Letter Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML11312A115	Transcript of the ACRS ESBWR Subcommittee Meeting, October 21, 2011 [OPEN] Pages 1-290. 290 Page(s)	21-Oct-2011	Meeting Transcript	NRC/ACRS		05200010
ML112901116	Memo: Summary of Audit of Seismic Evaluation of ESBWR Fuel in Spent Fuel and Buffer Pools. 5 Page(s)	7-Nov-2011	Memoranda	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML112860614	ESBWR Fuel Seismic Audit Summary. 5 Page(s)	7-Nov-2011	Audit Report	NRC/NRO/DNRL/BWR		05200010
ML112920258	Request for Withholding Information from Public Disclosure (MFN-11-204). 5 Page(s)	14-Nov-2011	Proprietary Information Review Letter	NRC/NRO/DNRL/BWR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML112500488	Economic Simplified Boiling Water Reactor Design Certification Rulemaking Schedule. 5 Page(s)	15-Nov-2011	Letter	NRC/NRO/DNRL	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML113120076	ACRS Memorandum - Final Rule - ESBWR Design Certification (RIN 3150-AI85). 6 Page(s)	22-Nov-2011	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12003A029	Transcript of ACRS on ESBWR for Fermi, Unit 3, R-COLA, Subcommittee Meeting, November 30, 2011, Pages 1-165. 287 Page(s)	30-Nov-2011	Meeting Transcript	NRC/ACRS		05200010 05200033
ML113430334	Supplemental Final Safety Evaluation Report For The Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Review. 1 Page(s)	12-Dec-2011	Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML11361A392	Submittal of Definitive ESBWR ITAAC Listing for COL Reference. 129 Page(s)	20-Dec-2011	Letter Graphics incl Charts and Tables	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML120170304	Economic Simplified Boiling Water Reactor Design Certification Rulemaking Schedule, Revision 2. 6 Page(s)	19-Jan-2012	Letter	NRC/NRO/DARR	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML120170309	ESBWR Technical Issues - Enclosure 1, Errors in Topical Reports Associated with Economic Simplified Boiling-Water Reactor Dryer Modeling. 4 Page(s)	19-Jan-2012	Technical Paper	NRC/NRO/DARR/APOB		05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML120190367	01/31/2012 - Notice of Meeting with GEH and Industry to Discuss the Economic Simplified Boiling-Water Reactor Design Certification Steam Dryer Concerns. 10 Page(s)	20-Jan-2012	Meeting Notice Meeting Agenda	NRC/NRO/DARR/APOB	NRC/NRO/DARR/APOB	05200010
ML120300012	1/31/2012 Slides for ESBWR Steam Dryer Public Meeting. 18 Page(s)	30-Jan-2012	Meeting Briefing Package/Handouts Slides and Viewgraphs	NRC/NRO/DARR/APOB		05200010
ML120470217	Letter - Audit Plan of the Economic Simplified Boiling Water Reactor Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document. 6 Page(s)	16-Feb-2012	Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML120450604	01/31/2012 - Summary of Public Meeting to Discuss Economic Simplified Boiling Water Reactor Steam Dryer Concerns. 9 Page(s)	17-Feb-2012	Meeting Summary Memoranda	NRC/NRO/DARR/APOB	NRC/NRO/DARR/APOB	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML120690025	Audit Plan Revision 1 of The Economic Simplified Boiling Water Reactor Steam Dryer Design Methodology Supporting Chapter 3 of The Economic Simplified Boiling Water Reactor Design Control Document. 1 Page(s)	12-Mar-2012	Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12073A165	Quality Assurance Implementation Inspection of Economic Simplified Boiling Water Reactor. 5 Page(s)	14-Mar-2012	Letter	NRC/NRO/DCIP/CQAB	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML120790454	Audit Plan Of ESBWR Steam Dryer Design Methodology Supporting Chapter 3 of ESBWR DCD, Revision 1 (Non-Proprietary Version). 12 Page(s)	20-Mar-2012	Audit Report	NRC/NRO/DNRL/LB3		05200010
ML120960622	04/18/12-Notice Of Forthcoming Meeting to Discuss Outstanding Requests for Additional Information Responses Relating to Steam Dryer Audit with General Electric Hitachi Nuclear Energy to Support The Economic Simplified Boiling Water Reactor. 8 Page(s)	6-Apr-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML121030136	4/25/2012 - Notice Of Forthcoming Meeting To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With GE Hitachi Nuclear Energy To Support The Economic Simplified Boiling Water Reactor Design Certification. 8 Page(s)	12-Apr-2012	Meeting Notice Meeting Agenda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML121100138	5/2/2012 Notice Of Forthcoming Meeting To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling Water Reactor Design .... 8 Page(s)	19-Apr-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML121150512	05/09/12-Notice of Forthcoming Meeting to Discuss Outstanding Requests For Addl. Info Responses Relating to Steam Dryer Audit With General Electric Hitachi Nuclear Energy to Support The Economic Simplified Boiling Water Reactor Design Certification. 8 Page(s)	24-Apr-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML121180063	Letter - Request For Additional Information Letter No. 414 Related To ESBWR Design Certification Application (DCD) Revision 9. 1 Page(s)	1-May-2012	NRO Safety Evaluation Report (SER)-Delayed Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML121230237	05/16/12, Notice of Forthcoming Meeting to Discuss Outstanding Requests For Additional Info Responses Relating to Steam Dryer Audit With General Electric Hitachi Nuclear Energy to Support Economic Simplified Boiling Water Reactor Design Certification. 8 Page(s)	2-May-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12124A233	04/18/2012-Summary of Meeting between NRC and GEH to Discuss Outstanding RAI Responses Relating To Steam Dryer Audit to Support The ESBWR Design Certification. 7 Page(s)	7-May-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12125A067	5/23/2012 - Notice of Meeting to Discuss Outstanding Requests for Additional Information Responses Relating to Steam Dryer Audit with General Electric Hitachi Nuclear Energy to Support the Economic Simplified Boiling Water Reactor Design Certification. 8 Page(s)	7-May-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12125A096	Notice of Forthcoming Meeting to Discuss Outstanding Requests for Additional Information Responses Relating to Steam Dryer Audit with General Electric Hitachi Nuclear Energy to Support the Economic Simplified Boiling Water Reactor Design Certification. 8 Page(s)	7-May-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12125A224	06/06/2012 Notice of Meeting to Discuss Outstanding Requests for Additional Information Responses Relating to Steam Dryer Audit with General Electric Hitachi Nuclear Energy to Support the Economic Simplified Boiling Water Reaction Design Certification. 8 Page(s)	10-May-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12128A002	6/13/2012 - Notice of Meeting to Discuss Outstanding Requests for Additional Information Responses Relating to Steam Dryer Audit with General Electric Hitachi Nuclear Energy to Support the Economic Simplified Boiling Water Reactor Design Certification. 8 Page(s)	10-May-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12128A026	6/20/2012 - Notice of Meeting to Discuss Outstanding Requests for Additional Information Responses Relating to Steam Dryer Audit with General Electric Hitachi Nuclear Energy to Support the Economic Simplified Boiling Water Reactor Design Certification. 8 Page(s)	10-May-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12128A045	6/27/2012 - Notice of Meeting to Discuss Outstanding Requests for Additional Information Responses Relating to Steam Dryer Audit with General Electric Hitachi Nuclear Energy to Support the Economic Simplified Boiling Water Reactor Design Certification. 8 Page(s)	10-May-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNR L/LB3	NRC/NRO/DNR L/LB3	05200010
ML12131A566	(Public-Letter) Handouts For 05-09-12 Telecom on Draft RAI Responses Related To The Audit of the ESBWR Stream Dryer Design Methodology. 25 Page(s)	10-May-2012	Meeting Briefing Package/Handouts	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNR L/LB3	05200010
ML121380119	Handouts for 5-16-12 Telecom on Draft RAI 3.9-290 Responses. 5 Page(s)	10-May-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNR L/LB4	05200010
ML121380111	Handouts for 5-16-12 Telecom on Draft RAI 3.9-282 Responses. 13 Page(s)	14-May-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNR L/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12143A108	4/25/2012 - Meeting Summary Between The U.S. Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit To Support The ESBWR DC. 7 Page(s)	22-May-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12152A249	5/2/2012-Meeting Summary Between The U.S. Nuclear Regulatory Commission Staff and General Electric-Hitachi Nuclear Energy to Discuss Outstanding Requests for Additional Information Responses Relating to Steam Dryer Audit... 7 Page(s)	31-May-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12152A064	5/30/2012 Handouts For Telecom On Draft RAI Responses Related To The Audit Of The ESBWR Stream Dryer Design Methodology (Public). 25 Page(s)	31-May-2012	Meeting Briefing Package/Handouts Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12173A109	GE-Hitachi's Response to NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - RAI 3.9-273. 12 Page(s)	1-Jun-2012	Letter Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO/DNRL/LB3	05200010
ML12171A090	NRC Requests for Additional Information Related to the Audit of the, Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - RAI 3.9-282. 16 Page(s)	5-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML12171A096	GE Hitachi Nuclear Energy Draft Response to NRC Requests for Additional Information Related RAI 3.8-281 to the Audit of Economic Simplified Boiling Water Reactor Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document. 10 Page(s)	5-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12171A094	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - RAI 3.9-272. 14 Page(s)	5-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML12159A087	RAI 3.9-281 Draft Response (public). 10 Page(s)	5-Jun-2012	Meeting Briefing Package/Handouts	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNRL/LB3	05200010
ML12159A108	RAI 3.9-284 Draft Response (public). 11 Page(s)	5-Jun-2012	Meeting Briefing Package/Handouts Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNRL/LB3	05200010
ML12165A310	Summary of NRC ESBWR Review Status. 3 Page(s)	7-Jun-2012	- No Document Type Applies	NRC/NRO		05200010
ML12171A098	Final Responses to NRC RAIs 3.9-289, 3.9-290 and 3.9-291 Related to the Audit of the Economic Simplified Boiling Water Reactor Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document. 26 Page(s)	7-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12165A438	GEH Proprietary Review of NRC Inspection Report 05200010-12-201 and Notice of Violation. 41 Page(s)	8-Jun-2012	Letter Notice of Violation Inspection Report	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12170B029	GE Hitachi Nuclear Energy Response to NRC RAI 3.9-281 Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document. 10 Page(s)	12-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML12165A164	5/16/2012 - Summary Of Meeting Between The U.S. Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With General Electric.... 7 Page(s)	13-Jun-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12170B031	Draft Response to NRC Requests for Additional Information RAI 3.9-280, Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document. 13 Page(s)	13-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML12166A093	Letter - Audit Report of the Steam Dryer Design Methodology Supporting Chapter 3 of the Economic Simplified Boiling Water Reactor Design Control Document. 6 Page(s)	14-Jun-2012	Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12166A127	Audit Report of the Economic Simplified Boiling Water Reactor Steam Dryer Design Methodology Supporting Chapter 3 of ESBWR Design Control Document. 8 Page(s)	14-Jun-2012	Audit Report	NRC/NRO/DNRL/LB3		05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML121710276	07/03/2012 - Notice of Forthcoming Meeting to Discuss Outstanding Requests for Additional Information Responses Relating to Steam Dryer Audit With General Electric Hitachi Nuclear Energy to Support the Economic Simplified Boiling Water Reactor Design. 8 Page(s)	19-Jun-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML121730330	NRC Requests for Additional Information (RAI) Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document – Draft Response for RAI 3.9-285. 15 Page(s)	19-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO/DNRL/LB3	05200010
ML121840083	NRC Requests for Additional Information re Audit of the ESBWR Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR DCD - Draft Response for RAI 3.9-286. 13 Page(s)	20-Jun-2012	Letter	NRC/NRO/DNRL/LB3 GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNRL	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML121720225	May 23, 2012 Summary Of Meeting Between The U.S. Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit To Support The Economic. 7 Page(s)	21-Jun-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML121840059	NRC Requests for Additional Information re Audit of the ESBWR Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Draft Response for RAI 3.9-278. 15 Page(s)	25-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNRL/LB2	05200010
ML121840040	NRC Requests for Additional Information re Audit of the Economic Simplified BWR (ESBWR) Steam Dryer Design Control Document - Draft Response RAI 3.9-271. 8 Page(s)	26-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML121840040	NRC Requests for Additional Information re Audit of the Economic Simplified BWR (ESBWR) Steam Dryer Design Control Document - Draft Response RAI 3.9-271. 8 Page(s)	26-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNRL/LB3	05200010
ML12178A038	05/30/2012 Summary of Meeting Between the U.S. Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy to Discuss Outstanding Requests For Additional Information Responses Relating to Steam Dryer Audit to Support The Economic..... 7 Page(s)	26-Jun-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12178A053	7/11/2012 - Notice Of Forthcoming Meeting To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design.... 7 Page(s)	26-Jun-2012	Meeting Notice Meeting Agenda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12300A097	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Draft Response for RAI 3.9-283. 11 Page(s)	27-Jun-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML12177A201	NRC Approval of GE Hitachi Energy Americas Request for Withholding Information from Public Disclosure. 4 Page(s)	3-Jul-2012	Letter	NRC/NRO/DCIP/CQAB	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12185A224	Summary of June 6, 2012, Meeting Between The U.S. Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit To Support The ..... 7 Page(s)	5-Jul-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12187A102	IR 05200010-12-201, on 4/16-20/2012, at the General Electric-Hitachi (GEH) Nuclear Energy Facility in Wilmington, NC; and Notice of Violation. 34 Page(s)	6-Jul-2012	Inspection Report Letter Notice of Violation	NRC/NRO/DCIP/CQAB	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12192A623	7/25/2012 - Notice Of Forthcoming Meeting To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design..... 8 Page(s)	11-Jul-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12192A623	7/25/2012 - Notice Of Forthcoming Meeting To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design..... 8 Page(s)	11-Jul-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12198A203	06/13/2012 Summary Of Meeting Between The U.S. Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit To Support The Economic..... 7 Page(s)	16-Jul-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12199A402	8/01/2012 Notice Of Forthcoming Meeting To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design. 8 Page(s)	18-Jul-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

ML12209A070	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Draft Response for RAI 3.9-276. 12 Page(s)	18-Jul-2012	Letter Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12209A072	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Second Draft Response for RAI 3.9-277. 15 Page(s)	19-Jul-2012	Legal-Affidavit Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML12201A121	06/20/2012 - Summary of Meeting Between The U.S. NRC Staff and General Electric-Hitachi Nuclear Energy to Discuss Outstanding RAI Responses Relating to Steam Dryer Audit to Support The Economic Simplified Boiling Water Reactor Design Certification. 7 Page(s)	19-Jul-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12206A377	8/8/2012 Notice Of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 8 Page(s)	24-Jul-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12209A074	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Draft Response for RAI 3.9-275. 16 Page(s)	25-Jul-2012	Legal-Affidavit Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO/DNRL/LB3	05200010
ML12207A384	6/27/2012 - Meeting Summary Between The U.S. NRC Staff and General Electric-Hitachi Nuclear Energy To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit To Support The Economic Simplified Boiling Water Reactor Design Certification. 7 Page(s)	25-Jul-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12213A531	8/15/2012-Notice of Forthcoming Meeting to Discuss Outstanding RAI Responses Relating to Steam Dryer Audit With General Electric Hitachi Nuclear Energy to Support the Economic Simplified Boiling-Water Reactor Design Certification. 8 Page(s)	31-Jul-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12214A229	7/3/2012-Meeting Summary Between The U.S. NRC Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit To Support The Economic Simplified Boiling Water Reactor Design Certification. 7 Page(s)	1-Aug-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12219A065	GE Hitachi Nuclear Energy Reply to a Notice of Violation, NRC Inspection Report 05200010/2012-201. 2 Page(s)	3-Aug-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML12219A067	Enclosure 2 - MFN 12-075, GEH Reply to NRC Notice of Violation Docket Number 05200010/2012-201-01, 05200010/2012-201-02, 05200010/2012-201-03. 8 Page(s)	3-Aug-2012	Licensee Response to Notice of Violation	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML12219A068	Enclosure 3 - MFN 12-075, GEH Reply to NRC Notice of Violation Docket Number 05200010-12-201-01,05200010-12-201-02, 05200010-12-201-03. 4 Page(s)	3-Aug-2012	Licensee Response to Notice of Violation	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12221A039	8/22/2012 - Notice of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 8 Page(s)	8-Aug-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12221A060	07/11/2012, Summary of Meeting Between the U.S. NRC Staff and General Electric-Hitachi Nuclear Energy to Discuss Outstanding RAI Responses Relating to Steam Dryer Audit to Support the Economic Simplified Boiling Water Reactor Design Certification. 7 Page(s)	8-Aug-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12228A666	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Draft Response for RAI 3.9-279. 9 Page(s)	13-Aug-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12242A389	Transcript of the ACRS ESBWR Subcommittee Meeting on August 16, 2012 [OPEN]. 473 Page(s)	16-Aug-2012	Meeting Transcript	NRC/ACRS		05200010 05200033
ML12234A555	09/05/2012 Notice of Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 8 Page(s)	23-Aug-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12222A372	Request for Withholding Information from Public Disclosure (MFN 12-038 Revision 1). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12220A590	Request For Withholding Information From Public Disclosure (MFN 12-046, Revision 1). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12228A398	Request For Withholding Information From Public Disclosure (MFN 12-047). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12228A424	Request For Withholding Information From Public Disclosure (MFN 12-048). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12222A217	Request For Withholding Information From Public Disclosure (MFN 12-051). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12221A302	Request For Withholding Information From Public Disclosure (MFN 12-052). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12222A259	Request For Withholding Information From Public Disclosure (MFN 12-052, Revision 1). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12222A367	Request For Withholding Information From Public Disclosure (MFN 12-066, Revision 1). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12220A037	Request For Withholding Information From Public Disclosure (MFN-12-070). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12228A447	Request For Withholding Information From Public Disclosure (MFN 12-072). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12228A468	Request For Withholding Information From Public Disclosure (MFN 12-086). 6 Page(s)	27-Aug-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12236A200	08/01/2012 - Summary of Meeting Between The U.S. NRC Staff and General Electric-Hitachi Nuclear Energy To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit To Support The Economic Simplified Boiling Water Reactor Design Certification. 7 Page(s)	28-Aug-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12234A643	09/12/2012-Notice of Forthcoming Meeting to Discuss Outstanding Requests for Additional Information Responses Relating to Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support the Economic Simplified Boiling-Water Reactor Design 8 Page(s)	28-Aug-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12240A115	08/08/2012 Summary of Meeting Between the U.S. NRC Staff and General Electric-Hitachi Nuclear Energy to Discuss Outstanding RAI Responses Relating to Steam Dryer Audit to Support the Economic Simplified Boiling Water Reactor Design Certification. 7 Page(s)	29-Aug-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12236A340	9/19/2012-Notice of Forthcoming Meeting to Discuss Outstanding RAI Responses Relating to Steam Dryer Audit With General Electric Hitachi Nuclear Energy to Support the Economic Simplified Boiling-Water Reactor Design Certification. 8 Page(s)	29-Aug-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12236A365	9/26/2012-Notice of Forthcoming Meeting to Discuss Outstanding RAI Responses Relating to Steam Dryer Audit With General Electric Hitachi Nuclear Energy to Support the Economic Simplified Boiling-Water Reactor Design Certification. 8 Page(s)	29-Aug-2012	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12241A360	NRC Approval of GE Hitachi Nuclear Energy Americas, LLC - Request for Withholding Information from Public Disclosure. 4 Page(s)	30-Aug-2012	Letter	NRC/NRO/DCIP/CQAB	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12248A292	GE Hitachi Nuclear Energy Responses to NRC Requests for Additional Information Related to Audit of Economic Simplified BWR (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Draft Response for RAI 3.9-293. 13 Page(s)	31-Aug-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12250A810	08/15/2012 Summary of Meeting Between The US Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit To Support The Economic Simplified Boiling Water Reactor..... 7 Page(s)	6-Sep-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML122561128	Minutes of the ACRS ESBWR Subcommittee Meeting, August 16, 2012 [OPEN] 479 Page(s)	6-Sep-2012	Meeting Minutes Memoranda	NRC/ACRS	NRC/ACRS	05200010 05200033
ML122550034	09/12/2012, Notice of Forthcoming Meeting To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design.... 8 Page(s)	11-Sep-2012	Meeting Notice Meeting Agenda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12262A410	8/22/2012- Summary Meeting Between The U.S. Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding RA IResponses Relating To Steam Dryer Audit To Support The Economic Simplified Boiling Water Reactor Design 7 Page(s)	18-Sep-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12261A456	10/3/2012 - Notice Of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 1 Page(s)	18-Sep-2012	Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12264A066	Response to NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Draft Response for RAI 3.9-287. 13 Page(s)	18-Sep-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12264A124	Canceled-09-26-12 Notice of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 8 Page(s)	20-Sep-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12264A094	Canceled-10-03-12 Notice Of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 8 Page(s)	20-Sep-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12268A342	8/29/2012 Summary Of Meeting between U.S. NRC Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit To Support The Economic Simplified Boiling Water Reactor Design Certification. 7 Page(s)	25-Sep-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12271A235	GE Hitachi Nuclear Energy Americas LLC. Response to NRC Inspection Report 05200010/2012-201, Notice of Violation. 2 Page(s)	27-Sep-2012	Letter Inspection Report	NRC/NRO/DCIP/CQAB	Hitachi-GE Nuclear Energy, Ltd	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12276A174	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Draft Response for RAIs 3.9-269 & 3.9-270. 132 Page(s)	27-Sep-2012	Letter Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12275A326	ESBWR Standard Plant Design 10 CFR 50.46 2012 Annual Report. 3 Page(s)	28-Sep-2012	Annual Operating Report Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12276A013	09/19/2012 Summary of Meeting with the U.S. Nuclear Regulatory Commission staff and General Electric-Hitachi Nuclear Energy to Discuss Outstanding RAI Responses Relating to Steam Dryer Audit to Support the ESBWR DC. 7 Page(s)	2-Oct-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12279A102	Response to NRC Requests for Additional Information Related to Audit of the Economic Simplified Boiling Water Reactor Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Draft Response for RAI 3.9-280. 17 Page(s)	3-Oct-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12269A259	Letter re: Request For Withholding Information From Public Disclosure (MFN 12-050). 6 Page(s)	4-Oct-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB4	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12269A279	Request For Withholding Information From Public Disclosure (MFN 12-065). 6 Page(s)	4-Oct-2012	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12284A079	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Revised Draft Response for RAI 3.9-277. 31 Page(s)	5-Oct-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12285A106	NRC Requests for Additional Information (RAI) Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Draft Response for RAI 3.9-288. 19 Page(s)	9-Oct-2012	Legal-Affidavit Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12291A404	Revised Draft Response for RAI 3.9-284 Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document. 13 Page(s)	12-Oct-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12289A412	10/31/2012 - Notice Of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 8 Page(s)	15-Oct-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12296A163	Response to NRC Requests for Additional Information Related to Audit of Economic Simplified Boiling Water Reactor Steam Dryer Design Methodology Supporting Chapter 3 of ESBWR Design Control Document, Revised Draft Responses for RAIs 3.9-285 & 3.9-286. 57 Page(s)	17-Oct-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12262A447	11/07/12 - Notice Of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 1 Page(s)	23-Oct-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12297A272	Response to your February 15, 2012, Letter to the Executive Director of the Advisory Committee on Reactor Safeguards Concerning a Request for Copies of Transcripts for Multiple ACRS Full Committee and ESBWR Subcommittee Meetings. 5 Page(s)	24-Oct-2012	Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12305A471	11/14/2012 - Notice Of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 2 Page(s)	31-Oct-2012	Meeting Agenda Meeting Notice	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12311A128	11/28/2012 - Notice Of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 1 Page(s)	6-Nov-2012	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12324A262	Comment (4) of Patricia Campbell of GE-Hitachi Nuclear Energy on NRC-2012-0237, Proposed Revision Treatment of Non-Safety Systems for Passive Advanced Light Water Reactors. 6 Page(s)	13-Nov-2012	General FR Notice Comment Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/ADM/DAS/RDEB	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12325A086	Summary of 10/31/2012, Meeting Between The U.S. Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit To Support The Economic Simplified Boiling Water Reactor. 7 Page(s)	21-Nov-2012	Meeting Summary Meeting Agenda Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12333A088	November 7, 2012 - Summary Of Meeting Between The U.S. NRC Staff And General Electric-Hitachi Nuclear Energy To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit To Support The Economic Simplified Boiling Water Reactor Design Certification. 7 Page(s)	28-Nov-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML12345A232	11/14/2012 Summary of Meeting with U.S. Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy to Discuss Outstanding RAI Responses Relating to Steam Dryer Audit to Support The Economic Simplified Boiling Water Reactor. 7 Page(s)	10-Dec-2012	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12348A139	Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document. 3 Page(s)	12-Dec-2012	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML12334A092	Request For Withholding Information From Public Disclosure (MFN 12-058). 6 Page(s)	7-Jan-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13022A533	Response to NRC Requests for Additional Information Related to the Audit of ESBWR Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Final Response for RAI 3.9-270. 2 Page(s)	21-Jan-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO Document Systems, Inc	05200010
ML13022A535	Enclosure 2 to MFN 13-003 - Final Response for RAI 3.9-270. 2 Page(s)	21-Jan-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13022A529	Enclosure 3 to MFN 13-003 - Affidavit of Jerald G. Head Requesting Withholding of Enclosure 1 from Public Disclosure. 4 Page(s)	21-Jan-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13024A230	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Final Response for RAI 3.9-276. 2 Page(s)	23-Jan-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13024A232	Enclosure 2 to MFN 12-048, Rev. 1, Final Response for RAI 3.9-276 Related to the Audit of the Economic Simplified Boiling Water Reactor Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document. 6 Page(s)	23-Jan-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13024A233	Enclosure 3 to MFN-048, Rev. 1. - Affidavit of Jerald G. Head Setting Forth the Basis for Requesting Enclosure 1 be Withheld from the Public, 4 Page(s)	23-Jan-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13025A232	Enclosure 2 to MFN 12-046, Revision 2, Final Response to RAI 3.9-272 (Public Version). 10 Page(s)	24-Jan-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13025A233	Enclosure 3 to MFN 12-046, Revision 2, Affidavit of Peter M. Yandow Re NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor Steam Dryer Design Methodology Support Chapter 3. 4 Page(s)	24-Jan-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13025A054	Enclosure 2 to MFN 12-050, Revision 1, Final Response for RAI 3.9-279 (Public Version). 3 Page(s)	24-Jan-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13025A055	Enclosure 3 to MFN 12-050, Rev. 1, Affidavit of Peter M. Yandow Re NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR.... 4 Page(s)	24-Jan-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13025A052	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Final Response for RAI 3.9-279. 2 Page(s)	24-Jan-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13025A229	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Final Response to RAI 3.9-272. 2 Page(s)	24-Jan-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13024A244	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Final Response to RAI 3.9-274. 2 Page(s)	24-Jan-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13024A246	Enclosure 2 to MFN 12-040, Revision 2, Final Response to RAI 3.9-274 (Public Version). 4 Page(s)	24-Jan-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13024A247	Enclosure 3 to MFN 12-040, Revision 2, Affidavit for Peter M. Yandow, GE-Hitachi Nuclear Energy Americas	24-Jan-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13031A478	Enclosure 2, Final Responses for RAIs 3.9-289, 3.9-290 and 3.9-291 (Public). 33 Page(s)	30-Jan-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13031A481	Enclosure 5, Affidavit. 4 Page(s)	30-Jan-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13031A476	NRC Requests for Additional Information (RAI) Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Final Responses for RAIs 3.9-289, 3.91. 5 Page(s)	30-Jan-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13031A480	Enclosure 4, Marked-up Pages for ESBWR DCD Section 3L Related to RAI 3.9- 291 Response. 8 Page(s)	30-Jan-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13031A479	Enclosure 3, History of Draft and Final Response for RAI 3.9-289, 3.9-290 and 3.9-291. 2 Page(s)	30-Jan-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13032A595	Response to NRC Requests for Additional Information Related to Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Final Response to RAI 3.9-278. 2 Page(s)	31-Jan-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13032A597	Enclosure 2 to MFN-12-049 Revision 1, GEH Final Response to RAI 3.9-278. 9 Page(s)	31-Jan-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13032A598	Enclosure 3 to MFN-12-049 Revision 1 - Affidavit of Jerald G. Head Regarding Information to be Withheld. 4 Page(s)	31-Jan-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13035A147	2/20/2013 - Notice Of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 8 Page(s)	5-Feb-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13052A161	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Final Responses to RAIs 3.9-285 and 3. 70 Page(s)	5-Feb-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13036A118	2/27/2013 - Notice Of Forthcoming Meeting To Discuss Outstanding RAI Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design Certification. 7 Page(s)	6-Feb-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13038A300	Enclosure 3 to MFN-12-043, Rev 1, Final Response to RAI 3.9-269. 5 Page(s)	7-Feb-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13038A303	Enclosure 6 to MFN-12-043, Rev. 1, Affidavit. 4 Page(s)	7-Feb-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13038A304	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Final Response for RAI 3.9-269. 3 Page(s)	7-Feb-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13039A334	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Final Response to RAI 3.9-271. 8 Page(s)	8-Feb-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13136A247	06/06/2013 - Notice Of Forthcoming Meeting To Discuss The Economic Simplified Boiling-Water Reactor Design Certification Request For Information Related To Bulletin 2012-01. 9 Page(s)	21-May-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13156A170	DTE Electric Company Response To U. S. Nuclear Regulatory Commission's Bulletin 2012-01 FERMI 2 COLA - Handout for June 6, 2013. 8 Page(s)	5-Jun-2013	Meeting Briefing Package/Handouts Slides and Viewgraphs	DTE Energy	NRC/NRO	05200010 05200030
ML13156A024	ESBWR Design Certification Steam Dryer Request for Additional Information - Handout for June 6, 2013. 9 Page(s)	5-Jun-2013	Meeting Briefing Package/Handouts Meeting Notice Slides and Viewgraphs	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010 05200033
ML13157A037	6/20/13 - Notice of Forthcoming Design Center Working Group Public Meeting With General Electric-Hitachi Nuclear Energy, DTE Electric Company, and Dominion Virginia Power To Discuss The Program Status Of The ESBWR Design Center. 9 Page(s)	7-Jun-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010 05200017 05200021 05200033

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13164A019	Submittal of Responses to NRC Requests for Additional Information Related to ESBWR Steam Dryers in Groupings. 2 Page(s)	12-Jun-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13170A245	Revised Completion Date for Corrective Action for NOV 05200010/2012-201-02 of NRC Inspection Report 05200010-12-201. 2 Page(s)	18-Jun-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13171A178	Revised Completion Date for Corrective Action for NOV 05200010/2012-201-02 of NRC Inspection Report 05200010/2012-201. 2 Page(s)	20-Jun-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13177A143	06/06/2013 Meeting Summary Between The U.S. Nuclear Regulatory Commission Staff, General Electric-Hitachi Nuclear Energy, and DTE Electric Company T Discuss The ESBWR Design Certification And Fermi Unit 3 COL RAI Related To Bulletin 2012-01. 11 Page(s)	1-Jul-2013	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010 05200033
ML13183A252	Request For Additional Information Letter No. 416 Related TO ESBWR Design Certification Application. 9 Page(s)	3-Jul-2013	Letter Request for Additional Information (RAI)	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13186A059	Letter- Response to Hitachi Nuclear Energy Letter Regarding Economic Simplified Boiling-Water Reactor Design Certification Final Rule. 4 Page(s)	12-Jul-2013	Letter	NRC/NRO/DARR	Hitachi-GE Nuclear Energy, Ltd	05200010
ML13197A469	6/20/13 - Summary Of Meeting Between The U. S. Nuclear Regulatory Commission Staff, General Electric-Hitachi Nuclear Energy, DTE Electric Company, And Dominion Virginia Power To Discuss The Program Status of the Economic Simplified Boiling-Water Reactor. 12 Page(s)	18-Jul-2013	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010 05200017 05200033
ML13211A206	GEH, Response to Request for Additional Information 8.1-22 Related to ESBWR Design Certification Application. 2 Page(s)	30-Jul-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13211A208	ESBWR DCD Marked-up Pages Associated with GEH Response to RAI 8.1-22. 6 Page(s)	30-Jul-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13211A207	GEH, Response to RAI 8.1-22. 10 Page(s)	30-Jul-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13217A459	8/22/13 - Notice of Forthcoming Design Center Working Group Public Meeting With GEH, DTE And Dominion To Discuss The ESBWR Design Center. 11 Page(s)	6-Aug-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010 05200017 05200033
ML13218B057	8/21/13 - Notice of Forthcoming Public Meeting with NINA to Support South Texas Project, Units 3 and 4 COLA. 9 Page(s)	7-Aug-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13225A040	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBRW) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Response to RAI's 3.9-271 & 3.9-271 S01. 9 Page(s)	9-Aug-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13225A280	NRC Requests for Additional Information re Audit of Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Response for RAI 3.9-269, Supplement 1. 22 Page(s)	9-Aug-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13225A039	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Response to RAI 3.9.294. 8 Page(s)	9-Aug-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12341A371	Request For Withholding Information From Public Disclosure (MFN 12-043). 6 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13036A235	Request For Withholding Information From Public Disclosure (MFN 12-046, Revision 2). 6 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13057A849	Request For Withholding Information From Public Disclosure (MFN 12-047, Revision 1). 5 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13036A243	Request For Withholding Information From Public Disclosure (MFN 12-048, Revision 1). 6 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13036A227	Request For Withholding Information From Public Disclosure (MFN 12-040, Revision 2). 7 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13042A161	Request For Withholding Information From Public Disclosure (MFN 12-049, Revision 1). 5 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12341A334	Request For Withholding Information From Public Disclosure (MFN 12-051, Revision 1)	12-Aug-2013	Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13057A869	Request For Withholding Information From Public Disclosure (MFN 12-051, Revision 2). 5 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12341A378	Request For Withholding Information From Public Disclosure (MFN 12-054). 6 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13057A871	Request For Withholding Information From Public Disclosure (MFN 12-054 Revision 1). 5 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12341A380	Request For Withholding Information From Public Disclosure (MFN 12-055, Revision 1). 6 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13057A874	Request For Withholding Information From Public Disclosure (MFN 12-055, Revision 2). 5 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13057A876	Request For Withholding Information From Public Disclosure (MFN 12-058, Revision 1). 5 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12341A384	Request For Withholding Information From Public Disclosure (MFN 12-059). 6 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13043A784	Request For Withholding Information From Public Disclosure (MFN 12-059, Revision 1). 5 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML12341A389	Request For Withholding Information From Public Disclosure (MFN 12-077, Revision 1). 6 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML12341A353	Request For Withholding Information From Public Disclosure (MFN 12-086, Revision 1). 6 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13057A898	Request For Withholding Information From Public Disclosure (MFN 12-086, Revision 2). 5 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13036A256	Request For Withholding Information From Public Disclosure (MFN 13-003). 6 Page(s)	12-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13074A391	Request For Withholding Information From Public Disclosure (MFN 13-010). 5 Page(s)	19-Aug-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13242A213	Enclosure 2 - ESBWR DCD Marked-up Pages Associated with GEH Response to RAI 8.1-22. 10 Page(s)	28-Aug-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13242A215	Enclosure 1: Description of Changes for ESBWR DCD Tier 2, Chapter 16 & Chapter 16B Markups re GEH Corrective Action. 3 Page(s)	28-Aug-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13242A220	Enclosure 2: Transmittal of ESBWR DCD Tier 1 Markups Related to GEH Corrective Action – DCD Markups. 29 Page(s)	28-Aug-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13242A211	GE Hitachi Nuclear Energy - NRC Request for Additional Information Related to ESBWR Design Certification Application – GEH Supplemental Response to RAI 8.1-22. 2 Page(s)	28-Aug-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13242A219	Enclosure 1: Transmittal of ESBWR DCD Tier 1 Markups Related to GEH Corrective Action – Description of Changes. 4 Page(s)	28-Aug-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13242A217	Enclosure 2: DCD Markups for ESBWR DCD Tier 2, Chapter 16 & Chapter 16B re GEH Corrective Action. 30 Page(s)	28-Aug-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13242A212	Enclosure 1 - GEH Supplemental Response to RAI 8.1-22. 4 Page(s)	28-Aug-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13242A218	Transmittal of ESBWR DCD Tier 1 Markups Related to GEH Corrective Action. 2 Page(s)	28-Aug-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13242A214	Transmittal of ESBWR DCD Tier 2, Chapter 16 and Chapter 16B Markups re GEH Corrective Action. 2 Page(s)	28-Aug-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13043A800	Request For Withholding Information From Public Disclosure (MFN 12-065, Revision 1). 5 Page(s)	4-Sep-2013	Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13042A167	Request For Withholding Information From Public Disclosure (MFN 12-066, Revision 2). 5 Page(s)	4-Sep-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB4	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13067A199	Request For Withholding Information From Public Disclosure (MFN 12-077, Revision 2). 5 Page(s)	4-Sep-2013	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13067A182	Request For Withholding Information From Public Disclosure (MFN 13-007). 5 Page(s)	4-Sep-2013	Proprietary Information Review Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13255A204	10/01/2013 Notice of Forthcoming Meeting To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With General Electric-Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor Design..... 6 Page(s)	16-Sep-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3		
ML13262A232	GE Hitachi Nuclear Energy, Enclosure 2 to MFN 12-051, Rev. 3, GEH Response to RAI 3.9-280 S01. 6 Page(s)	18-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13261A400	GE Hitachi Nuclear Energy, Enclosure 2 to MFN 12-065, GEH Response to RAIs 3.9-293 S01, S02, S03 (Public Version). 10 Page(s)	18-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13262A237	Enclosure 3 (Affidavit). 4 Page(s)	18-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy	NRC/NRO	05200010
ML13262A236	Enclosure 2, GEH Response to RAI 3.9-296 (Public Information). 21 Page(s)	18-Sep-2013	Report, Miscellaneous	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13261A406	Enclosure 3 (Affidavit). 4 Page(s)	18-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13262A235	NRC Request for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Response to RAI 3.9-298. 4 Page(s)	18-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13262A238	NRC Requests for Additional Information (RAI) Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Response to RAI 3.9-280 S01. 2 Page(s)	18-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13262A426	Enclosure 2 - MFN 12-038, Revision 2 Response for RAI 3.9-273, S01 - Public Information. 10 Page(s)	18-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13261A517	NRC Request for Additional Information (RAI) Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Response to RAI 3.9-297. 5 Page(s)	18-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13262A433	NRC Request for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Response to RAI 3.9-273 S01. 2 Page(s)	18-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13261A398	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Response to RAIs 3.9-293 S01, S02, S03. 2 Page(s)	18-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13262A234	GE Hitachi Nuclear Energy, Enclosure 3 to MFN 12-051, Affidavit of Jerald G. Head Regarding NRC Requests for RAI Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Support Chapter 3 of the.... 4 Page(s)	18-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13261A472	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document GEH Response to RAI 3.9-295..... 2 Page(s)	18-Sep-2013	Letter	General Electric Hitachi Morris Operation	NRC/Document Control Desk NRC/NRO	05200010
ML13261A405	Enclosure 2, GEH Response to RAI 3.9-295 (Public Information). 2 Page(s)	18-Sep-2013	Report, Miscellaneous	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13262A427	Enclosure 3 - MFN 12-038, Revision 2, "Affidavit". 4 Page(s)	18-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13262A230	NRC Request for Additional Information (RAI) Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Response to RAI 3.9-296. 2 Page(s)	18-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO NRC/Document Control Desk	05200010
ML13261A401	GE Hitachi Nuclear Energy, Enclosure 3 to MFN 12-065, ESBWR Design Control Document Marked-Up Pages. 5 Page(s)	18-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13261A402	GE Hitachi Nuclear Energy, Enclosure 4 to MFN 12-065, Revision 2, Affidavit for Jerald G. Head. 4 Page(s)	18-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13260A433	08/22/2013 Summary of Meeting Between The NRC Staff, General Electric Hitachi Nuclear Energy, DTE Electric Company And Dominion Virginia Power To Discuss The Program Status Of The Economic Simplified Boiling Water Reactor Design Center. 12 Page(s)	19-Sep-2013	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010 05200017 05200033

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13263A121	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Revised Response to RAI 3.9-277. 3 Page(s)	19-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13263A127	Enclosure 2 to MFN 12-086, Rev 3 - GEH Revised Response to RAI 3.9-277 36 Page(s)	19-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13263A125	Enclosure 4 to MFN 12-086, Rev 3, GEH Response to RAI 3.9-277 S01. 5 Page(s)	19-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13263A126	Enclosure 5 to MFN 12-086, Revision 3 - Affidavit of Jerald G. Head Regarding NRC RAI Related to the Audit of the Economic Simplified Boiling	19-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13267A247	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Response to RAI 3.9-288 S01. 2 Page(s)	24-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13267A249	Enclosure 2 - MFN 12-059, Rev. 2, "GEH Response to RAI 3.9-288 S01". 13 Page(s)	24-Sep-2013	- No Document Type Applies	GE-Hitachi Global Laser Enrichment, LLC	NRC/NRO	05200010
ML13267A250	Enclosure 3 - MFN-12-059, Rev. 2 Affidavit. 4 Page(s)	24-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13268A459	NRC Requests For Additional Information Related To The Audit Of The Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 Of The ESBWR Design Control Document - GEH Revised Response To RAI 3.9-285..... 3 Page(s)	24-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13268A461	GE Hitachi Nuclear Energy, Enclosure 2 - MFN 12-077, Revision 3 GEH Revised Response to RAI 3.9-285 and Response to RAI 3.9-285 S01 (Public Information). 32 Page(s)	24-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13268A462	GE Hitachi Nuclear Energy, Enclosure 3-MFN 12-077, Revision 3 (Affidavit). 4 Page(s)	24-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13269A174	NRC Requests for Additional Information (RAI) Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Responses to RAIs 3.9-289, 3.9-290.... 3 Page(s)	25-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13269A176	Enclosure 2 to MFN 12-066, Rev. 3 - Responses to RAIs 3.9-289, 3.9-290 and 3.9-291; Including 3.9-291 S01 through S05. 40 Page(s)	25-Sep-2013	Report, Miscellaneous	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13269A178	Encl. 4 to MFN 12-066, Rev. 3, Affidavit Regarding NRC Requests for Additional Information (RAI) Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design.... 4 Page(s)	25-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13269A177	Enclosure 3 to MFN 12-066, Rev. 3 - History of Draft and Final Response for RAI 3.9-289, 3.9-291; and Associated Supplemental RAIs. 2 Page(s)	26-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13270A191	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document Response for RAI 3.9-292, S01. 2 Page(s)	26-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13270A193	Enclosure 1 - MFN 13-046 Response to RAI 3.9-292 S01. 3 Page(s)	26-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13270A195	Enclosure 3 - MFN 13-046 - GE Hitachi Nuclear Energy, ESBWR Steam Dryer Acoustic Load Definition, NEDO-33312P, Revision 4, Class I. 27 Page(s)	26-Sep-2013	Report, Technical	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13270A196	Enclosure 4 MFN 13-046 (Affidavit). 4 Page(s)	26-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13270A269	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Response to RAI 3.9-292 S02 and..... 3 Page(s)	26-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13270A273	Enclosure 4 to MFN 13-007, Revision 1, GE Hitachi Nuclear Energy, NEDO-33313, "ESBWR Steam Dryer Structural Evaluation," Class I (Non-Proprietary) Revision 4, September 2013 (Public Information). 60 Page(s)	26-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13270A274	Enclosure 5 to MFN 13-007, Revision 1, Affidavit of Jerald G. Head Regarding NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of.... 4 Page(s)	26-Sep-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13275A043	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - Revised Engineering Report NEDE-33408. 3 Page(s)	27-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13275A049	NRC Requests for Additional Information Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Response to RAI 3.9-292 S03. 7 Page(s)	27-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13275A050	Enclosure 1: "GEH Response to RAI 3.9-292 S03" and Enclosure 2: "Matrix of NRC RAIs and Responses". 55 Page(s)	27-Sep-2013	- No Document Type Applies Graphics incl Charts and Tables	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13275A051	Enclosure 3: "ESBWR Design Control Document Changes List" and Enclosure 4: "ESBWR Design Control Document Marked-Up Pages". 65 Page(s)	27-Sep-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13274A516	NRC Requests for Additional Info. Related to Audit of Economic Simplified Boiling Water Reactor Steam Dryer Design Methodology Supporting Chapter 3 of ESBWR Design Control Document-GEH Revised Response to RAI 3.9-286 & Response to RAI 3.9-286 S01. 3 Page(s)	27-Sep-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13274A517	Enclosure 2 - MFN 13-075 - GEH Revised Response to RAI 3.9-286 and Response to RAI 3.9-286 S01. 66 Page(s)	27-Sep-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13225A282	NEDO-33408, Revision 3, "ESBWR Steam Dryer - Plant Based Load Evaluation Methodology - PBLE01 Model Description". 172 Page(s)	30-Sep-2013	Legal-Affidavit Report, Technical	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13275A044	NEDO-33408, Rev. 4, "ESBWR Steam Dryer - Plant Based Load Evaluation Methodology PBLE01 Model Description." 186 Page(s)	30-Sep-2013	Topical Report	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13275A188	10/17/13- Notice of Forthcoming Meeting With General Electric-Hitachi Nuclear Energy To Discuss Proposed Revisions To the Economic Simplified Boiling-Water Reactor Design Certification Document That Addresses Steam Dryer Issues. 7 Page(s)	2-Oct-2013	Meeting Agenda Meeting Notice	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13291A291	11/01/2013 Notice of Forthcoming Meeting With General Electric Hitachi Nuclear Energy To Discuss Proposed Revisions To The Economic Simplified Boiling-Water Reactor Design Certification. 7 Page(s)	21-Oct-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL	NRC/NRO/DNRL/LB3	05200010
ML13298A479	ESBWR Standard Plant Design Annual 10 CFR 50.46 Report for 2013. 3 Page(s)	25-Oct-2013	Annual Report Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13302A393	10/1/13- Summary Of Meeting between U.S. Nuclear Regulatory Commission Staff and General Electric-Hitachi Nuclear Energy to Discuss Outstanding Requests For Additional Information Responses Relating to Steam Dryer Audit to Support ESBWR DCD. 6 Page(s)	29-Oct-2013	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13301A341	11/14/2013 Meeting Notice of Forthcoming Design Center Working Group Public Meeting With GEH, DTE, and Dominion To Discuss The	29-Oct-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010 05200017 05200033
ML13308A002	Draft ESBWR Steam Dryer RAIs (10-28-2013), Public. 15 Page(s)	1-Nov-2013	Meeting Briefing Package/Handouts Request for Additional Information (RAI)	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13309A037	Transmittal of Discussion Points Document to Support Closed Portion of Meeting on November 1, 2013, Regarding ESBWR Steam Dryers. 6 Page(s)	1-Nov-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13316B229	Request For Additional Information Letter No. 417 Related To ESBWR Design Certification Application (DCD) Revision 9. 5 Page(s)	13-Nov-2013	Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13316B654	Supplemental Requests For Additional Information Letter No. 417 Related To ESBWR Design Certification Application (DCD) Revision 9 (Public). 11 Page(s)	13-Nov-2013	Request for Additional Information (RAI)	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13318A060	11/25/13 - Notice of Forthcoming Closed Meeting To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With General Electric-Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water DC. 8 Page(s)	14-Nov-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13318A943	Audit Plan Of Request For Additional Information Letter No. 417 Concerning ESBWR Steam Dryer Design Methodology. 2 Page(s)	15-Nov-2013	Audit Plan	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13318A949	Letter To GEH Regarding Audit Plan Of Request for Additional Information Letter No. 417 Concerning ESBWR Steam Dryer Design Methodology. 5 Page(s)	15-Nov-2013	Letter	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13324A095	Revised Completion Date for Corrective Action for NOV 05200010/2012-201-02 of NRC Inspection Report 05200010-12-201 2 Page(s)	19-Nov-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13324A482	NRC Request for Additional Information Related to ESBWR Design Certification Application GEH Supplemental Response to RAI 8.1-22. 2 Page(s)	20-Nov-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13324A487	Enclosure 1 - MFN 13-040, Supplement 2, "GEH Supplemental Response to RAI 8.1-22. 4 Page(s)	20-Nov-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13324A484	Enclosure 2 - MFN 13-040, Supplement 2, "ESBWR DCD Marked-up Pages Associated with GEH Response to RAI 8.1-22. 9 Page(s)	20-Nov-2013	- No Document Type Applies	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13329A372	11/27/13 - Notice Of Forthcoming Meeting With GEH To Support The Economic Simplified Boiling Water Reactor Design Certification. 7 Page(s)	26-Nov-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13331A883	Revised Audit Plan of Request For Additional Information Letter NO. 417 Concerning ESBWR Steam Dryer Design. 2 Page(s)	27-Nov-2013	Audit Plan	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13336B339	12/05/2013 Notice Of Forthcoming Closed Meeting To Discuss Outstanding Requests For Additional Information Responses Relating To Steam Dryer Audit With General Electric Hitachi Nuclear Energy To Support The Economic Simplified Boiling-Water Reactor..... 6 Page(s)	4-Dec-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML13336B364	November 1, 2013, Summary of Meeting Between The U.S. Nuclear Regulatory Commission Staff And General Electric-Hitachi Nuclear Energy To Discuss Proposed Revisions To The Economic Simplified 7 Page(s)	5-Dec-2013	Meeting Summary Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13338A352	12/17/13 - Notice Of Forthcoming Design Center Working Group Teleconference Meeting With GEH, DTE, And Dominion To Discuss The ESBWR Design Center. 7 Page(s)	5-Dec-2013	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010 05200017 05200033
ML13344B155	NRC RAI Letter Number 417 Related to the Audit of the Economic Simplified Boiling Water Reactor (ESBWR) Steam Dryer Design Methodology Supporting Chapter 3 of the ESBWR Design Control Document - GEH Response for RAIs 3.9-299 through 3.9-303. 3 Page(s)	6-Dec-2013	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML13344B156	Enclosure 5 to MFN 13-091: GEH Response for RAIs 3.9-299 through 3.9-303. 33 Page(s)	6-Dec-2013	Graphics incl Charts and Tables	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13344B157	Enclosure 6 to MFN 13-091: GE Hitachi Nuclear Energy, "ESBWR Steam Dryer Acoustic Load Definition," NEDO-33312, Rev. 5, Class I (Non-Proprietary), December 2013. 24 Page(s)	6-Dec-2013	Report, Technical	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13344B158	Enclosure 7 to MFN 13-091: GE Hitachi Nuclear Energy, "ESBWR Steam Dryer Structural Evaluation," NEDO-33313, Rev. 5, Class I (Non-Proprietary), December 2013. 84 Page(s)	6-Dec-2013	Report, Technical	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13344B159	Enclosure 8 to MFN 13-091: GE Hitachi Nuclear Energy, "ESBWR Steam Dryer - Plant Based Load Evaluation Methodology - PBLE01 Model Description," NEDO-33408, Rev. 5 193 Page(s)	6-Dec-2013	Report, Technical	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13344B160	Enclosure 9 to MFN 13-091: ESBWR Design Control Document Marked-up Pages. 56 Page(s)	6-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13344B161	Enclosure 10 to MFN 13-091: Affidavit. 4 Page(s)	6-Dec-2013	Legal-Affidavit	GE-Hitachi Nuclear Energy Americas, LLC	NRC/NRO	05200010
ML13343A055	Revision 2 Audit Plan of Request for Additional Information Letter No. 417 Concerning ESBWR Steam Dryer Design Methodology. 2 Page(s)	9-Dec-2013	Audit Plan	NRC/NRO/DNRL/LB3	NRC/NRO/DNRL/LB3	05200010
ML14010A349	GE-Hitachi ESBWR Design Control Document Tier 1, Rev. 10 - ESBWR DCD Revision 10 Tier 1 853 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14010A353	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 00 - Table of Contents 230 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A354	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 01 - Introduction and General Description of Plant 409 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A356	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 01A to 01D - Introduction and General Description of Plant 96 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A361	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 03.01 to 03.08 - Design of Structures, Components, Equipment, and Systems 349 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A363	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 03.09 to 03.11 - Design of Structures, Components, Equipment, and Systems 174 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14010A372	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 05 - Reactor Coolant System and Connected Systems 165 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A379	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 08 - Electric Power 74 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A383	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 09 Change List 1 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A386	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 09B - Auxiliary Systems 16 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A387	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 09B Change List 1 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A388	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 10 - Steam and Power Conversion System 92 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14010A394	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 13 - Conduct of Operations 22 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A398	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 15 - Safety Analyses 580 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A400	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 15 Change List 1 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A402	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 16 - Generic Technical Specifications 286 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A405	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 16B Change List 2 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14010A412	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 19 - Probabilistic Risk Assessment and Severe Accidents 274 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14010A413	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 19 Change List 1 Page(s)	11-Dec-2013	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14072A344	GEH Proprietary Review of NRC Audit Report of Responses to Request for Additional Information Letter No. 417 Re Steam Dryer Design Methodology Supporting Chapter 3 of the Economically Simplified Boiling Water Reactor Design Certification Document. 2 Page(s)	13-Mar-2014	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14077A364	Federal Register Notice Regarding the 613th ACRS Meeting, April 10-12, 2014. 5 Page(s)	26-Mar-2014	Federal Register Notice Meeting Notice	NRC/SECY		05200010 PROJ0776
ML14093A141	GE Hitachi Nuclear Energy, Transmittal of Reference Documents for NEDE-31758P-A and NEDO-31758-A. 2 Page(s)	28-Mar-2014	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14093A144	Transmittal of Reference Documents for NEDE-31959P and NEDO-31959. 2 Page(s)	29-Mar-2014	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14093A249	GE Hitachi Nuclear Energy - Transmittal of Public Version of Licensing Topical Report NEDC-32992P-A. 3 Page(s)	31-Mar-2014	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14092A297	Completion of Corrective Actions for NOV 05200010/2012-201-02 of NRC Inspection Report 05200010/2012-201. 2 Page(s)	31-Mar-2014	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	
ML14093A138	Submittal of Licensing Topical Report, "EBWR Safeguards Assessment Report," NEDO-33391 Revision 3, March 2010 (Public Version). 5 Page(s)	31-Mar-2014	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14093A140	Implementation of Improved GE Steady-State Nuclear Methods. 6 Page(s)	31-Mar-2014	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	
ML14100A493	GE-Hitachi ESBWR Design Control Document Tier 1, Rev. 10 - ESBWR DCD Revision 10 Tier 1 853 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A495	GE-Hitachi ESBWR Design Control Document Tier 1, Rev. 10 - ESBWR DCD Revision 10 Tier 1 Change List 5 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14100A498	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 00 - Table of Contents 230 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A499	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 01 - Introduction and General Description of Plant 410 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A500	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 01 Change List 5 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A501	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 01A to 01D - Introduction and General Description of Plant 96 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A503	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 01A to 01D Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A504	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 02 - Site Characteristics 39 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14100A505	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 02 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A506	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 03.01 to 03.08 - Design of Structures, Components, Equipment, and Systems 349 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A507	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 03.01 to 03.08 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A508	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 03.09 to 03.11 - Design of Structures, Components, Equipment, and Systems 174 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A509	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 03.09 to 03.11 Change List 3 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14100A510	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 03A to 03F - Design of Structures, Components, Equipment, and Systems 335 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A512	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 03A to 03F Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A513	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 03G to 03L - Design of Structures, Components, Equipment, and Systems 524 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A515	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 03G to 03L Change List 7 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A517	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 04 - Reactor 180 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A518	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 04 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14100A520	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 05 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A521	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 06 - Engineered Safety Features 645 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A522	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 06 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A523	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 07 - Instrumentation and Control Systems 518 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A524	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 07 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A525	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 08 - Electric Power 74 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A526	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 08 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14100A527	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 09 - Auxiliary Systems 331 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A528	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 09 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A529	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 09A - Auxiliary Systems 269 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A530	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 09A Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A531	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 09B - Auxiliary Systems 16 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A532	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 09B Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A533	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 10 - Steam and Power Conversion System 92 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14100A534	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 10 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A536	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 11 - Radioactive Waste Management 144 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A537	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 11 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A539	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 12 - Radiation Protection 349 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A540	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 12 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A541	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 13 - Conduct of Operations 22 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

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ML14100A544	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 13 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A545	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 14 - Initial Test Program 157 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A546	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 14 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A548	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 15 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A550	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 16 - Generic Technical Specifications 286 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A551	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 16 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14100A552	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 16B - GTS Bases 537 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A553	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 16B Change List 2 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A554	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 17 - Quality Assurance 26 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A555	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 17 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A556	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 18 - Human Factors Engineering 65 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A557	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 18 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14100A559	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 19 - Probabilistic Risk Assessment and Severe Accidents 274 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14100A560	GE-Hitachi ESBWR Design Control Document Tier 2, Rev. 10 - Chapter 19 Change List 1 Page(s)	1-Apr-2014	Design Control Document (DCD)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14101A028	ESBWR Standard Plant Design Certification Application Design Control Document, Revision 10, Tier 1 and Tier 2 - Resubmittal. 4 Page(s)	1-Apr-2014	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML12159A531	Updated Economic Simplified Boiling Water Reactor Design Certification Rulemaking Schedule. 6 Page(s)	15-Apr-2014	Letter	NRC/NRO/DAR R	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML14043A134	Advanced Supplemental Safety Evaluation Report For The Economic Simplified Boiling-Water Reactor Standard Plant Design. 91 Page(s)	17-Apr-2014	Draft Safety Evaluation Report (DSER)	NRC/NRO/DNR L/LB3	NRC/NRO/DNR L/LB3	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13301A201	Request for Withholding Information from Public Disclosure (MFN-13-016). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13311A808	Request for Withholding Information from Public Disclosure (MFN-12-059, Revision 2). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13311A898	Request For Withholding Information From Public Disclosure (MFN 12-086, Revision 3). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13298A632	Letter - Request For Withholding Information From Public Disclosure (MFN 13-015). 7 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13317A748	Request For Withholding Information From Public Disclosure (MFN 13-075). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13317B705	Request For Withholding Information From Public Disclosure (MFN 12-043, Revision 3). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML13318A038	Request For Withholding Information From Public Disclosure (MFN 12-065, Revision 2). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13319A923	Request For Withholding Information From Public Disclosure (MFN 12-051, Revision 3). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13253A290	Request For Withholding Information From Public Disclosure (MFN 12-043, Revision 2). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13311A829	Request For Withholding Information From Public Disclosure (MFN 12-077, Revision 3). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13330A838	Request For Withholding Information From Public Disclosure (MFN 12-038, Revision 2). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML13301A201	Request For Withholding Information From Public Disclosure (MFN-13-016). 5 Page(s)	28-Apr-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14099A519	NUREG-1966 Vol 1 "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design (Chapters 1-3) 751 Page(s)	30-Apr-2014	Final Safety Evaluation Report (FSER) NUREG	NRC/NRO		05200010
ML14099A522	NUREG-1966 Vol 2 "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design (Chapters 4-8). 691 Page(s)	30-Apr-2014	Final Safety Evaluation Report (FSER) NUREG	NRC/NRO		05200010
ML14099A532	NUREG-1966 Vol 3 "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design (Chapters 9-15). 753 Page(s)	30-Apr-2014	Final Safety Evaluation Report (FSER) NUREG	NRC/NRO		05200010
ML14100A187	NUREG-1966 Vol 4 "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design (Chapters 16-24). 675 Page(s)	30-Apr-2014	Final Safety Evaluation Report (FSER) NUREG	NRC/NRO		05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML1400A190	NUREG-1966 Vol 5 "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design (Appendix A). 743 Page(s)	30-Apr-2014	Final Safety Evaluation Report (FSER) NUREG	NRC/NRO		05200010
ML1400A194	NUREG-1966 Vol 6 "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design (Appendices B-F). 499 Page(s)	30-Apr-2014	Final Safety Evaluation Report (FSER) NUREG	NRC/NRO		05200010
ML14079A154	Request For Withholding Information From Public Disclosure (MFN-13-091, Supplement 1). 5 Page(s)	5-May-2014	Letter Proprietary Information Review	NRC/NRO/DNRL/LB3	GE-Hitachi Nuclear Energy Americas, LLC	05200010
ML14147A422	Initial Response to NRC's Request for Additional Information Letter Number 1 related to Chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. 2 Page(s)	23-May-2014	Letter	NRC/NRO/DNRL	GE-Hitachi Nuclear Energy Americas, LLC	05200010

Accession Number	Title & Estimated Page Count	Document Date	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
ML14154A094	GEH Request for NRC to Retire the ESBWR Final Design Approval Upon Issuance of the ESBWR Final Design Certification Rule. 2 Page(s)	6-Jun-2014	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14204A848	ESBWR Standard Plant Design Certification Application Design Control Document, Revision 10, Tier 1 and Tier 2 - Resubmittal. 4 Page(s)	7-Jul-2014	Letter; License-Application for Combined License (COLA)	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010
ML14209A117	Economic Simplified Boiling Water Reactor, Design Certification - Supplemental Final Safety Evaluation.	28-Jul-2014	Letter	GE-Hitachi Nuclear Energy Americas, LLC	NRC/Document Control Desk NRC/NRO	05200010

APPENDIX B. REFERENCES

This supplement to Appendix B identifies only those references used by the staff in this supplemental final safety evaluation report.

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Summary of November 1, 2013, Public Meeting to Discuss the U.S Nuclear Regulatory Commission Bulletin 2012-01, “Design Vulnerability in Electric Power System”, (ADAMS Accession No. ML13309B117.)

APPENDIX C. ACRONYMS

ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
ASME	American Society of Mechanical Engineers
BPV	Boiler and Pressure Vessel
B/U	bias errors and uncertainties
C	Celsius
CFD	computational fluid dynamics
CFR	*Code of Federal Regulations*
CRD	control rod drive
CRGT	control rod guide tube
CS	core support
CSDRS	certified seismic design response spectra
DCD	design control document
DCIS	distributed control and information system
EPU	extended power uprate
ESBWR	economic simplified boiling water reactor
F	Fahrenheit
FE	finite element
FEM	finite element model
FIV	flow-induced vibration
FRF	frequency response functions
FSER	final safety evaluation report
GDC	general design criterion
GDCS	gravity-driven cooling system
GEH	General Electric – Hitachi Nuclear Energy
GGNS	Grand Gulf Nuclear Station
GWMS	gaseous waste management system
HF	high frequency
Hz	hertz
IC	isolation condenser
ICGT	incore guide tube
ICMGT	incore monitor guide tube
ITAAC	Inspections, tests, analyses, and acceptance criteria
Kpa	kilopascals
ksi	kilopounds per square inch
LF	low frequency
LOCA	loss-of-coolant accident
LTR	licensing topical report
m/s	meters per second
MASR	minimum alternating stress ration
MCR	main control room
MPa	megapascals
MSL	main steamline
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRO	New Reactors Licensing

OGS	offgas system
OLTP	Originally Licensed Thermal Power
PAT	Power Ascension Test
PBLE	plant-based load evaluation
PSD	power spectra density
psi	Pounds per square inch
RAI	request for additional information
RANS	Reynolds-averaged Navier-Strokes
RAT	reserve auxiliary transformer
RBV	reactor building vibration
RG	regulatory guide
RMS	root mean square
rpm	revolutions per minute
RPV	reactor pressure vessel
RSD	replacement steam dryer
RWCU	reactor water cleanup
SDC	shutdown cooling
SDMP	steam dryer monitoring plan
SF	singularity factors
SER	safety evaluation report
SFSQ	spent fuel seismic qualification
SLC	standby liquid control
SMT	scale model test
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SRV	safety relief valve
SS	stainless steel
SSC	structures, systems, and components
SSE	safe shutdown earthquake
SSES	Susquehanna Steam Electric Station
SV	safety valve
TB	turbine building
UAT	unit auxiliary transformer
VY	Vermont Yankee Nuclear Power Station
VPF	vane passing frequency

APPENDIX D. PRINCIPAL CONTRIBUTORS

This supplement to Appendix D identifies only those contributors to this supplemental final safety evaluation report.

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Robert Fitzpatrick	Electrical Engineering
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James M. Gilmer	Reactor Systems
Tekia Govan	Project Management
Zachary Gran	Health Physics, Dose Consequences
Stacy Joseph	Project Management
Andrea Keim	Quality Assurance
David Misenhimer	Project Management
Bruce Musico	Emergency Preparedness
Richard Pelton	Operator Training, Organization
Paul Pieringer	Human Factors
Mohammad Sadollah	Project Management
Thomas G. Scarbrough	Mechanical Engineering
Terri Spicher	Mechanical Engineering
James Strnisha	Mechanical Engineering
Dinesh Taneja	Instrumentation and Control
George Tartal	Rulemaking
George Thomas	Reactor Systems

Name	Area of Responsibility
Ian Tseng	Mechanical Engineering
Christopher Van Wert	Reactor Systems
Vince Williams	Physical Security
Yuken Wong	Dynamic Analysis and Testing
Jim Xu	Structural Engineering

APPENDIX E. INDEX OF NRC'S REQUESTS FOR ADDITIONAL INFORMATION

This supplement to Appendix E identifies only those requests for additional information used by the staff in this supplemental final safety evaluation report.

RAI Number	Date	ADAMS Accession Number	Author	GEH Letter Number
RAI: 3.9-140				
	November 22, 2006	ML063410346	GEH	MFN 06-464
	October 10, 2006	ML062760404	NRC	
RAI: 3.9-142				
	August 7, 2007	ML072250094	GEH	MFN 06-464, Supplement 3
	November 22, 2006	ML063410346	GEH	MFN 06-464
	October 10, 2006	ML062760404	NRC	
RAI: 3.9-143				
	May 5, 2008	ML081290190	GEH	MFN 08-425
	April 10, 2008	ML080950374	NRC	
	November 29, 2007	ML073390648	GEH	MFN 07-308, Supplement 1
	June 6, 2007	ML071770544	GEH	MFN 07-308
	October 10, 2006	ML062760404	NRC	
RAI: 3.9-147				
	February 24, 2008	ML080370162	GEH	MFN 06-464, Supplement 7
	November 22, 2006	ML063410346	GEH	MFN 06-464
	October 10, 2006	ML062760404	NRC	
RAI: 3.9-148				
	May 20, 2008	ML081430072	GEH	MFN 08-479
	April 10, 2008	ML080950374	NRC	
	December 14, 2007	ML073511765	GEH	MFN 07-652
	October 10, 2006	ML062760404	NRC	
RAI: 3.9-150				
	November 19, 2007	ML073380040	GEH	MFN 06-464, Supplement 4
	November 22, 2006	ML063410346	GEH	MFN 06-464
	October 10, 2006	ML062760404	NRC	
RAI: 3.9-214				
	December 4, 2009	ML093410600	GEH	MFN 09-760
	November 5, 2009	ML093090208	NRC	
	July 6, 2009	ML091890749	GEH	MFN 09-435
	May 26, 2009	ML091400731	NRC	
	January 30, 2009	ML090340672	GEH	MFN 09-069
	July 29, 2008	ML082060534	NRC	

RAI Number	Date	ADAMS Accession Number	Author	GEH Letter Number
RAI: 3.9-245				
	December 2, 2009	ML093380718	GEH	MFN 09-748
	November 5, 2009	ML093090208	NRC	
	July 8, 2009	ML091890963	GEH	MFN 09-438
	May 26, 2009	ML091400731	NRC	
	October 20, 2008	ML082960405	GEH	MFN 08-786
	July 29, 2008	ML082060534	NRC	
RAI: 3.9-269				
	February 7, 2013	ML13038A300	GEH	MFN 12-043, Revision 1
	September 27, 2012	ML12276A174	GEH	MFN 12-043
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-269 S01				
	August 9, 2013	ML13225A280	GEH	MFN 12-043, Revision 2
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-270				
	January 21, 2013	ML13022A535	GEH	MFN 13-003
	September 27, 2012	ML12276A174	GEH	MFN 12-043
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-271				
	August 9, 2013	ML13225A040	GEH	MFN 12-045, Revision 2
	February 8, 2013	ML13039A334	GEH	MRN 12-045, Revision 1
	June 26, 2012	ML121840040	GEH	MFN 12-045
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-271 S01				
	August 9, 2013	ML13225A040	GEH	MFN 12-045, Revision 2
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-272				
	January 24, 2013	ML13025A232	GEH	MFN 12-046, Revision 2
	June 5, 2012	ML12171A094	GEH	MFN 12-046, Revision 1
	May 17, 2012	ML14139A038	GEH	MFN 12-046
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-273				
	June 1, 2012	ML12173A109	GEH	MFN 12-038, Revision 1
	May 7, 2012	ML14135A272	GEH	MFN 12-038
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-273 S01				
	September 18, 2013	ML13262A426	GEH	MFN 12-038, Revision 2
	March 27, 2013	ML13121A173	NRC	

RAI Number	Date	ADAMS Accession Number	Author	GEH Letter Number
RAI: 3.9-274				
	January 24, 2013	ML13024A246	GEH	MFN 12-040, Revision 2
	June 18, 2012	ML14136A301	GEH	MFN 12-040, Revision 1
	May 8, 2012	ML12131A566	GEH	MFN 12-040
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-275				
	February 13, 2013	ML13045A073	GEH	MFN 12-047, Revision 1
	July 25, 2012	ML12209A074	GEH	MFN 12-047
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-276				
	January 23, 2013	ML13024A232	GEH	MFN 12-048, Revision 1
	July 18, 2012	ML12209A070	GEH	MFN 12-048
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-277				
	September 19, 2013	ML13263A127	GEH	MFN 12-086, Revision 3
	February 11, 2013	ML13043A073	GEH	MNF 12-086. Revision 2
	October 5, 2012	ML12284A079	GEH	MFN 12-086, Revision 1
	July 19, 2012	ML12209A072	GEH	MFN 12-086
	May 8, 2012	ML12131A566	GEH	MFN 12-040
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-277 S01				
	September 19, 2013	ML13263A125	GEH	MFN 12-086, Revision 3
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-278				
	January 31, 2013	ML13032A597	GEH	MFN 12-049, Revision 1
	June 25, 2012	ML121840059	GEH	MFN 12-049
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-279				
	January 24, 2013	ML13025A054	GEH	MFN 12-050, Revision 1
	August 13, 2012	ML12228A666	GEH	MFN 12-050
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-280				
	February 15, 2013	ML13051A052	GEH	MFN 12-051, Revision 2
	October 3, 2012	ML12279A102	GEH	MFN 12-051, Revision 1
	June 13, 2012	ML12170B031	GEH	MFN 12-051
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-280 S01				
	September 18, 2013	ML13262A232	GEH	MFN 12-051, Revision 3
	March 27, 2013	ML13121A173	NRC	

RAI Number	Date	ADAMS Accession Number	Author	GEH Letter Number
RAI: 3.9-281				
	June 12, 2012	ML12170B029	GEH	MFN 12-052, Revision 1
	June 5, 2012	ML12159A087	GEH	MFN 12-052
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-282				
	June 5, 2012	ML12171A090	GEH	MFN 12-070
	May 14, 2012	ML121380111	GEH	MFN 12-053
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-283				
	February 14, 2013	ML13046A087	GEH	MFN 12-054, Revision 1
	June 27, 2012	ML12300A097	GEH	MFN 12-054
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-284				
	February 14, 2013	ML13046A160	GEH	MFN 12-055, Revision 2
	October 12, 2012	ML12291A404	GEH	MFN 12-055, Revision 1
	June 5, 2012	ML12159A108	GEH	MFN 12-055
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-285				
	September 24, 2013	ML13268A461	GEH	MFN 12-077, Revision 3
	February 15, 2013	ML13052A161	GEH	MFN 12-077, Revision 2
	October 17, 2012	ML12296A163	GEH	MFN 12-077, Revision 1
	June 19, 2012	ML121730330	GEH	MFN 12-076
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-285 S01				
	September 24, 2013	ML13268A461	GEH	MFN 12-077, Revision 3
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-286				
	September 27, 2013	ML13274A517	GEH	MFN13-075
	February 15, 2013	ML13052A161	GEH	MFN 12-077, Revision 2
	October 17, 2012	ML12296A163	GEH	MFN 12-077, Revision 1
	June 20, 2012	ML121840083	GEH	MFN 12-077
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-286 S01				
	September 27, 2013	ML13274A517	GEH	MFN13-075
	September 24, 2013	ML13267A249	GEH	MFN 12-059, Revision 2
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-287				
	February 14, 2013	ML13046A084	GEH	MFN 12-058, Revision 1
	September 18, 2012	ML12264A066	GEH	MFN 12-058
	May 1, 2012	ML120950046	NRC	

RAI Number	Date	ADAMS Accession Number	Author	GEH Letter Number
RAI: 3.9-288				
	February 8, 2013	ML13039A304	GEH	MFN 12-059, Revision 1
	October 9, 2012	ML12285A106	GEH	MFN 12-059
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-288 S01				
	September 24, 2013	ML13267A249	GEH	MFN 12-059, Revision 2
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-289				
	September 25, 2013	ML13269A176	GEH	MFN 12-066, Revision 3
	January 30, 2013	ML13031A478 ML13031A479	GEH	MFN 12-066, Revision 2
	June 7, 2012	ML12171A098	GEH	MFN 12-066, Revision 1
	May 25, 2012	ML12152A064	GEH	MFN 12-066
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-290				
	September 25, 2013	ML13269A176	GEH	MFN 12-066, Revision 3
	January 30, 2013	ML13031A478 ML13031A479	GEH	MFN 12-066, Revision 2
	June 7, 2012	ML12171A098	GEH	MFN 12-066, Revision 1
	May 25, 2012	ML12152A064	GEH	MFN 12-066
	May 10, 2012	ML121380119	GEH	MFN 12-061
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-291				
	September 25, 2013	ML13269A176	GEH	MFN 12-066, Revision 3
	January 30, 2013	ML13031A478 ML13031A479	GEH	MFN 12-066, Revision 2
	June 7, 2012	ML12171A098	GEH	MFN 12-066, Revision 1
	May 25, 2012	ML12152A064	GEH	MFN 12-066
	May 7, 2012	ML14135A272	GEH	MFN 12-038
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-291 S01				
	September 25, 2013	ML13269A176	GEH	MFN 12-066, Revision 3
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-291 S02				
	September 25, 2013	ML13269A176	GEH	MFN 12-066, Revision 3
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-291 S03				
	September 25, 2013	ML13269A176	GEH	MFN 12-066, Revision 3
	March 27, 2013	ML13121A173	NRC	

RAI Number	Date	ADAMS Accession Number	Author	GEH Letter Number
RAI: 3.9-291 S04				
	September 25, 2013	ML13269A176	GEH	MFN 12-066, Revision 3
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-291 S05				
	September 25, 2013	ML13269A176	GEH	MFN 12-066, Revision 3
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-292				
	February 19, 2013	ML13053A303	GEH	MFN 13-007
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-292 S01				
	September 26, 2013	ML13270A193	GEH	MFN 13-046
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-292 S02				
	September 26, 2013	ML13270A271	GEH	MFN 13-007, Revision 1
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-292 S03				
	September 27, 2013	ML13275A050	GEH	MFN 13-082
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-293				
	February 8, 2013	ML13042A071	GEH	MFN 12-065, Revision 1
	August 31, 2012	ML12248A292	GEH	MFN 12-065
	May 1, 2012	ML120950046	NRC	
RAI: 3.9-293 S01				
	September 18, 2013	ML13261A400	GEH	MFN 12-065, Revision 2
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-293 S02				
	September 18, 2013	ML13261A400	GEH	MFN 12-065, Revision 2
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-293 S03				
	September 18, 2013	ML13261A400		MFN 12-065, Revision 2
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-294				
	August 9, 2013	ML13225A039	GEH	MFN 13-014
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-295				
	September 18, 2013	ML13261A405	GEH	MFN 13-015
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-296				
	September 18, 2013	ML13262A236	GEH	MFN 13-016
	March 27, 2013	ML13121A173	NRC	

RAI Number	Date	ADAMS Accession Number	Author	GEH Letter Number
RAI: 3.9-297				
	September 18, 2013	ML13261A517	GEH	MFN 13-017
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-298				
	September 18, 2013	ML13262A235	GEH	MFN 13-018
	March 27, 2013	ML13121A173	NRC	
RAI: 3.9-299				
	December 6, 2013	ML13344B156	GEH	MFN 13-091
	November 13, 2013	ML13316B654	NRC	
RAI: 3.9-300				
	December 6, 2013	ML13344B156	GEH	MFN 13-091
	November 13, 2013	ML13316B654	NRC	
RAI: 3.9-301				
	December 6, 2013	ML13344B156	GEH	MFN 13-091
	November 13, 2013	ML13316B654	NRC	
RAI: 3.9-302				
	December 6, 2013	ML13344B156	GEH	MFN 13-091
	November 13, 2013	ML13316B654	NRC	
RAI: 3.9-303				
	December 6, 2013	ML13344B156	GEH	MFN 13-091
	November 13, 2013	ML13316B654	NRC	
RAI: 4.5-19				
	June 16, 2006	ML061740336	GEH	MFN 06-178
	May 17, 2006	ML061360248	NRC	
RAI: 8.1-22				
	November 20, 2013	ML13324A487	GEH	MFN 13-040, Supplement 2
	August 28, 2013	ML13242A212	GEH	MFN 13-040, Supplement 1
	July 30, 2013	ML13211A207	GEH	MFN 13-040
	July 5, 2013	ML13183A252	NRC	

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)

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

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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.

<u>ESBWR Design Description</u>

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.

<u>Safety Enhancement Features</u>

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.

<u>Level of Detail for DAC/ITAAC</u>

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>:

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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

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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

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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

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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NRC FORM 335 (12-2010) NRCMD 3.7 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-1966, Supplement 1
2. TITLE AND SUBTITLE Supplemental Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design	3. DATE REPORT PUBLISHED MONTH: September YEAR: 2014 4. FIN OR GRANT NUMBER
5. AUTHOR(S) See Appendix D	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates)

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report supplements the final safety evaluation report (FSER) for the Economic Simplified Boiling-Water Reactor (ESBWR) standard plant design. The FSER was issued by the U.S. Nuclear Regulatory Commission (NRC) as NUREG-1966 in April 2014 to document the NRC staff's technical review of the ESBWR design. The application for the ESBWR design was submitted on August 24, 2005, by General Electric-Hitachi (GEH) in accordance with Subpart B, "Standard Design Certifications," of 10 CFR Part 52. This supplement documents the NRC staff's review of GEH's changes to the ESBWR design documentation in the design control document (DCD) since the issuance of the FSER. On the basis of the evaluation described in the ESBWR FSER (NUREG-1966) and this report, the NRC staff concludes that the changes to the DCD (up to and including Revision 10 to the ESBWR DCD) are acceptable and that GEH's application for design certification meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the ESBWR standard Plant design.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
Advanced Reactor, Advanced Light Water Reactor (ALWR) Final Safety Evaluation Report Passive Plant Standardization Combined License (COL) Design Certification Regulatory Treatment of Non-Safety-Related Systems (RTNSS) Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)	unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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NUREG-1966
Supplement 1

Final Safety Evaluation Report Related to Certification of the
Economic Simplified Boiling-Water Reactor Standard Design

September 2014