FEDERAL REGISTER

Vol. 79 Wednesday,
No. 199 October 15, 2014

Part II

Nuclear Regulatory Commission

10 CFR Part 52
Economic Simplified Boiling Water Reactor Design Certification; Final Rule

Nuclear Regulatory Commission

10 CFR Part 52

[NRC–2010–0135]

RIN 3150–AI85

Economic Simplified Boiling-Water Reactor Design Certification

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is adopting a new rule certifying the Economic Simplified Boiling-Water Reactor (ESBWR) standard plant design. This action is necessary so that applicants or licensees intending to construct and operate an ESBWR design may do so by referencing this design certification rule (DCR). The applicant for certification of the ESBWR design is GE-Hitachi Nuclear Energy (GEH).

DATES: This final rule is effective on November 14, 2014. The incorporation by reference of certain publications listed in this regulation is approved by the Director of the Office of the Federal Register (OFR) as of November 14, 2014.

ADDRESSES: Please refer to Docket ID NRC–2010–0135 when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

• Federal Rulemaking Web site: Go to *http://www.regulations.gov* and search for Docket ID NRC–2010–0135. Address questions about NRC dockets to Carol Gallagher, telephone: 301–287–3422; email: *Carol.Gallagher@nrc.gov.* For technical questions, contact the individuals listed in the **FOR FURTHER INFORMATION CONTACT** section of this document.

• NRC's Agencywide Documents Access and Management System (ADAMS): You may obtain publicly available documents online in the ADAMS Public Documents collection at *http://www.nrc.gov/reading-rm/adams.html.* To begin the search, select ''ADAMS Public Documents'' and then select ''Begin Web-based ADAMS Search.'' For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1–800–397–4209, 301–415–4737, or by email to *pdr.resource@nrc.gov.* For the convenience of the reader, instructions about obtaining materials referenced in this document are provided in a table in Section VII, ''Availability of Documents,'' of this document.

• NRC's PDR: You may examine and purchase copies of public documents at the NRC's PDR, Room O1–F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

FOR FURTHER INFORMATION CONTACT: George M. Tartal, Office of New Reactors, telephone: 301–415–0016, email: *George.Tartal@nrc.gov;* or David Misenhimer, Office of New Reactors, telephone: 301–415–6590, email: *David.Misenhimer@nrc.gov;* U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

SUPPLEMENTARY INFORMATION:

Executive Summary

A. Need for the Regulatory Action

The NRC is amending its regulations related to licenses, certifications, and approvals for nuclear power plants. This final rule certifies the ESBWR standard plant design. This action is necessary so that applicants or licensees intending to construct and operate an ESBWR design may do so by referencing this DCR.

B. Major Provisions

Major provisions of the final rule include changes to:

• specify which documents contain the requirements for the ESBWR design,

• specify how a nuclear power plant license applicant can reference the ESBWR design,

• describe how the NRC considers matters within the scope of the design to be resolved for proceedings involving a license or application referencing the ESBWR design, and

• describe the processes for changes to and departures from the ESBWR design.

C. Costs and Benefits

The NRC did not prepare a regulatory analysis to determine the expected quantitative or qualitative costs and benefits of the final rule. The NRC prepares regulatory analyses for rulemakings that establish generic regulatory requirements applicable to all licensees. Design certifications are not generic rulemakings in the sense that design certifications do not establish standards or requirements with which all licensees must comply. Rather, design certifications are NRC approvals of specific nuclear power plant designs by rulemaking, which then may be voluntarily referenced by an applicant for a combined license (COL). Furthermore, design certification rulemakings are initiated by an applicant for a design certification, rather than the NRC. Preparation of a regulatory analysis in this circumstance would not be useful because the design to be certified is proposed by the applicant rather than the NRC. For these reasons, the NRC concludes that preparation of a regulatory analysis is neither required nor appropriate.

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I. Background

Part 52 of Title 10 of the *Code of Federal Regulations* (10 CFR), ''Licenses, Certifications, and Approvals for Nuclear Power Plants,'' subpart B, presents the process for obtaining standard design certifications. On August 24, 2005, GEH tendered its application for certification of the ESBWR standard plant design (ADAMS Accession No. ML052450245) with the NRC. The NRC published a notice of receipt of the application in the **Federal Register** (70 FR 56745; September 28, 2005). GEH submitted this application in accordance with subpart B of 10 CFR part 52. On December 1, 2005, the NRC formally accepted the application as a docketed application for design certification (Docket No. 52–010) (70 FR 73311; December 9, 2005). The pre-application information submitted before the NRC formally accepted the application can be found in ADAMS under Docket No. PROJ0717 (Project No. 717).

The NRC staff issued a final safety evaluation report (FSER) for the ESBWR design in March 2011. The FSER is available in ADAMS under Accession No. ML103470210. The NRC subsequently published the FSER in April 2014 as NUREG–1966, ''Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design'' (ADAMS Accession No. ML14100A304). The NRC also published a proposed rule to certify the ESBWR design in the **Federal Register** on March 24, 2011 (76 FR 16549), and a supplemental proposed rule on May 6, 2014 (79 FR 25715). The FSER and the proposed rule were based on the NRC's review of Revision 9 of the ESBWR DCD.

On April 17, 2014, the NRC issued an advanced supplemental safety evaluation report (SER) (ADAMS Accession No. ML14043A134) to address several matters identified by the NRC and revisions to the ESBWR DCD in Revision 10. The advanced supplemental SER was referenced in the supplemental proposed rule (79 FR 25715; May 6, 2014). The supplemental FSER will be published as Supplement No. 1 to NUREG–1966 before this final rule becomes effective. Because Revision 10 of the DCD was issued after the ESBWR proposed rule was published, all of the substantive changes in Revision 10 of the DCD are addressed in the **SUPPLEMENTARY INFORMATION** section of this document, including a discussion of why the change was or was not addressed in a supplemental proposed rule.

In its application for design certification, GEH also requested the NRC to provide an SDA for the ESBWR design. An SDA for the ESBWR design was issued in March 2011 (ADAMS Accession No. ML110540310) following the NRC staff's issuance of the ESBWR FSER. On June 3, 2014, GEH requested that the NRC retire the SDA at the time of issuance of the final ESBWR design certification rule (ADAMS Accession No. ML14154A094). After this final rule is published, the NRC intends, as a separate action from this rulemaking, to withdraw the SDA.

The application for design certification of the ESBWR design has been referenced in the following COL applications as of the date of this document: (1) Detroit Edison Company, Fermi Unit 3, Docket No. 52–033 (73 FR 73350; December 2, 2008); (2) Dominion Virginia Power, North Anna Unit 3, Docket No. 52–017 (73 FR 6528; February 4, 2008); (3) Entergy Operations, Inc., Grand Gulf Unit 3, Docket No. 52–024 (73 FR 22180; April 24, 2008) (APPLICATION SUSPENDED); (4) Entergy Operations, Inc., River Bend Unit 3, Docket No. 52–036 (73 FR 75141; December 10, 2008) (APPLICATION SUSPENDED); and (5) Exelon Nuclear Texas Holdings, LLC, Victoria County Station Units 1 and 2, Docket Nos. 52–031 and 52–032 (73 FR 66059; November 6, 2008) (APPLICATION WITHDRAWN).

II. Summary and Analysis of Public Comments on the ESBWR Proposed Rule and Supplemental Proposed Rule

A. Overview of Public Comments

The NRC published a proposed rule to certify the ESBWR design in the **Federal Register** on March 24, 2011 (76 FR 16549). The period for submitting comments on the proposed DCR, ESBWR DCD, or draft environmental assessment (EA) closed on June 7, 2011. The NRC received a total of 10 public comments on the proposed rule. The types of comments, the organization of comments, the comment identification format, and comment responses follow.

The NRC also published a supplemental proposed rule to request public comments on two specific topics regarding the ESBWR design certification. The supplemental proposed rule was published in the **Federal Register** on May 6, 2014 (79 FR 25715). The period for submitting comments on these specific topics closed on June 5, 2014. The NRC received no public comments on the supplemental proposed rule.

Types of Comments

The NRC received two types of comment submissions on the proposed rule for the ESBWR design certification. A comment submission means a communication or document, submitted to the NRC by an individual or entity, with one or more individual comments addressing a subject or an issue. The two types of comment submissions were:

1. Comment submissions that were not identical or similar in content (unique comment submissions); and
2. Comment submissions self-characterized as ''petitions'' or comment submissions related to such ''petitions'' (petitions).

The NRC received four unique comment submissions, including three comment submissions from private citizens and one comment submission from a non-government organization. Table 1 provides summary information on the unique comment submissions and their ADAMS Accession numbers.

In addition, in light of the Fukushima Dai-ichi accident and during the public comment period on the proposed rule, the NRC received a series of petitions to suspend adjudicatory, licensing, and rulemaking activities, including the ESBWR design certification rulemaking. The NRC subsequently authorized responsive and supplemental filings on these petitions. In its *Memorandum and Order,* CLI–11–05, September 9, 2011, 74 NRC 141 (2011) (this decision is available on the NRC Web site in Volume 74 at *http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0750/*), the Commission addressed the petitions and the responsive and supplemental filings and determined that the petitions should be denied in the relevant adjudicatory proceedings; and, on its own motion referred the petitions to the NRC staff for consideration as comments in the ESBWR rulemaking. The staff considered the petitions and the responsive and supplemental filings and identified six comment submissions applicable to the ESBWR rulemaking. Table 2 provides summary information on these ''petition-related'' comment submissions and their ADAMS Accession numbers. Four of those comment submissions were ''petitions'' filed during the public comment period. One of the comment submissions was a responsive filing to the ''petitions.''

The sixth of these comment submissions, self-characterized as a ''petition'' and referred to the NRC staff in CLI–11–05, was received on August 15, 2011, after the close of the public comment period. As stated in the proposed rule, comments received after June 7, 2011, ''will be considered if it is practical to do so, but assurance of consideration cannot be given'' to comments received after this date. The NRC determined that it was practical to consider this comment. This comment opposed issuance of the final ESBWR rule.

TABLE 1—UNIQUE COMMENT SUBMISSIONS

Comment submission No.	Commenter	ADAMS Accession No.
1	Paul Daugherty	ML110880057
2	Farouk Baxter	ML110880315
3	Patricia T. Birnie, Chairman, General Electric Stockholders' Alliance	ML11158A088
4	Anonymous	ML11187A303

TABLE 2—COMMENT SUBMISSIONS SELF-CHARACTERIZED AS PETITIONS AND RESPONSIVE FILINGS

Comment submission No.	Commenter	ADAMS Accession No.
1 (Note 1)	Various organizations and individuals	ML111040472
2 (Note 1)	Various organizations and individuals	ML111080855
3	Various organizations and individuals	ML111100618
4	Jerald G. Head, Senior VP, Regulatory Affairs, GE Hitachi Nuclear Energy	ML11124A103
5	Various organizations and individuals	ML111260637
6	ESBWR Intervenors	ML112430118

Note 1: Petition comment submission 2 was submitted as an amendment to petition comment submission 1. Therefore, the NRC is only addressing comments on petition comment submission 2 in this final rule and no further response is needed on petition comment submission 1.

Organization of Comments and Responses

Comments and the NRC's responses are organized into two categories: Comments on technical issues presented in the DCD, and comments regarding Fukushima lessons learned. Comments on technical issues include the inclusion of beyond-design-basis accidents into the design, design of the ancillary diesel generators, safety-related battery design, control rod drive design, and control room flood protection. Comments regarding Fukushima lessons learned include delaying certification of the ESBWR design until lessons learned have been incorporated and the NRC's obligation under the National Environmental Policy Act (NEPA) to evaluate new information (such as the NTTF report, ADAMS Accession No. ML111861807) relevant to the environmental impact of its actions prior to certifying the ESBWR design. The NRC received comments related to the draft EA for this rule but those comments did not include anything to suggest that: (i) A rule certifying the ESBWR standard design would be a major Federal action, or (ii) the severe accident mitigation design alternatives (SAMDA) evaluation omitted a design alternative that should have been considered or incorrectly considered the costs and benefits of the alternatives it did consider. Therefore, no change to the EA was warranted. The NRC received no comments on the two specific topics in the supplemental proposed rule. The detailed comment summaries and the NRC's responses are provided in Sections II.B and II.C of this document.

Comment Identification Format

All comments are identified uniquely by using the format [W][X]–[Y], where:

[W] represents the comment submission type (S = unique comment submission, P = petition).

[X] represents the comment submission identification number (refer to the comment submission tables).

[Y] represents the comment number, which the NRC assigned to the comment. In some instances, lower-case alphabetic characters [Ya, Yb, Yc * * *] were added to a comment number after the initial designation of comments.

The NRC has created a document (ADAMS Accession No. ML113130141) which compiles all comment submissions and annotates each comment submission with the comment number indicated in the right hand margin.

B. Comments Regarding Technical Content in the DCD

Design-Basis Accidents

Comment: Beyond-Design-Basis Accidents (DBAs) should be included in the design, final safety analysis report (FSAR), and Technical Specifications (TS). (S1–1)

NRC Response: The NRC agrees that beyond-DBAs should be considered in the ESBWR design and the FSAR. In its 1985 policy statement on severe accidents (50 FR 32138), the Commission defined the term ''severe accident'' as an event that is ''beyond the substantial coverage of design basis events,'' (DBE) including events in which there is substantial damage to the reactor core (whether or not there are serious offsite consequences). Consistent with the objectives of standardization and early resolution of design issues, 10 CFR 52.47(a)(23) requires applicants for design certification to include a description and analysis of severe accident prevention and mitigation features in the new reactor designs. These features are discussed in Chapter 19 of the DCD (equivalent to an FSAR), and the staff's evaluation of them is found in Chapter 19 of the FSER.

The NRC disagrees that beyond-DBAs should be included in the TS. The TS prescribe safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, and administrative controls associated with DBEs, but need not prescribe limits or settings for conditions that could be experienced during a beyond-DBE.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: The NRC's current regulatory scheme requires significant re-evaluation and revision in order to expand or upgrade the design-basis for reactor safety as recommended by its NTTF report. (P6–1)

NRC Response: The NRC considers this comment to be outside the scope of the ESBWR design certification rulemaking. The comment deals with the adequacy of the NRC's overall regulatory scheme for nuclear power reactors and does not directly address the adequacy of the ESBWR design certification.

Nonetheless, the NRC disagrees with the comment. The NRC's rules and regulations provide reasonable assurance of adequate protection of public health and safety and the common defense and security. However, the Commission has ''initiated a comprehensive examination of the implications of the Fukushima accident. . . . As a result [of that examination], the NRC may implement changes to its regulations and regulatory processes.'' CLI–11–05, 74 NRC at 168. If such changes are warranted, the NRC's ''regulatory processes provide sufficient time and avenues to ensure that design certifications and COLs satisfy any Commission-directed changes before any new power plant commences operations. . . . Whether [the Commission] adopt[s] the Task Force recommendations or require[s] more, or different, actions associated with certified designs or COL applications, [the Commission has] the authority to ensure that certified designs and combined licenses include appropriate Commission-directed changes before operation.'' *Id.* at 162–163.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: The ESBWR environmental documents do not address the radiological consequences of DBAs or demonstrate that those reactors can be operated without undue risk to the health and safety of the public and conclude that any health effects resulting from the DBAs are negligible. This conclusion is based on a review of the DBAs considered in the ESBWR DCD (WEC 2008) and NUREG–0800, Standard Review Plan (SRP). The findings of the Fukushima NTTF report call into question whether this represents a full, accurate description and examination of all DBAs having the potential for releases to the environment. See Makhijani Declaration at 7. If the design-basis for the reactors does not incorporate accidents that should be considered in order to satisfy the adequate protection standard, then it is not possible to reach a conclusion that the design of the reactor adequately protects against accident risks. See Makhijani Declaration at 9. (P6–3)

NRC Response: The NRC disagrees with this comment. The NRC notes that the Makhijani Declaration citations do not address DBAs as discussed in the comment, but rather the declaration specifically refers to beyond-DBEs. The NRC interprets the comment to be referring to the environmental report required to be provided by the design certification applicant per 10 CFR 52.47, ''Contents of applications; technical information,'' and 10 CFR 51.55, ''Environmental report—standard design certification.'' The environmental report (NEDO–33306; ADAMS Accession No. ML102990433) referenced in Chapter 19 of the ESBWR DCD and evaluated in Chapter 19 of the FSER, as well as the NRC's EA, addresses costs and benefits of severe accident mitigation design alternatives. Conversely, DBAs for the ESBWR, and their associated radiological consequences, are not addressed in the environmental report but rather are addressed in Chapter 15 of the ESBWR DCD and evaluated in Chapter 15 of the FSER. The environmental report addresses the costs and benefits of severe accident mitigation design alternatives but does not address the design basis accidents discussed in the comment. In any event, the Commission has stated that, if warranted and after ''a comprehensive examination of the implications of the Fukushima accident . . ., the NRC may implement changes to its regulations and regulatory processes.'' CLI–11–05, 74 NRC at 168. The NRC's ''regulatory processes provide sufficient time and avenues to ensure that design certifications and COLs satisfy any Commission-directed changes before any new power plant commences operations. . . .'' *Id.* at 162–163.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Electrical Systems

Comment: The ESBWR design is flawed because it has failed to comply with the requirements of Institute of Electrical and Electronics Engineers (IEEE) Standard 603, which requires the electrical portion of the safety systems that perform safety functions—specifically, alternating current (ac) power from the Ancillary Diesel Generators (ADGs)—be classified as Class 1E. The DCD acknowledges that ac power from the ADGs is not needed for the first 72 hours of an accident, but are needed to perform Class 1E functions (recharging the Class 1E direct current (dc) batteries that provide power during the first 72 hours of an accident) when no other sources of power are available. The ESBWR design has classified these ac power sources as commercial grade, nonsafety-related, and non-Class 1E (S2–1, referencing ADAMS Accession No. ML102350160).

NRC Response: The NRC disagrees with the comment. The NRC's position remains as stated in the separate correspondence between the commenter and the NRC that is attached to the comment letter. Specifically, the NRC stated that the events described in the commenter's previous letters (no ac power available to the plant for 72 hours after initiation of the accident and all batteries are depleted) are not DBEs but are beyond design-basis, for which the requirements of IEEE Standard 603 do not apply. As stated in the staff requirements memorandum (SRM), dated January 15, 1997, concerning SECY–96–128, ''Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design,'' dated June 12, 1996, the Commission approved Item IV—Post-72 Hour Actions. The approval specified that the post-72 hour systems, structures, and components (SSCs) are not required to be safety-related. In addition, as stated in NUREG–1242, Volume 3, Part 1, ''NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document: Passive Plant

Designs, Chapter 1,'' August 1994, a passive advanced light-water reactor, such as the ESBWR design, need not include or rely upon an active safety-related ac power source to support safety system functions after 72 hours from the onset of an accident, but may rely on electrical power sources that are not safety-related after that time. Specifically, the ESBWR is designed so that safety-related passive systems are able to perform all safety functions for 72 hours after initiation of a DBE without the need for operator actions. The DBE is assumed to be resolved (except for long-term cooling) within 72 hours, and thus, the Class 1E batteries are designed for and need only function for 72 hours without being recharged.

In the ESBWR, the ADGs, which are the subject of the commenter's concern, are not used to recharge the Class 1E batteries. Rather, the ADGs provide power directly to post accident monitoring instrumentation, main control room lighting, the reactor pressure vessel (RPV) makeup pump, and containment cooling systems, among others. After 72 hours, consistent with NUREG–1242, nonsafety-related systems other than the ADGs are used to replenish safety-related passive systems so that they will perform long-term core cooling and containment integrity functions. These nonsafety-related systems are designed in accordance with quality standards commensurate with the importance of these functions and that provide reasonable assurance they will function when needed. In the event that the ADGs are not available, the Seismic Category I firewater storage tanks and Seismic Category I diesel pump and fire protection piping can be used to provide post-accident makeup water to the Isolation Condenser and Passive Containment Cooling System (PCCS) pools and Spent Fuel Pool (SFP) using the Fuel and Auxiliary Plant Cooling System (FAPCS) for long-term cooling beyond 72 hours.

The NRC also stated in its May 15, 2009, letter (in the referenced document) that the offsite power system, a nonsafety-related power source, is the preferred source of power for safety-related systems at all current plants. Further, the station blackout (SBO) rule, 10 CFR 50.63, ''Loss of all alternating current power,'' does not require the use of safety-related alternative ac power sources to cope with an SBO. Therefore, neither of these ac power sources—offsite power or alternate ac power source—is required to be safety-related or classified as Class 1E under IEEE 603. Thus, the ADGs need not be classified as Class 1E power sources as well.

In summary, the design bases of the passive safety systems are centered on the 72-hour capability and these safety-related systems must remain functional to assure the integrity of the reactor coolant pressure boundary and the capability to shut down the reactor and maintain it in a safe shutdown condition without operator action or support from nonsafety systems for the first 72 hours following the initiation of a DBE. Beyond 72 hours, these systems must continue to remain functional to provide such assurance for the following 4 days, with allowance for operator actions and support from nonsafety SSCs consistent with NUREG–1242.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: The NRC should require GEH to relocate the safety-related dc batteries and their related systems above grade level so that they are not subject to external flooding. This recommendation is supported by the following points:

1. There is a fair chance of a failure of the dc supply as safety-related battery banks (Class-1E grade batteries) are housed below grade in the reactor building, as well as their electrical penetration to primary containment. In a natural disaster they may not remain watertight, as water may enter through the doors and incapacitate the battery banks.

2. Water may also enter the battery rooms if those doors are open for maintenance, testing, or replacement of cells.

3. ESBWR emergency core cooling systems (ECCS) are dependent on this dc supply. If the dc supply is lost, emergency cooling and depressurization systems will fail. There is no diversity for the core cooling and depressurization systems if the dc supply fails. (S4–1)

NRC Response: The NRC disagrees with the comment. The safety-related dc batteries and their related systems do not need to be relocated above grade level. The NRC has reviewed the ESBWR DCD and has determined that the ESBWR safety-related SSCs (including the reactor building, which houses the dc batteries) are designed to withstand the effects of external flooding. With the exception of loads due to hurricane winds and wind-generated missiles beyond those considered in the ESBWR DCD, the NRC concluded that the ESBWR DCD meets the requirements of 10 CFR part 50, appendix A, ''General Design Criteria for Nuclear Power Plants,'' (GDC) 2, which requires the design bases of SSCs important to safety to include protection against natural phenomena (including earthquakes, tornadoes, floods, hurricanes, and tsunami) such that these SSCs will not lose the capability to perform their safety functions as a result of such phenomena. This conclusion is documented in the NRC's FSER for the ESBWR design.

In the following paragraphs, the NRC addresses each of the three supporting points for the comment.

Supporting Point 1: The NRC agrees that safety-related batteries are located below grade per the ESBWR DCD, Tier 2, Figure 1.2–2. This is acceptable because all components of safety-related dc electric systems are housed in structures which provide protection against external flood damage. The structures that may be subjected to a design-basis flood are designed to withstand the flood level by locating the plant grade elevation 1 ft. (0.30 m) above the flood level and incorporating structural provisions into the plant design to protect the SSCs from the postulated flood conditions. GEH's application for design certification was submitted with proposed vendor-specified site parameters. These values are provided in Table 2.0–1 (Tier 2) and in Table 5.1–1 (Tier 1) of the DCD. For the ESBWR design, the maximum groundwater level is 2 ft. (0.61 m) below plant grade and the maximum flood level is 1 ft. (0.30 m) below plant grade. The ESBWR design was evaluated using the vendor-specified flood levels and found to be safe. All exterior access openings are above flood level. The flood design incorporates reinforced concrete walls designed to resist the static and dynamic forces of the design-basis flood and water stops at construction joints to prevent in-leakage. External surfaces below flood and ground water levels are waterproofed. Penetrations are sealed and also capable of withstanding the static and dynamic forces of the design-basis flood. Watertight doors provide physical separation of flood zones. In addition, the applicant has specified the site parameters, design characteristics, and any additional requirements and restrictions necessary for a COL applicant to ensure that safety-related SSCs will be adequately protected from the site-specific probable maximum flood conditions. Based on the evaluation in Section 3.4 of the FSER, the NRC concludes that the ESBWR design regarding flood protection provides reasonable assurance that safety-related SSCs (including the safety-related dc batteries and their related systems) will maintain their structural integrity or are located within structures that will maintain their integrity, and will perform their intended safety functions when subjected to a design-basis flood, and therefore, satisfy the requirements of GDC 2.

Supporting Point 2: The comment stated that water may enter the battery rooms if the watertight doors are open for maintenance, testing, or replacement of the battery cells. The NRC agrees that this scenario is possible for one division of safety-related battery banks. The ESBWR TS, under limiting condition of operation 3.8.1, restricts maintenance, testing, or replacement of the battery cells during plant operation to only one required division of safety-related battery banks. In addition, the COL applicant is required to develop plant operating and maintenance procedures that provide control for activities that are important to the safe operation of the facility, including limiting conditions of operation. However, there are four divisions of safety-related battery banks, which are physically separated by concrete walls and watertight doors. Only two divisions of dc systems are required for safe shutdown of the plant. If one of the safety-related battery room doors is open during a flood, as suggested in the comment, the other batteries will still be adequately protected by design features for physical separation to ensure the safety-related SSCs can perform their functions.

Supporting Point 3: The comment stated that the ESBWR ECCS is dependent on dc power, and if dc power is lost, emergency cooling and depressurization systems will fail. The ESBWR ECCS consists of the Gravity Driven Cooling System, the Isolation Condenser System, the Standby Liquid Control System, and the Automatic Depressurization System. The Gravity Driven Cooling System, Standby Liquid Control System, and the Automatic Depressurization System do rely on dc power for actuation (as pointed out in the comment). The four trains of Isolation Condenser System, on the other hand, automatically begin removal of decay heat and control RPV level above the top of active fuel upon loss of all ac and dc power because the only valve in the system relied upon to change position upon initiation of the system fails in the safe (open) position upon loss of power. Beginning 4 hours after the start of an accident, the Isolation Condenser System upper and lower header vent valves are opened periodically to remove non-condensable gases to maintain optimum heat removal and allow continued reactor cooldown. These valves are solenoid-operated valves and rely upon electric power to open.

The comment also suggests that there is no diversity for several systems that rely on the dc power supply. The NRC agrees that the Automatic Depressurization System, Gravity Driven Cooling System, the Suppression Pool Equalization Line Valves, and the Standby Liquid Control System all require safety-related dc power in order to perform their safety functions and therefore lack diversity in that regard, but does not agree that the Basemat Internal Melt Arrest Coolability (BiMAC) cooling system requires safety-related dc power to perform its safety function. As discussed below, the BiMAC cooling system—a non-safety system—is designed to automatically fire squib valves and drain water to the area below the RPV upon sensing high temperatures in the BiMAC without dependence on any of the four safety-related power sources. Also, as discussed above, the four trains of the Isolation Condenser System automatically begin removal of decay heat and control RPV level above the top of active fuel upon loss of all ac and dc power because the only valve in the system relied upon to change position upon initiation of the system fails in the safe (open) position upon loss of power. Decay heat can be removed with the Isolation Condenser System for 72 hours without any additional action. The ESBWR is designed such that the Isolation Condenser System heat exchanger pool can be replenished after 72 hours with the diesel driven fire pump to allow continued cooling with the Isolation Condenser System. Safety-related dc power is not needed to operate this pump. In light of these facts, the NRC concludes that the capability of the ESBWR to remove decay heat from the reactor core following an accident is sufficiently diverse. It should also be noted that the ESBWR safety-related 120 volts ac uninterruptible power supply (UPS) input is normally supplied by offsite power or a nonsafety-related onsite power system. During a loss of offsite and nonsafety-related onsite power, the UPS gets its power from 250 volts dc batteries. The ESBWR design includes an offsite power system, nonsafety-related standby diesel generators, and ADGs, any of which can mitigate the consequences of an accident if available. Safety-related UPS systems are housed in seismic Category I structures and meet GDCs 2, 4, and 17.

Common cause failure of the safety-related batteries in the ESBWR design would clearly be an event of substantial safety significance because dc power is used to power the distributed control and instrumentation system, which is used to actuate passive safety systems. However, the ESBWR design includes a number of defense-in-depth features for reducing the likelihood of losing all ability to accomplish key safety functions. As previously stated, the Isolation Condenser System automatically begins removal of decay heat and controls RPV level above the top of active fuel upon loss of all ac and dc power. All safety divisions (including concrete walls and watertight doors that separate the four safety-related battery banks) are physically separated.

The ESBWR design also includes design features specifically for the purpose of injecting water into the containment to flood the containment floor and cover core debris. The BiMAC cooling system is designed to automatically fire squib valves and drain water to the area below the RPV upon sensing high temperatures in the BiMAC, indicating core debris below the RPV. This occurs without operator action and without dependence on any of the four safety-related power sources.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Control Rod Drive System

Comment: Two Control Rod Drives (CRD) are scrammed by one hydraulic control unit (HCU). A single failure of one HCU will affect the scram function of two CRDs. It is done for cost saving. This is not acceptable in a safety system. (S4–2)

NRC Response: The NRC disagrees with the comment. In Section 4.6.3 of the FSER, the NRC stated that a single failure in an HCU may result in the failure of two control rods. The DCD describes that the control rods are assigned to HCUs in a manner such that no 4X4 array of rods contain both rods connected to the same HCU. This arrangement assures that shutdown is achieved (among other things) assuming a single failure of an HCU. The NRC reviewed the effects of an HCU failure and concluded in Section 4.3 of the FSER that sufficient shutdown margin exists in the case of an HCU failure. In addition, TS 3.1.5 requires that all control rod scram accumulators are operable during Modes 1 (Power Operation) and 2 (Start-Up). If an accumulator is inoperable, the associated control rod pair is declared inoperable and Limiting Condition of Operation (LCO) 3.1.3, Control Rod Operability, is entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying the intended function in accordance with actions of LCO 3.1.3. If an accumulator is inoperable, TS require the affected control rod to be inserted and hence the scram function of two CRDs is satisfied. Finally, the ESBWR has a diverse method to scram the reactor. An electric motor is provided for each CRD for scram in addition to the hydraulic scram using the accumulator. Accordingly, the NRC has determined that the CRD system design is adequate.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Control Room

Comment: For safety reasons, the Control Room should be located at a sufficient height from the ground to prevent its flooding during a tsunami, tornado, hurricane, heavy rain, etc. (S4–3)

NRC Response: The NRC agrees that the control room should be protected from flooding. GEH's application for SDA and design certification was submitted with proposed vendor-specified site parameters. The values for maximum groundwater is 2 feet (0.61 m) below plant grade as provided in Table 2.0–1 (Tier 2) of the DCD and the maximum flood level is 1 foot (0.30 m) below plant grade as provided in Table 5.1–1 (Tier 1) of the DCD.

The ESBWR design was evaluated using the vendor-specified flood levels and found to be safe. As described in Chapter 3 of the DCD, the ESBWR construction incorporates several water proofing features: The external walls below groundwater and flood levels are designed to withstand hydrostatic loads, construction and expansion joints have water stops, external surfaces below groundwater and flood levels are waterproofed, penetrations below groundwater and flood levels are sealed, and there are no exterior openings below grade.

If a COL application referencing the ESBWR design is submitted to the NRC, the COL applicant must demonstrate that the site-specific characteristics are bounded by the DCD site parameters. During the review of a COL application using this design, the staff will perform an independent analysis to verify that the flood levels and other relevant site characteristics are within the DCD parameters.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Spent Fuel Pool

Comment: The ESBWR design has an elevated SFP. This is a particularly troublesome feature in common with the Mark I BWR design, which is the design of the Fukushima reactors. (P2–2)

NRC Response: The NRC disagrees with this comment. The ESBWR SFP design is different from the Mark I BWR design in that the ESBWR SFP is located entirely below grade. The ESBWR design does include an additional buffer pool located above grade in the reactor building. The buffer pool contains a small array of spent fuel racks that is used for temporary storage of spent fuel during refueling operations and also includes a location to store new fuel assemblies during power operations.

GDC 2 requires that the ESBWR spent fuel storage facilities (SFP and buffer pool) and the structure within which they are housed, as SSCs important to safety, be protected against the effects of natural phenomena without loss of their safety function. In addition, GDC 61 requires that the design prevents drainage of coolant inventory below an adequate shielding depth, provides adequate coolant flow to the spent fuel racks, and provides a system for detecting and containing pool liner leakage.

The reactor building and the concrete containment, which houses the SFP and additional buffer pool, are seismic Category I structures that are designed to meet the requirements of GDC 2 for protection against natural phenomena such as an earthquake, tornado, or hurricane in combination with normal and accident condition loads considering the effects due to the elevated location of the buffer pool. Information relating to the analysis and design of the reactor building is provided in DCD Sections 3.7 and 3.8 and Appendices 3A, 3B, 3F, and 3G. Through analysis and review of the design, the NRC determined that the reactor building and the concrete containment are structurally adequate to withstand all design-basis loads. The NRC concluded in the FSER that both pools are adequately protected from the effects of natural phenomena without loss of capability to perform their safety functions.

The NRC also concluded in its FSER that, because the SFP and buffer pools have anti-siphoning devices on all submerged Fuel and Auxiliary Pools Cooling System (FAPCS) piping, and there are no other drainage paths by which the level in the SFP or buffer pool could be reduced, coolant will not drain below an adequate shielding depth in either pool.

Cooling of spent fuel located in either the SFP or buffer pool is provided by the FAPCS. In the unlikely event that a loss of active cooling to the spent fuel assemblies occurs, there is enough water to keep the fuel assemblies cooled for a minimum of 72 hours before operator actions are needed. After 72 hours, additional water can be provided through safety-related connections to the fire protection system or another onsite or offsite water source. The NRC concluded in the FSER that cooling for both ESBWR SFP and buffer pools will be maintained.

Finally, the NRC concluded in the FSER that, because the spent fuel pool and buffer pool are equipped with stainless steel liners, concrete walls, and leak detection drains, both detection and containment of pool liner leakage capability are provided.

No change was made to the rule, the DCD, or the EA as a result of this comment.

C. Comments Regarding the NRC's Response to Fukushima Dai-ichi Accident

Some commenters favored delaying (in some fashion) the ESBWR rulemaking until lessons are learned from the Fukushima Dai-ichi Nuclear Power Plant (Fukushima) accident that occurred on March 11, 2011, and the NRC applies the lessons learned to United States (U.S.) nuclear power plants, including the ESBWR design. Background on how the Commission responded to the Fukushima accident and how the ESBWR design addresses Fukushima NTTF recommendations is discussed in Section III of the **SUPPLEMENTARY INFORMATION** section of this document.

As discussed in Section III of the **SUPPLEMENTARY INFORMATION** section of this document, the NRC concludes that no changes to the ESBWR design are warranted at this time to provide reasonable assurance of adequate protection of public health and safety. Moreover, even if the Commission concludes at a later time that some additional action is needed for the ESBWR design, the NRC has ample opportunity and legal authority to modify the ESBWR DCR to implement design changes, as well as to take any necessary action to ensure that COLs that reference the ESBWR make any necessary design changes.

Comment: The NRC should suspend the certification of the ESBWR reactor design and rescind the final design approval it granted on March 9, 2011. Based on the recent events at the Fukushima Dai-ichi site, the NRC should first undertake a far more *rigorous, long-term review of the design and the regulatory implication of the events, implement new regulations to protect public health and safety, and revise the environmental analyses to evaluate the potential health, environmental and economic costs of reactor and SFP accidents.* (S3–1, P3–1, P3–2)

NRC Response: The NRC declines to suspend the ESBWR rulemaking. See *Memorandum and Order,* CLI–11–05, 74 NRC 141 (2011) (ADAMS Accession No. ML112521106).

Background on how the Commission responded to the Fukushima accident and how the ESBWR design addresses Fukushima NTTF recommendations is discussed in Section III of the **SUPPLEMENTARY INFORMATION** section of this document. In that section, the NRC concludes that no changes to the ESBWR design are required at this time to provide reasonable assurance of adequate protection of public health and safety. If the Commission concludes at a later time that some additional action is needed for the ESBWR design, the NRC has ample opportunity and legal authority to modify the ESBWR DCR to implement design changes, as well as to take any necessary action to ensure that COLs that reference the ESBWR also make any necessary design changes.

For these reasons the NRC does not regard delays in the ESBWR design certification process to be appropriate. No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: The Atomic Energy Act (AEA) and NEPA preclude the NRC from approving standardized plant designs until it has completed the investigation of the Fukushima accident and considered the safety and environmental implications of the accident with respect to its regulatory program. NEPA imposes on agencies a continuing obligation to gather and evaluate new information relevant to the environmental impact of its actions. The need to supplement under NEPA when there is new and significant information is also found throughout the NRC regulations, e.g., 10 CFR 51.92(a)(2), 51.50(c)(iii), 51.53(b), and 51.53(c)(3)(iv). The conclusions and recommendations presented in the NTTF report constitute ''new and significant information'' whose environmental implications must be considered before the NRC may certify the ESBWR design and operating procedures. (P2–2, P6–2)

NRC Response: The NRC disagrees with this comment. The comment did not explain what particular provision of the AEA precludes the NRC from issuing a standard DCR. Furthermore, NEPA has no ''continuing obligation'' to gather and evaluate new information relevant to the environmental impact of its actions, because the Commission has determined that issuance of a standard DCR is not a major Federal action significantly affecting the quality of the human environment. See the EA at page 1 (ADAMS Accession No. ML111730382).

No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: The whole nuclear culture must be reviewed before any reactor designs are certified for potential construction, and that all licensing of new reactor designs be put on hold until the NRC's systems of regulations, oversight, and enforcement are thoroughly reviewed and, where required, are made more restrictive. (S3–2)

NRC Response: The NRC considers this comment to be outside the scope of the ESBWR design certification rulemaking. The comment addresses overall nuclear industry safety culture and does not directly address the adequacy of the ESBWR design certification.

Nonetheless, the NRC disagrees with the comment. The NRC considers that its regulatory framework and requirements provide a rigorous and comprehensive design certification and license review process that examines the full extent of siting, system design, and operations of nuclear power plants.

The NRC will continue to process existing applications for new design certifications and licenses in accordance with the schedules that have been established.

Background on how the Commission responded to the Fukushima accident and how the ESBWR design addresses Fukushima near-term task force recommendations is discussed in Section III of the **SUPPLEMENTARY INFORMATION** section of this document. In that section, the NRC concludes that no changes to the ESBWR design are warranted at this time to provide reasonable assurance of adequate protection of public health and safety. Moreover, even if the Commission concludes at a later time that some additional action is needed for the ESBWR design, the NRC has ample opportunity and legal authority to modify the ESBWR DCR to implement design changes, as well as to take any necessary action to ensure that COLs that reference the ESBWR also make any necessary design changes.

For these reasons the NRC does not regard delays in the ESBWR design certification process to be appropriate. No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: The NRC should include a review of public health challenges worldwide from radiation in its decision-making process. (S3–3)

NRC Response: The NRC considers this comment to be outside the scope of the ESBWR DCR. The comment addresses the NRC's generic process and criteria for regulatory decision making, and does not directly address the adequacy of the ESBWR design.

Nonetheless, the NRC disagrees with the comment. The NRC interprets the comment's reference to the ''decision-making process'' to mean the Commission's decision whether to certify the ESBWR design. The NRC reviewed the design and has found that it complies with the NRC's regulations, which provide reasonable assurance of adequate protection of public health and safety, including protection of the public from radiation. The comment did not provide any data, analyses, or other technical information to suggest why the EBSWR design would be unable to provide adequate protection of the public from radiation. No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: The NTTF recommended that licensees reevaluate the seismic and flooding hazards at their sites and if necessary update the design-basis and SSCs important to safety to protect against the updated hazards. NTTF Report, page 30. The ESBWR environmental documents must be supplemented in light of this new and significant information. The NTTF's findings and recommendations are directly relevant to environmental concerns and have a bearing on the proposed action and its impacts. They demonstrate a need to reevaluate the seismic and flooding hazards on the ESBWR reactors, the environmental consequences such hazards could pose, and what, if any, design measures could be implemented (i.e., through NEPA's requisite ''alternatives'' analysis) to ensure that the public is adequately protected from these risks. (P6–4)

NRC Response: The NRC disagrees with the comment. Recommendation 2 of the NTTF, which is the subject of the comment, was focused on licensees of nuclear power reactors and was addressed through site-specific evaluations of the adequacy of the design of the reactors as applied to the site-specific seismic and flooding characteristics. By contrast, the ESBWR design certification—as any other design certification—is not approved for use on any specific site. Rather, the ESBWR design specifies ''design parameters,'' including maximum flood levels and seismic ground motion frequencies and magnitudes, representing the values for which the NRC has determined the ESBWR may safely be placed. A nuclear power plant applicant intending to use the ESBWR must show that the actual site characteristics for the site that the applicant intends to use for the ESBWR fall within the ESBWR-specified design parameters. Thus, NTTF Recommendation 2 is not relevant to the adequacy of the ESBWR design certification. Rather, the NRC regards this NTTF recommendation as an issue relevant to the determination whether a referenced design certification has been adequately demonstrated to be appropriate at the COL applicant's designated site.

In addition, the NRC does not agree that NTTF Recommendation 2 demonstrates that the NRC must ''reevaluate the seismic and flooding hazards on the ESBWR reactors, the environmental consequences such hazards could pose, and what, if any, design measures could be implemented'' through a NEPA ''alternatives'' analysis. Recommendation 2 of the NTTF can best be thought of as a determination to ensure that each site's seismic and flooding characteristics are adequately justified based upon current information. The recommendation does not concern the adequacy of the NRC's substantive regulatory requirements governing protection against seismic and flooding events or their application to any specific reactor design (such as the ESBWR). Thus, even if Recommendation 2 were adopted in full by the Commission and fully implemented, those implementing actions would be directed at licensees of existing nuclear power plants and applicants for new nuclear power plants. The NRC's implementing actions would not be directed at the ESBWR design certification. For these reasons, the NRC does not agree with the comment that ESBWR's EA must be supplemented to address the NTTF Recommendation 2 and implementing actions.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: The NTTF report makes several significant findings when it comes to increasing and improving mitigation measures for new reactor designs and recommends a number of specific steps licensees could take in this regard. Accordingly, the ESBWR environmental report must be supplemented to consider the use of these additional mitigation measures to reduce the project's environmental impacts. See 40 CFR 1502.14(f), 1502.16, 1508.25(b)(3). (P6–5)

NRC Response: The NRC disagrees with the comment. The NTTF report explicitly states that by the ''nature of their passive designs and inherent 72-hour coping capability for core, containment, and SFP cooling with no operator action required, the ESBWR and AP1000 designs have many of the design features and attributes necessary to address the Task Force recommendations. The Task Force supports completing those design certification rulemaking activities without delay.'' (see NTTF Report, pages 71–72). Specifically, the NTTF report does not recommend any actions for the ESBWR design in the near term.

NEPA's obligation to evaluate new information relevant to the environmental impact does not attach unless and until the Commission determines whether ''new and significant'' information has arisen and there is a ''major Federal action'' being undertaken by the NRC for which the new information is relevant and material. The Commission has stated that ''[a]lthough the Task Force completed its review and provided its recommendations to us, the agency continues to evaluate the accident and its implications for U.S. facilities and the full picture of what happened at Fukushima is still far from clear. In short, we do not know today the full implications of the Japan event for U.S. facilities. Therefore, any generic NEPA duty—if one were appropriate at all—does not accrue now. If, however, new and significant information comes to light that requires consideration as part of the ongoing preparation of application-specific NEPA documents, the agency will assess the significance of that information as appropriate.'' CLI–11–05, 74 NRC at 167.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: Before certifying the ESBWR, the NRC must evaluate the relative costs and benefits of adopting all of the NTTF report recommendations, and specifically Recommendations 4 and 7, in light of the NRC's increased understanding regarding accident risks and the strength of its regulatory program to prevent or mitigate them. (P6–6)

NRC Response: The NRC disagrees with the comment. The NTTF report explicitly states that by ''nature of their passive designs and inherent 72-hour coping capability for core, containment, and SFP cooling with no operator action required, the ESBWR and AP1000 designs have many of the design features and attributes necessary to address the Task Force recommendations. The Task Force supports completing those design certification rulemaking activities without delay.'' *Id.,* at 71–72. Specifically, the NTTF report does not recommend any actions, to include Recommendations 4 and 7, for the ESBWR design in the near term. Any potential need to address these recommendations, by addressing ''prestaging of any needed equipment for beyond 72 hours,'' and the establishment of inspection, test, analysis, and acceptance criteria (ITAACs) ''to confirm effective implementation of minimum and extended coping, as described in detailed Recommendation 4.1'' of the NTTF report would be placed on COL applicants referencing the ESBWR design. *Id.,* at 72.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: The comment questions the summary conclusions in Section 7 of the NTTF report regarding Recommendations 4 and 7. Both of these recommendations are contrary to the certification process as currently followed by the NRC in which an applicant for a COL can incorporate by reference a certified reactor design. Directly contrary to this long-standing process, the process suggested in the NTTF report pushes the Fukushima lessons learned onto a COL applicant rather than resolved these issues during the design certification process. Each reactor then becomes a prototype as case-by-case review of potential design and operational changes are made after construction begins. If the phrase ''completing those design certification rulemaking activities without delay'' is an endorsement of the current rulemaking on the ESBWR DCD Revision 9 without consideration of the other Fukushima-driven recommendations (or the subsequent revision to the DCD), the comment questions the depth into which the NTTF analyzed the ESBWR reactor design. (P6–7)

NRC Response: The NRC considers this comment to be outside the scope of the ESBWR design certification rulemaking. The comment presents the commenter's views on Recommendations 4 and 7 of the NTTF Report, but does not address the adequacy of the ESBWR design, the rule, or the EA.

Nonetheless, the NRC disagrees with the comment. The NTTF suggestions that COL applicants or holders address Recommendations 4 and 7, rather than the design certification applicant during the certification process, would not necessitate those COLs to be considered ''prototypes.'' The Commission has stated that ''the agency continues to evaluate the accident and its implications for U.S, facilities and the full picture of what happened at Fukushima is still far from clear. In short, we do not know today the full implications of the Japan event for U.S. facilities.'' CLI–11–05, 74 NRC at 167. Should changes need to be made to the ESBWR design as a result of the evaluation of the Fukushima event, the Commission has stated that ''we have the authority to ensure that certified designs and combined licenses include appropriate Commission-directed changes before operation.'' *Id.* at 163. Further, it is not contrary to the certification process to require changes resulting from Fukushima lessons learned on COLs. The NRC may, under 10 CFR 52.97(c), place conditions upon the COL that the ''Commission deems necessary and appropriate.'' Further, the requirements under 10 CFR 52.63(a)(1) provide a mechanism for the NRC to modify certified designs. Such design changes would be applied to all COL holders referencing this design under 10 CFR 52.63(a)(3). As a result, all COL holders referencing the certified design would be required to make such changes. Moreover, in appropriate (but relatively limited) circumstances the NRC could also impose changes as an ''administrative exemption'' to the issue finality provisions of 10 CFR 52.63 and the ESBWR analogous to what the NRC did in the aircraft impact assessment (AIA) final rule, 10 CFR 50.150 (72 FR 56287; October 3, 2007).

No change was made to the rule, the DCD, or the EA as a result of this comment.

Emergency Petition

NRC Note: The Emergency Petition is comment submissions P1 and P2 in this ESBWR design certification rulemaking proceeding.

Comment: The emergency petition is out of process and should be dismissed on that basis alone. However, if this petition is not so dismissed, the NRC should treat this petition, for aspects related to the single issue specifically regarding the ESBWR design certification rulemaking, as a public comment on the proposed rule. (P4–1)

NRC Response: The NRC need not address, in this rulemaking, the comment's suggestion that the emergency petition is out of process because the Commission considered the merits of it and related filings in its *Memorandum and Order,* CLI–11–05, 74 NRC at 141 (2011) (ADAMS Accession No. ML112521106). The Commission determined that the Emergency Petition should be denied in the relevant adjudicatory proceedings and, on its own motion referred the emergency petition to the NRC staff for consideration as comments in the ESBWR rulemaking.

To the extent that it is relevant to the ESBWR design certification rulemaking, the NRC agrees that the Emergency Petition should be treated as a public comment on the proposed rule. Comments in the Emergency Petition are addressed in this comment response portion of this statement of considerations for the final ESBWR DCR.

No change was made to the rule, the DCD, or the EA as a result of this comment.

Comment: The responses, filed by various industry representatives and COL applicants in accordance with an April 19, 2011, Commission Order (ADAMS Accession No. ML111101277) and setting forth those representatives' and applicants' views on an ''Emergency Petition'' (ADAMS Accession No. ML111080855), were based on mischaracterizations of the Emergency Petition, incorrect representations regarding the NRC's response to the Three Mile Island accident, and incorrect interpretations of the law. Therefore, the responses should be rejected and the Emergency Petition should be granted. (P5–1)

NRC Response: On September 9, 2011, the Commission issued a Memorandum and Order on the Emergency Petition, CLI–11–05, 74 NRC 141 (ADAMS Accession No. ML112521106), which referred both the Emergency Petition and certain documents filed with the NRC to the NRC staff for ''consideration as comments'' in the applicable design certification rulemaking. CLI–11–05, 74 NRC at 176. Comment submission P5 was one of the documents referred by the Commission to the staff for consideration as comments. In accordance with the Commission's direction in CLI–11–05, comment submission P5 has been considered in the ESBWR rulemaking in a manner consistent with other comment submissions filed in the ESBWR rulemaking. Thus, the NRC reviewed the submission to determine the nature of the comments within this comment submission, if it is within the scope of the ESBWR rulemaking, and if so, what substantive response is appropriate. Based upon that review, the NRC determined that comment submission P5 is essentially a procedural reply to responses filed by other entities on the Emergency Petition. The NRC has determined that the reply does not contain any new substantive comments on the adequacy of the ESBWR design that were not already presented in the Emergency Petition and, therefore, has concluded that no further response is needed. No change was made to the rule, the DCD, or the EA as a result of this comment.

III. Regulatory and Policy Issues

This document addresses the regulatory and policy issues that were addressed in the March 2011 proposed rule, the May 2014 supplemental proposed rule, and those not addressed in either the proposed rule or the supplemental proposed rule. The regulatory and policy issues addressed in the March 2011 proposed rule are: (1) Access to safeguards information (SGI) and sensitive unclassified non-safeguards information (SUNSI), and (2) human factors engineering (HFE) operational program elements exclusion from finality. An additional regulatory and policy issue addressed in the May 2014 supplemental proposed rule is incorporation by reference of public documents and issue resolution associated with non-public documents. The NRC provided an opportunity for public comment in the supplemental proposed rule on the issue resolution associated with non-public documents, but not for incorporation by reference of public documents. A number of regulatory and policy issues were not included in either the March 2011 proposed rule or the May 2014 supplemental proposed rule. These are: (1) How the ESBWR design addresses Fukushima NTTF recommendations, (2) changes to Tier 2* information, (3) change control for severe accident design features, and (4) other changes to the ESBWR rule language and difference between the ESBWR rule and other DCRs.

Each of these issues identified above is discussed below.[1]

[1] Some of the regulatory and policy issues discussed below arose after the close of the public comment period on the March 24, 2011, proposed rule. The public was afforded an opportunity to comment on some of these issues in the May 16, 2014, supplemental proposed rule. Section V of the **SUPPLEMENTARY INFORMATION** section of this document describes the NRC's bases for not offering a comment opportunity for some of the regulatory and policy issues that arose after the close of the public comment period on the proposed rule.

A. How the ESBWR Design Addresses Fukushima NTTF Recommendations

The application for certification of the ESBWR design was prepared and submitted, and the NRC staff's review of the application was completed, before the March 11, 2011, Great Tohoku earthquake and tsunami and subsequent events at the Fukushima Dai-ichi Nuclear Power Plant in Japan. In response to the events at Fukushima, the NRC established the NTTF to conduct a systematic and methodical review of NRC processes and regulations to: (1) Determine whether the agency should make additional improvements to its regulatory system; and (2) make recommendations to the Commission for policy directions. On July 12, 2011, the NTTF issued a 90-day report, SECY–11–0093 (ADAMS Accession Number ML11186A950), ''Near Term Report and Recommendations for Agency Actions Following the Events in Japan,'' identifying 12 recommendations. Among other recommendations, the NTTF supported completing the ESBWR design certification rulemaking activity without delay (see NTTF Report, pages 71–72).

On September 9, 2011, in SECY–11–0124, ''Recommended Actions to Be Taken Without Delay from NTTF Report,'' (ADAMS Accession No. ML11245A144) the NRC staff submitted to the Commission for its consideration NTTF recommendations that should be partially or entirely initiated without delay. In SECY–11–0124, the NRC staff concluded that the following subset of actions would provide the greatest potential for improving safety in the near term:

(1) Recommendation 2.1: Seismic and Flood Hazard Reevaluations
(2) Recommendation 2.3: Seismic and Flood Walkdowns
(3) Recommendation 4.1: Station Blackout Regulatory Actions
(4) Recommendation 4.2: Equipment Covered under 10 CFR 50.54(hh)(2) (subsequently renamed ''Mitigation Strategies for Beyond-Design-Basis External Events'' with the issuance of Order EA–12–049)
(5) Recommendation 5.1: Reliable Hardened Vents for Mark I Containments
(6) Recommendation 8: Strengthening and Integration of Emergency Operating Procedures, Severe Accidents Management Guidelines, and Extensive Damage Mitigation Guidelines
(7) Recommendation 9.3: Emergency Preparedness Regulatory Actions (staffing and communications).

On October 3, 2011, in SECY–11–0137, ''Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned'' (ADAMS Accession No. ML11272A203), the NRC staff identified two additional actions that would have the greatest potential for improving safety in the near term. The additional actions are: (1) Inclusion of Mark II containments in the staff's recommendation for reliable hardened vents associated with NTTF Recommendation 5.1 and (2) the implementation of SFP instrumentation proposed in Recommendation 7.1.

The NRC staff determined that the following two near term recommendations are applicable and should be considered for the ESBWR design certification: (1) Recommendation 4.2, Mitigation Strategies for Beyond-Design-Basis External Events (onsite equipment and connections only) and (2) Recommendation 7.1, SFP Instrumentation. The remaining Commission-approved near term recommendations are applicable only to COLs and existing plants (Recommendations 2.1 and 9.3), only to existing plants (Recommendations 2.3 and 5.1), or are planned to be addressed through rulemaking (Recommendations 4.1, 4.2, 7.1, 8, and 9.3).

On February 17, 2012, in SECY–12–0025, ''Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami,'' (ADAMS Accession No. ML12039A103) the NRC staff provided the Commission with proposed orders and requests for information to be issued to all power reactor licensees and holders of construction permits. In SECY–12–0025, the staff indicated its intent to address similar requirements in its reviews of pending and future design certification and COL applications.

On March 9, 2012, in the SRM to SECY–12–0025, the Commission approved issuing the proposed orders with some modifications. On March 12, 2012, the NRC issued Order EA–12–049, ''Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events''; and Order EA 12–051, ''Order Modifying Licenses With Regard to Reliable Spent Fuel Pool Instrumentation'' to the appropriate licensees and permit holders (ADAMS Accession Nos. ML12054A735 and ML12054A679, respectively).

The NRC staff provides 6-month updates to the Commission on all Fukushima-related activities, including the NTTF recommendations that will be addressed in the longer term. The latest update is provided in SECY–14–0046, ''Fifth 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Subsequent Tsunami,'' dated April 17, 2014 (ADAMS Accession No. ML14064A523).

The NRC considered Recommendation 4.2, as modified by SRM–SECY–12–0025, using the requirements in Order EA–12–049. SECY–12–0025 outlines a three-phase approach to developing the strategies. The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling without alternating current power or loss of normal access to the ultimate heat sink. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from offsite. The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely.

As discussed in multiple sections of the DCD, and in the FSER, the ESBWR is designed such that the reactor core and associated coolant, control, and protection systems, including station batteries and other necessary support systems, provide sufficient capacity and capability to ensure that the core will be cooled and there will be appropriate containment integrity and adequate cooling for the spent fuel for 72 hours in the event of an SBO—loss of all normal and emergency ac power.

The ESBWR design credits the isolation condenser system for the first 72 hours of an event in which all ac power sources are lost. Beyond the first 72 hours, the isolation condenser system pool and SFP need to be refilled. The ESBWR design includes provisions to refill the isolation condenser system pool and SFP with onsite equipment without reliance on ac power, such as by the diesel-driven fire pump. In addition, after the first 72 hours of an event, accident mitigation is achieved through the ancillary diesel, which supplies ac power to various components such as: PCCS vent fans, motor driven fire pump, control room habitability area ventilation system air handling units, and emergency lighting. The standby diesels are also needed to support FAPCS operations. Both the ancillary and standby diesels supply short-term and long-term safety loads.

For the reasons set forth in Section 22.5 of the FSER, the NRC found that the applicant has included sufficient nonsafety-related equipment in the RTNSS program to ensure that safety functions relied upon in the post-72-hour period are successful. Emergency procedures are to be developed by the COL applicant to support emergencies, which includes the period after 72 hours from the onset of the loss of all ac power. Further, the nonsafety-related equipment relied upon in the post-72-hour period has been designed in accordance with Commission policy (as described in Section 22.5.6.2 of the FSER) for use of augmented design standards for protection from external hazards and the NRC is engaging with COL applicants to ensure they have established appropriate availability controls for this equipment. Availability controls will be addressed in connection with a COL application referencing the ESBWR standard design.

The ESBWR design supports a COL applicant refilling the pools with offsite equipment, such as local fire pumpers. In the period beyond seven days from the onset of the event, the COL applicant will be responsible for describing how it will make available offsite sources, such as diesel fuel oil for the ancillary and standby diesel generators and water makeup to support long term cooling. The COL applicant must address the ability of offsite support to sustain these functions indefinitely, including procedures, guidance, training and acquisition, staging or installing needed equipment. Therefore, the NRC concludes that the ESBWR design, as described in the DCD, satisfies the underlying purpose of Order EA–12–049 insofar as it includes additional equipment to maintain or restore core and spent fuel pool cooling and containment function in the event of the loss of all ac power. While the ESBWR design includes all of the necessary design features in this respect, the COL applicant must address the programmatic aspects of Order EA–12–049. The NRC staff has already engaged with COL applicants on these arrangements. To the extent a COL applicant proposes to rely on additional equipment to perform required functions in the event of a loss of all ac power, that equipment is outside the scope of the standard ESBWR design and the NRC staff will evaluate it in connection with the COL application.

The NRC considered Recommendation 7.1, as modified by SRM–SECY–12–0025, using the requirements in Order EA–12–051, which describes the key parameters to be used to determine that a level instrument is considered reliable. JLD–ISG–2012–03, Revision 0, ''Compliance with Order EA–12–051, Reliable Spent Fuel Pool Instrumentation,'' (ADAMS Accession No. ML12221A339) endorses with exceptions and clarifications the methodologies described in the industry guidance document NEI 12–02, Revision 1, ''Industry Guidance for Compliance with NRC Order EA–12–051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,'' (ADAMS Accession No. ML122400399) and provides an acceptable approach for satisfying the applicable requirements.

The NRC finds that the ESBWR design has design features that satisfy the underlying purpose of Order EA–12–051 for reliable SFP level instrumentation, except for two matters. The exceptions are whether the safety-related level instrumentation: (1) Are designed to allow the connection of an independent power source, and (2) will maintain its design accuracy following a power interruption or change in power source without recalibration. While the ESBWR design includes all of the necessary design features in this respect, the DCD did not include any information addressing these two matters. In addition, the NRC is currently developing a rulemaking which would address spent fuel pool instrumentation for beyond design basis events/accidents. This rulemaking may adopt different requirements than what is currently considered acceptable to meet the underlying purpose of order EA–12–051 and its related guidance. For these reasons, the NRC is excluding from issue finality and issue resolution these two aspects of the ESBWR spent fuel pool instrumentation design features. The exclusions have two consequences. First, any combined license applicant referencing the ESBWR design certification rule will have to provide information demonstrating that the NRC's requirements on these two matters are met. Second, the NRC need not address the factors of 10 CFR 52.63 either when it reviews the combined license application for adequacy with respect to these two matters, or in connection with any amendment of the ESBWR design certification rule imposing requirements to govern those matters.

B. Incorporation by Reference of Public Documents and Issue Resolution Associated With Non-Public Documents

In Section III, ''Scope and Contents,'' of the proposed ESBWR DCR (76 FR 16549; March 24, 2011), the only document for which the NRC proposed to obtain approval from the Office of the Federal Register (OFR) for incorporation by reference into the ESBWR design certification rule was the ESBWR DCD, Revision 9 (DCD Revision 9). Such approval would make DCD Revision 9 a legally-binding requirement on any referencing combined license applicant and holder by virtue of publication in the **Federal Register** as a final rule. This was based upon the assumption that the DCD specified all necessary requirements in Tier 1 and Tier 2 (with the exception of non-public documents containing proprietary information,[2] security-related information,[3] and SGI).

After the close of the public comment period, the NRC recognized that Tier 2, Section 1.6, ''Material Incorporated by Reference and General Reference Material,'' of the ESBWR DCD states that a number of documents are ''incorporated by reference'' into Tier 2 of the ESBWR DCD, and which contain information intended to be requirements. These documents were listed in Tables 1.6–1, ''Referenced GE/GEH Reports,'' and 1.6–2, ''Referenced non-GE/GEH Topical Reports,'' of the DCD Revision 9. Although some of the documents contain information which is intended to be requirements (based on the text of the DCD), neither Tables 1.6–1 and 1.6–2 of the DCD nor Section III of the proposed ESBWR design certification rule clearly stated which of these documents were intended as requirements. Documents intended as requirements (and which are publicly available) should have been listed in Section III of the ESBWR design certification rule as being approved for incorporation by reference by the Director of the OFR. Tables 1.6–1 and 1.6–2 also included documents that, although ''incorporated by reference'' into DCD Revision 9, were not intended to be requirements, but were references ''for information only.'' Thus, the ESBWR proposed rule did not clearly differentiate between these two different classes of documents. Finally, Tables 1.6–1 and 1.6–2 of DCD Revision 9 included both publicly-available documents and non-publicly available documents,[4] but for some of the documents which were not publicly available, GEH had not created a publicly-available version of that document to support the public comment process. The creation of publicly-available versions of non-public documents to support the public commenting process and transparency has been a long-standing practice for

[2] For purposes of this discussion, ''proprietary information'' constitutes trade secrets or commercial or financial information that are privileged or confidential, as those terms are used under the Freedom of Information Act and the NRC's implementing regulation at 10 CFR part 9.

[3] For purposes of this discussion, ''security-related information'' means information subject to non-disclosure under 10 CFR 2.390(a)(7)(vi).

[4] The non-publicly available documents contain proprietary, security-related, and/or safeguards information.

both design certification rulemakings and licensing actions.

To address the NRC's concerns, for those non-public documents which include information intended to be treated as requirements and for which publicly-available versions were not previously created, GEH created publicly-available versions of those non-public documents. GEH also submitted Revision 10 to the DCD (DCD Revision 10), which included three tables in Section 1.6 that superseded Tables 1.6–1 and 1.6–2 in DCD Revision 9. These three tables—Tables 1.6–1, ''GE/GEH Reports Incorporated by Reference,'' 1.6–2, ''Non-GE/GEH Reports Incorporated by Reference,'' and 1.6–3, ''Referenced Reports (not Incorporated by Reference,''—collectively clarify which documents are intended to be requirements and which documents are references only.

The supplemental proposed rule (79 FR 25715; May 6, 2014): (1) Announced the availability of DCD Revision 10; (2) described the distinction between those documents intended as requirements versus those which were for information only; (3) requested public comments on the NRC's intent to treat 50 non-public, referenced documents in DCD Revision 10 (listed in Table 2 of the supplemental proposed rule) as requirements and matters resolved in subsequent licensing and enforcement actions for plants referencing the ESBWR design certification; and (4) clarified, but did not request public comments on, the NRC's intent to obtain approval for incorporation by reference from the Director of the OFR for both DCD Revision 10 and the 20 publicly-available documents referenced in DCD Revision 10 (listed in Table 3 of the supplemental proposed rule), which are intended by the NRC to be requirements.

The 50 non-publicly available documents listed in Table 3 below are considered by the NRC to be requirements applicable to any combined license applicant or holder of a combined license referencing the ESBWR design certification rule, where the language of DCD Revision 10 makes clear that any one of those documents is intended to be a requirement. In addition, the 50 non-public documents are within the scope of issue resolution under Section VI of Appendix E, and are accorded issue finality protection under that Section VI and 10 CFR 52.63.

TABLE 3—50 NON-PUBLIC DOCUMENTS WHICH THE NRC REGARDS AS REQUIREMENTS, ARE MATTERS RESOLVED UNDER PARAGRAPH VI, ISSUE RESOLUTION, OF THE ESBWR DESIGN CERTIFICATION RULE, AND ARE ACCORDED ISSUE FINALITY PROTECTION

Document No.	Document title	Publicly-available ADAMS Accession No.	Non-publicly available ADAMS Accession No.
NEDE–33391, NEDO–33391 ..	GE Hitachi Nuclear Energy, ''ESBWR Safeguards Assessment Report,'' NEDE–33391, Class III (Safeguards, Security-Related, and Proprietary), Revision 3, March 2010, and NEDO–33391, Class I (Non-safeguards, Non-security related, and Non-proprietary), Revision 3, March 2014.	ML14093A138	N/A (Safeguards information cannot be placed in ADAMS)
NEDC–31959P, NEDO–31959	GE Nuclear Energy, ''Fuel Rod Thermal-Mechanical Analysis Methodology (GSTRM),'' NEDC–31959P (Proprietary), April 1991, and NEDO–31959 (Non-proprietary), April 1991.	ML14093A145	ML14093A146
NEDC–32992P–A, NEDO–32992–A.	GE Nuclear Energy, J.S. Post and A.K. Chung, ''ODYSY Application for Stability Licensing Calculations,'' NEDC–32992P–A, Class III (Proprietary), July 2001, and NEDO–32992–A, Class I (Non-proprietary), July 2001.	ML14093A250	ML012610605
NEDC–33139P–A, NEDO–33139–A.	Global Nuclear Fuel, ''Cladding Creep Collapse,'' NEDC–33139P–A, Class III (Proprietary), July 2005, and NEDO–33139–A, Class I (Non-proprietary), July 2005.	ML14094A227	ML14094A228
NEDE–31758P–A, NEDO–31758–A.	GE Nuclear Energy, ''GE Marathon Control Rod Assembly,'' NEDE–31758P–A (Proprietary), October 1991, and NEDO–31758–A (Non-proprietary), October 1991.	ML14093A142	ML14093A143
NEDC–32084P–A, NEDO–32084–A.	GE Nuclear Energy, ''TASC–03A, A Computer Program for Transient Analysis of a Single Channel,'' NEDC–32084P–A, Revision 2, Class III (Proprietary), July 2002, and NEDO–32084–A, Class 1 (Non-proprietary), Revision 2, September 2002.	ML100220484	ML100220485
NEDC–32601 P–A, NEDO–32601–A.	GE Nuclear Energy, ''Methodology and Uncertainties for Safety Limit MCPR Evaluations,'' NEDC–32601P–A, Class III (Proprietary), and NEDO–32601–A, Class I (Non-proprietary), August 1999.	ML14093A216	ML003740145
NEDC–32983P–A, NEDO–32983–A.	GE Nuclear Energy, ''GE Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations,'' Licensing Topical Report NEDC–32983P–A, Class III (Proprietary), Revision 2, January 2006, and NEDO–32983–A, Class I (Non-proprietary), Revision 2, January 2006.	ML072480121	ML072480125
NEDC–33075P–A, NEDO–33075–A.	GE Hitachi Nuclear Energy, ''General Electric Boiling Water Reactor Detect and Suppress Solution—Confirmation Density,'' NEDC–33075P–A, Class III (Proprietary), and NEDO–33075–A, Class I (Non-proprietary), Revision 6, January 2008.	ML080310396	ML080310402
NEDC–33079P, NEDO–33079	GE Nuclear Energy, ''ESBWR Test and Analysis Program Description,'' NEDC–33079P, Class III (Proprietary), Revision 1, March 2005, and NEDO–33079, Class I (Non-proprietary), Revision 1, November 2005.	ML053460471	ML051390233

TABLE 3—50 NON-PUBLIC DOCUMENTS WHICH THE NRC REGARDS AS REQUIREMENTS, ARE MATTERS RESOLVED UNDER PARAGRAPH VI, ISSUE RESOLUTION, OF THE ESBWR DESIGN CERTIFICATION RULE, AND ARE ACCORDED ISSUE FINALITY PROTECTION—Continued

Document No.	Document title	Publicly-available ADAMS Accession No.	Non-publicly available ADAMS Accession No.
NEDC–33083P–A, NEDO–33083–A.	GE Nuclear Energy, ‘‘TRACG Application for ESBWR,’’ NEDC–33083P–A, Revision 1, Class III (Proprietary), September 2010, and NEDO–33083–A, Revision 1, Class I (Non-proprietary), September 2010.	ML102770606	ML102770608
NEDC–33237P–A, NEDO–33237–A.	Global Nuclear Fuel, ‘‘GE14 for ESBWR—Critical Power Correlation, Uncertainty, and OLMCPR Development,’’ NEDC–33237P–A, Revision 5, Class III (Proprietary), and NEDO–33237–A, Revision 5, Class I (Non-proprietary), September 2010.	ML102770246	ML102770244
NEDC–33238P, NEDO–33238	Global Nuclear Fuel, ‘‘GE14 Pressure Drop Characteristics,’’ NEDC–33238P, Class III (Proprietary), and NEDO–33238, Class I (Non-proprietary), December 2005.	ML060050328	ML060050330
NEDC–33239P–A, NEDO–33239P–A.	Global Nuclear Fuel, ‘‘GE14 for ESBWR Nuclear Design Report,’’ NEDC–33239P–A, Class III (Proprietary), and NEDO–33239–A, Class I (Non-proprietary), Revision 5, October 2010.	ML102800405	ML102800408 (part 1) ML102800425 (part 2)
NEDC–33240P–A, NEDO–33240–A.	Global Nuclear Fuel, ‘‘GE14E Fuel Assembly Mechanical Design Report,’’ NEDC–33240P–A, Revision 1, Class III (Proprietary), and NEDO–33240–A, Revision 1, Class I (Non-proprietary), September 2010.	ML102770060	ML102770061
NEDC–33242P–A, NEDO–33242–A.	Global Nuclear Fuel, ‘‘GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report,’’ NEDC–33242P–A, Revision 2, Class III (Proprietary), and NEDO–33242–A, Revision 2, Class I (Non-proprietary), September 2010.	ML102730885	ML102730886
NEDC–33326P–A, NEDO–33326–A.	Global Nuclear Fuel, ‘‘GE14E for ESBWR Initial Core Nuclear Design Report,’’ NEDC–33326P–A, Revision 1, Class III (Proprietary), and NEDO–33326–A, Revision 1, Class I (Non-proprietary), September 2010.	ML102740191	ML102740193 (part 1) ML102740194 (part 2)
NEDC–33374P–A, NEDO–33374–A.	GE-Hitachi Nuclear Energy, ‘‘Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants,’’ NEDC–33374P–A, Revision 4, Class III (Proprietary), September 2010, and NEDO–33374–A, Revision 4, Class I (Non-proprietary), September 2010.	ML102860687	ML102860688
NEDC–33456P, NEDO–33456	Global Nuclear Fuel, ‘‘Full-Scale Pressure Drop Testing for a Simulated GE14E Fuel Bundle,’’ NEDC–33456P, Class III (Proprietary), and NEDO–33456, Class I (Non-proprietary), Revision 0, March 2009.	ML090920867	ML090920868
NEDE–10958–PA, NEDO–10958–A.	General Electric Company, ‘‘General Electric Thermal Analysis Basis Data, Correlation and Design Application,’’ NEDE–10958–PA, Class III (Proprietary), and ‘‘General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application,’’ NEDO–10958–A, Class I (Non-proprietary), January 1977.	ML102290144	ML092820214
NEDE–24011–P–A–16, NEDO–24011–A–16.	Global Nuclear Fuel, ‘‘GESTAR II General Electric Standard Application for Reactor Fuel,’’ NEDE–24011–P–A–16, Class III (Proprietary), and NEDO–24011–A–16, Class I (Non-proprietary), Revision 16, October 2007.	ML091340077	ML091340081
NEDE–24011–P–A–US–16, NEDO–24011–A–US–16.	Global Nuclear Fuel, ‘‘GESTAR II General Electric Standard Application for Reactor Fuel, Supplement for United States,’’ NEDE–24011–P–A–US–16, Class III (Proprietary), and NEDO–24011–A–US–16, Class I (Non-proprietary), Revision 16, October 2007.	ML091340080	ML091340082
NEDE–30130–P–A, NEDO–30130–A.	General Electric Company, ‘‘Steady State Nuclear Methods,’’ NEDE–30130–P–A, Class III (Proprietary), April 1985, and NEDO–30130–A, Class I (Non-proprietary), May 1985.	ML14104A064	ML070400570
NEDE–31152P, NEDO–31152	Global Nuclear Fuel, ‘‘Global Nuclear Fuels Fuel Bundle Designs,’’ NEDE–31152P, Revision 9, Class III (Proprietary), May 2007, and NEDO–33152, Revision 9, Class I (Non-proprietary), May 2007.	ML071510287	ML071510289
NEDE–32176P, NEDO–32176	GE Hitachi Nuclear Energy, J.G.M. Andersen, et al., ‘‘TRACG Model Description,’’ NEDE–32176P, Revision 4, Class III (Proprietary), January 2008, and NEDO–32176, Class I (Non-proprietary), Revision 4, January 2008.	ML080370271	ML080370276

TABLE 3—50 NON-PUBLIC DOCUMENTS WHICH THE NRC REGARDS AS REQUIREMENTS, ARE MATTERS RESOLVED UNDER PARAGRAPH VI, ISSUE RESOLUTION, OF THE ESBWR DESIGN CERTIFICATION RULE, AND ARE ACCORDED ISSUE FINALITY PROTECTION—Continued

Document No.	Document title	Publicly-available ADAMS Accession No.	Non-publicly available ADAMS Accession No.
NEDE–33083 Supplement 1P–A, NEDO–33083 Supplement 1–A.	GE Hitachi Nuclear Energy, B.S. Shiralkar, et al, "TRACG Application for ESBWR Stability Analysis," NEDE–33083, Supplement 1P–A, Revision 2, Class III (Proprietary), September 2010, and NEDO–33083, Supplement 1–A, Revision 2, Class I (Non-proprietary), September 2010.	ML102770552	ML102770550
NEDE–33083 Supplement 2P–A, NEDO–33083 Supplement 2–A.	GE Hitachi Nuclear Energy, "TRACG Application for ESBWR Anticipated Transient Without Scram Analyses," NEDE–33083, Supplement 2P–A, Revision 2, Class III (Proprietary), October 2010 and NEDO–33083, Supplement 2–A, Revision 2, Class I (Non-proprietary), October 2010.	ML103000353	ML103000355
NEDE–33083 Supplement 3P–A, NEDO–33083 Supplement 3–A.	GE Hitachi Nuclear Energy, "TRACG Application for ESBWR Transient Analysis," NEDE–33083, Supplement 3P–A, Revision 1, Class III (Proprietary), and NEDO–33083, Supplement 3–A, Revision 1, Class I (Non-proprietary), September 2010.	ML102770606	ML102770608
NEDE–33197P–A, NEDO–33197–A.	GE Hitachi Nuclear Energy, "Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring," NEDE–33197P–A, Revision 3, Class III (Proprietary), and NEDO–33197–A, Revision 3, Class I, (Non-proprietary), October 2010.	ML102810320	ML102810341
NEDE–33217P, NEDO–33217	GE Hitachi Nuclear Energy, "ESBWR Man-Machine Interface System and Human Factors Engineering Implementation Plan," NEDE–33217P, Class III (Proprietary), and NEDO–33217, Class I (Non-proprietary), Revision 6, February 2010.	ML100480284	ML100480285
NEDE–33220P, NEDO–33220	GE Hitachi Nuclear Energy, "ESBWR Human Factors Engineering Allocation of Function Implementation Plan," NEDE–33220P, Class III (Proprietary), and NEDO–33220, Class I (Non-proprietary), Revision 4, February 2010.	ML100480209	ML100480202
NEDE–33221P, NEDO–33221	GE Hitachi Nuclear Energy, "ESBWR Human Factors Engineering Task Analysis Implementation Plan," NEDE–33221P, Class III (Proprietary), and NEDO–33221, Class I (Non-proprietary), Revision 4, February 2010.	ML100480212	ML100480213
NEDE–33226P, NEDO–33226	GE Hitachi Nuclear Energy, "ESBWR—Software Management Program Manual," NEDE–33226P, Class III (Proprietary), Revision 5, February 2010, and NEDO–33226, Class I (Non-proprietary), Revision 5, February 2010.	ML100550837	ML100550844
NEDE–33243P–A, NEDO–33243–A.	GE Hitachi Nuclear Energy, "ESBWR Control Rod Nuclear Design," NEDE–33243P–A, Revision 2, Class III (Proprietary), September 2010, and NEDO–33243–A, Revision 2, Class I (Non-proprietary), September 2010.	ML102740171	ML102740178
NEDE–33244P–A, NEDO–33244–A.	GE Hitachi Nuclear Energy, "ESBWR Marathon Control Rod Mechanical Design Report," NEDE–33244P–A, Class III (Proprietary), Revision 2, September 2010, and NEDO–33244–A, Revision 2, Class I (Non-proprietary), September 2010.	ML102770208	ML102770209
NEDE–33245P, NEDO–33245	GE Hitachi Nuclear Energy, "ESBWR—Software Quality Assurance Program Manual," NEDE–33245P, Class III (Proprietary), Revision 5, February 2010, and NEDO–33245, Class I (Non-proprietary), Revision 5, February 2010.	ML100550839	ML100550847
NEDE–33259P–A, NEDO–33259–A.	GE Hitachi Nuclear Energy, "Reactor Internals Flow Induced Vibration Program," NEDE–33259P–A, Class III (Proprietary), Revision 3, October 2010, and NEDO–33259–A, Class I (Non-proprietary), Revision 3, October 2010.	ML102920241	ML102920248
NEDE–33261P, NEDO–33261	GE Hitachi Nuclear Energy, "ESBWR Containment Load Definition," NEDE–33261P, Class III (Proprietary), and NEDO–33261, Class I (Non-proprietary), Revision 2, June 2008.	ML082600720	ML082600721
NEDE–33268P, NEDO–33268	GE Hitachi Nuclear Energy, "ESBWR Human Factors Engineering Human-System Interface Design Implementation Plan," NEDE–33268P, Class III (Proprietary), and NEDO–33268, Class I (Non-proprietary), Revision 5, February 2010.	ML100480179	ML100480180

TABLE 3—50 NON-PUBLIC DOCUMENTS WHICH THE NRC REGARDS AS REQUIREMENTS, ARE MATTERS RESOLVED UNDER PARAGRAPH VI, ISSUE RESOLUTION, OF THE ESBWR DESIGN CERTIFICATION RULE, AND ARE ACCORDED ISSUE FINALITY PROTECTION—Continued

Document No.	Document title	Publicly-available ADAMS Accession No.	Non-publicly available ADAMS Accession No.
NEDE–33276P, NEDO–33276	GE Hitachi Nuclear Energy, ''ESBWR Human Factors Engineering Verification and Validation Implementation Plan,'' NEDE–33276P, Class III (Proprietary), and NEDO–33276, Class I (Non-proprietary), Revision 4, February 2010.	ML100480182	ML100480183
NEDE–33295P, NEDO–33295	GE Hitachi Nuclear Energy, ''ESBWR Cyber Security Program Plan,'' NEDE–33295P, Class III (Proprietary), Revision 2, September 2010, and NEDO–33295, Class I (Non-proprietary), Revision 2, September 2010.	ML102880103	ML102880104
NEDE–33304P, NEDO–33304	GE Hitachi Nuclear Energy, ''GEH ESBWR Setpoint Methodology,'' NEDE–33304P, Class III (Proprietary), and NEDO–33304, Class I (Non-proprietary), Revision 4, May 2010.	ML101450251	ML101450253
NEDE–33312P, NEDO–33312	GE Hitachi Nuclear Energy, ''ESBWR Steam Dryer Acoustic Load Definition,'' NEDE–33312P, Class III (Proprietary), Revision 5, December 2013, and NEDO–33312, Class I (Non-proprietary), Revision 5, December 2013.	ML13344B157	ML13344B163
NEDE–33313P, NEDO–33313	GE Hitachi Nuclear Energy, ''ESBWR Steam Dryer Structural Evaluation,'' NEDE–33313P, Class III (Proprietary), Revision 5, December 2013, and NEDO–33313, Class I (Non-proprietary), Revision 5, December 2013.	ML13344B158	ML13344B164
NEDE–33408P, NEDO–33408	GE Hitachi Nuclear Energy, ''ESBWR Steam Dryer—Plant Based Load Evaluation Methodology, PBLE01 Model Description,'' NEDE–33408P, Class III (Proprietary), Revision 5, December 2013, and NEDO–33408, Class I (Non-proprietary), Revision 5, December 2013.	ML13344B159	ML13344B176 (part 1) ML13344B175 (part 2)
NEDE–33440P, NEDO–33440	GE Hitachi Nuclear Energy ''ESBWR Safety Analysis—Additional Information,'' NEDE–33440P, Class III (Proprietary), and NEDO–33440, Class I (Non-proprietary), Revision 2, March 2010.	ML100920316	ML100920317 (part 1) ML100920318 (part 2)
NEDE–33516P–A, NEDO–33516–A.	GE Hitachi Nuclear Energy, ''ESBWR Qualification Plan Requirements for a 72-Hour Duty Cycle Battery,'' NEDE–33516P–A, Revision 2, Class III (Proprietary), September 2010, and NEDO–33516–A, Revision 2, Class I (Non-proprietary), September 2010.	ML102880499	ML102880500
NEDE–33536P, NEDO–33536	GE Hitachi Nuclear Energy, ''Control Building and Reactor Building Environmental Temperature Analysis for ESBWR,'' NEDE–33536P, Class III (Security-Related and Proprietary), Revision 1, October 2010, and NEDO–33536, Class I (Non-security Related and Non-proprietary), Revision 1, October 2010.	ML102780329	ML102780330
NEDE–33572P, NEDO–33572	GE Hitachi Nuclear Energy, ''ESBWR ICS and PCCS Condenser Combustible Gas Mitigation and Structural Evaluation,'' NEDE–33572P, Class II (Proprietary), Revision 3, September 2010, and NEDO–33572, Revision 3, Class I (Non-proprietary), September 2010.	ML102740579	ML102740566
Letter w/attachment	Letter from R.J. Reda (GE) to R.C. Jones, Jr. (NRC), MFN 098–96, ''Implementation of Improved Steady-State Nuclear Methods,'' Class III (Proprietary), July 2, 1996, and Letter from J.G. Head (GEH) to NRC Document Control Desk, MFN 098–96 Supplement 1, Class I (Non-proprietary), March 31, 2014.	ML14093A140	ML14094A240

Table 3 Note: Documents whose document number contains ''NEDC'' or ''NEDE'' are non-public and documents whose document number contains ''NEDO'' are public.

C. Changes to Tier 2 Information*

The NRC is making three changes from the proposed rule regarding Tier 2* matters under Section VIII, ''Processes for Changes and Departures,'' of the ESBWR rule language. These changes are described below.

First, paragraph VIII.B.6.c(1) is changed from ''ASME Boiler and Pressure Vessel Code, Section III'' to ''ASME Boiler and Pressure Vessel Code, Section III, Subsections NE (Division 1) and CC (Division 2) for containment vessel design.'' This redesignation of Tier 2* information in paragraph VIII.B.6.c.(1) applies only to the ASME BPV Code, Section III, Subsections NE (Division 1) and CC (Division 2) for the design of ASME BPV Code Class MC (metal containment) and CC (concrete containment) pressure-retaining components (e.g., the containment vessel). This change does not apply to the design and construction of mechanical pressure-boundary components because they are required to meet the design and construction requirements in Section III for ASME BPV Code Class 1, 2, and 3 mechanical pressure-boundary components, which are incorporated by reference into 10 CFR 50.55a. The regulations in 10 CFR 50.55a include provisions in paragraphs 50.55a(c)(3), (d)(2) and (e)(2) for reactor coolant pressure boundary, Quality Group B, and Quality Group C (i.e., ASME BPV Code Classes 1, 2, and 3 components, respectively. These paragraphs provide the necessary regulatory controls on the use of later edition and addenda to the ASME BPV Code, Section III through the conditions the NRC established on the use of paragraph NCA–1140 of the ASME BPV Code, Section III. As a result, these rule requirements adequately control the ability of a licensee to use later editions or addenda of the ASME BPV Code, Section III such that a Tier 2* designation is not necessary.

Second, paragraph VIII.B.6.c(3) is changed from ''Motor-operated valves'' to ''Power-operated valves.'' This change is necessary to correct an error in the proposed rule text. Consistent with Revisions 9 and 10 of the ESBWR DCD, which were the versions of the DCD available for public comment, the only valves that are described in Tier 2* information in an ESBWR nuclear power plant are air-operated rather than motor-operated.

Third, the NRC discussed in the supplemental proposed rule its proposal to designate the revised ESBWR steam dryer analysis methodology as Tier 2* information throughout the life of any license referencing the ESBWR DCR. This is a change from Revision 9 of the ESBWR DCD, which identified much of this information (in its earlier form before the revisions reflected in Revision 10) as Tier 2. Therefore, the ESBWR steam dryer analysis methodology was not identified as Tier 2* information in the proposed rule.

In the supplemental proposed rule, the NRC proposed to designate the revised ESBWR steam dryer pressure load analysis methodology as Tier 2* for two reasons. First, the NRC's experience with other applications using this methodology highlights the importance of the proper application of the steam dryer pressure load analysis methodology. Therefore, it is necessary for the NRC to review any changes a referencing applicant or licensee proposes to the methodology from that which the NRC previously reviewed and approved. Second, in Revision 10 to the ESBWR DCD, GEH revised the designation of this methodology to Tier 2* and, therefore, the rule's designation is consistent with GEH's designation in the DCD.

The supplemental proposed rule provided an opportunity for public comment on the proposed designation as Tier 2* of certain information related to the pressure load analysis methodology supporting the ESBWR steam dryer design. The NRC staff did not receive any public comments on the proposal to designate information related to the ESBWR steam dryer pressure load analysis methodology as Tier 2* information. Therefore, the final rule designates the revised ESBWR steam dryer pressure load analysis methodology as Tier 2* information throughout the life of any license referencing the ESBWR DCR.

D. Change Control for Severe Accident Design Features

The **SUPPLEMENTARY INFORMATION** section of the amendment to 10 CFR part 52 (72 FR 49392, at 49394; August 28, 2007), states that the Commission codified separate criteria in paragraph B.5.c of Section VIII of each DCR for determining if a departure from design information that resolves these severe accident issues would require a license amendment. Originally, the final rule was applied specifically to changes to ex-vessel severe accident design features. In the SRM to SECY–12–0081, ''Risk-Informed Regulatory Framework for New Reactors,'' dated October 22, 2012, the Commission directed the staff to make the change process in paragraph B.5.c of Section VIII applicable to severe accident design features, both ex-vessel and non-ex-vessel, that are described in the plant-specific DCD. This policy was changed after issuance of the proposed ESBWR rule. The policy was changed to ensure that, for changes to Tier 2 information, the effects on all severe accident design features—and not just ex-vessel severe accident design features—are considered.

However, the NRC has not changed the rule language in paragraph B.5.c of Section VIII for the ESBWR rulemaking because all of the relevant severe accident design features (i.e., those that are non-ex-vessel) are described in Tier 1 information. Tier 1 information, by definition, includes change controls in Section VIII of the rule text that meet the underlying purpose of the Commission's direction. Therefore, this change was not necessary for the ESBWR design certification.

E. Access to Safeguards Information (SGI) and Sensitive Unclassified Non-Safeguards Information (SUNSI)

In the four currently approved design certifications (10 CFR part 52, appendices A through D), paragraph VI.E sets forth specific directions on how to obtain access to proprietary information and SGI on the design certification in connection with a license application proceeding referencing that DCR. These provisions were developed before the events of September 11, 2001. After September 11, 2001, Congress changed the statutory requirements governing access to SGI and the NRC has revised its rules, procedures, and practices governing control of and access to SGI and SUNSI. The NRC has determined that generic direction on obtaining access to SGI and SUNSI is no longer appropriate for newly approved DCRs. Accordingly, the specific requirements governing access to SGI and SUNSI contained in paragraph VI.E of the four currently approved DCRs are not included in the DCR for the ESBWR. Instead, the NRC will specify the procedures to be used for obtaining access at an appropriate time in the COL proceeding referencing the ESBWR DCR.

F. Human Factors Engineering (HFE) Operational Program Elements Exclusion From Finality

In the December 6, 1996, SRM (ADAMS Accession No. ML003754873) to SECY–96–077, ''Certification of Two Evolutionary Designs,'' dated April 15, 1996, the Commission set forth a policy that operational programs should be excluded from finality except where necessary to find design elements acceptable. For HFE programs for the ESBWR standard design, the Commission is implementing this policy in a manner different than for other existing DCRs. The difference in treatment of HFE for the ESBWR design arises from the level of detail of HFE review for the ESBWR as compared to earlier certified standard designs. For the earlier designs, the NRC staff reviewed the HFE programs at a ''programmatic'' level of design, while for the ESBWR, the staff reviewed the HFE programs at a more detailed ''implementation plan'' level of design. In providing this additional detail, GEH addressed existing NRC guidelines in NUREG–0711, Revision 2, ''Human Factors Engineering Program Review Model,'' which are comprehensive and go beyond the operational program information needed as input to the HFE design. Therefore, GEH included, in the DCD, details on two HFE operational program elements (procedures and training) that are not used to determine the adequacy of the HFE design. In keeping with the established Commission policy of not approving operational program elements through design certification except where necessary to find design elements acceptable, the NRC is excluding these two HFE operational program elements in the ESBWR DCD from the scope of the design approved in the rule. This is done explicitly in Section VI, Issue Resolution, of the ESBWR rule, by excluding the two HFE operational program elements from the issue finality and issue resolution accorded to the design. In addition, the procedures and training elements included in the HFE program are redundant to what is reviewed as part of the operational programs described in Chapter 13, ''Conduct of Operations,'' of the SRP. Accordingly, the NRC is revising the HFE regulatory guidance in NUREG–0711, Revision 3, ''Human Factors Engineering Program Review Model,'' to address this overlap, but the corresponding revision to the SRP has not yet been completed. This exclusion is unique to the ESBWR design because all other DCDs for the previously certified designs do not include operational program descriptions of HFE procedures and training and the respective DCRs did not include specific exclusions from finality for them.

G. Other Changes to the ESBWR Rule Language and Differences Between the ESBWR Rule and Other DCRs

The language of the ESBWR design certification rule differs from the rule language of other DCRs in two substantive areas. First, paragraph IX was reserved for future use because the substantive requirements in this paragraph (for other DCRs) has since been incorporated into 10 CFR part 52 in a 2007 rulemaking (72 FR 49352; August 28, 2007) and thus are no longer needed in the four existing DCR appendices. The NRC intends to remove these requirements from Section IX of the four existing DCR appendices in future amendment(s) separate from this rulemaking.

The second difference involves documents incorporated by reference into the ESBWR design certification rule. In the first four DCRs, the DCD is the only document identified in Section III of the rule language as being approved by the Office of the Federal Register for incorporation by reference. However, the ESBWR final rule identifies the ESBWR DCD and 20 publicly-available documents referenced in the DCD, Tier 2, Section 1.6 as approved for incorporation by reference. These 20 documents, which are intended by the NRC and GEH to be requirements, are listed in a table in Section III of the ESBWR final rule language. By being approved by the Office of the Federal Register for incorporation by reference, Revision 10 of the DCD and the 20 publicly-available documents are considered to be requirements as if they had been published in the **Federal Register**.

IV. Technical Issues

The NRC issued an FSER for the ESBWR design in March 2011, and subsequently published the FSER as NUREG–1966 in April 2014. The NRC issued an advanced supplemental SER in April 2014 (ADAMS Accession No. ML14043A134) and plans to publish Supplement No. 1 to NUREG–1966, as described in Section III of the SUPPLEMENTARY INFORMATION section of this document, before this final rule becomes effective. The FSER and its supplement provide the basis for issuance of a design certification under subpart B to 10 CFR part 52.

The significant technical issues that were resolved during the initial review of the ESBWR design (i.e., the NRC staff's review of Revision 9 of the ESBWR DCD and development of an FSER) are: (1) Regulatory treatment of nonsafety systems (RTNSS), (2) containment performance, (3) control room cooling, (4) feedwater temperature operating domain, (5) steam dryer analysis methodology, (6) aircraft impact assessment, (7) the use of ASME Code Case N–782, and (8) an exemption for the safety parameter display system. These issues were discussed in the March 2011 proposed rule. No public comments were received on these issues.

After publishing the proposed rule, the NRC addressed several issues that were changed in Revision 10 of the DCD or required a change to the FSER. The NRC staff reviewed these changes and developed an advanced supplemental SER as described above. The issues that were resolved in the advanced supplemental SER are: (1) Steam dryer analysis methodology, (2) loss of one or more phases of offsite power, (3) spent fuel assembly integrity in spent fuel racks, (4) Turbine Building Offgas System design requirements, (5) ASME Code statement in Chapter 1 of the ESBWR DCD, and (6) clarification of ASME component design ITAACs. The NRC also made changes to the advanced supplemental SER after the publication of the supplemental proposed rule.

After publication of the proposed rule, the NRC addressed two issues that were not addressed in Revision 10 of the DCD or in the advanced supplemental FSER. These issues are: (1) Hurricane-generated winds and missiles, and (2) changes to Tier 2* information.

Each of these issues identified above is discussed below. The public was afforded an opportunity to comment on some of these issues in the May 6, 2014 supplemental proposed rule. Section V of the SUPPLEMENTARY INFORMATION section of this document describes the NRC's bases for not offering a supplemental comment opportunity for any of the other technical issues that arose after the close of the public comment period on the proposed rule.

A. Regulatory Treatment of Nonsafety Systems (RTNSS)

The ESBWR safety analysis credits passive systems to perform safety functions for 72 hours following an initiating event. After 72 hours, nonsafety systems, either passive or active, replenish the passive systems in order to keep them operating or perform post-accident recovery functions directly. The ESBWR design also uses nonsafety-related active systems to provide defense-in-depth capabilities for key safety functions provided by passive systems. The challenge during the review was to identify the nonsafety SSCs that should receive enhanced regulatory treatment and to identify the appropriate regulatory treatment to be applied to these SSCs. Such SSCs are denoted as ''RTNSS SSCs'' in the context of the ESBWR design. As a result of the NRC's review, the applicant added Appendix 19A to the DCD to identify the nonsafety systems that perform these post-72 hour or defense-in-depth functions and the basis for their selection. The applicant's selection process was based on the guidance in SECY–94–084, ''Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs.''

To provide reasonable assurance that RTNSS SSCs will be available if called upon to function, the applicant established availability controls in DCD Tier 2, Appendix 19ACM, and TS in DCD Tier 2, Chapter 16, when required by 10 CFR 50.36, ''Technical specifications.'' The applicant also included all RTNSS SSCs in the reliability assurance program described in Chapter 17 of DCD Tier 2 and applied augmented design standards as described in DCD Tier 2, Section 19A.8.3. For the reasons set forth in Section 22.5 of the FSER, the NRC finds the applicant's treatment of the RTNSS SSCs, as described in the DCD, acceptable.

B. Containment Performance

The PCCS maintains the containment within its design pressure and temperature limits for DBAs. The system is passive and does not rely upon moving components or external power for initiation or operation for 72 hours following a loss-of-coolant accident (LOCA). The PCCS and its design basis are described in detail in Section 6.2.2 of the DCD Tier 2. The NRC identified a concern regarding the PCCS long-term cooling capability for the period from 72 hours to 30 days following a LOCA. To address this concern, the applicant proposed additional design features credited after 72 hours to reduce the long-term containment pressure. The features are the PCCS vent fans and passive autocatalytic hydrogen recombiners as described in DCD Tier 2, Section 6.2.1. These SSCs have been identified in DCD Appendix 19A as RTNSS SSCs.

The NRC staff's review of the PCCS design is documented in Section 6.2.2 of the FSER. The following is a summary of key points of that review. The applicant provided calculation results to demonstrate that the long-term containment pressure would be acceptable and that the design complies with GDC 38. The NRC's independent calculations confirmed the applicant's conclusion and the NRC accepts the proposed design and licensing basis. The NRC also raised a concern regarding the potential accumulation of high concentrations of hydrogen and oxygen in the PCCS and Isolation Condenser System, which could lead to combustion following a LOCA. The applicant modified the design of the PCCS and Isolation Condenser System heat exchangers to withstand potential hydrogen detonations. Accordingly, the NRC concludes that the design changes to the PCCS and Isolation Condenser System are acceptable and meet the applicable requirements.

C. Control Room Cooling

The ESBWR primarily relies on the mass and structure of the control building to maintain acceptable temperatures for human and equipment performance for up to 72 hours on loss of normal cooling. The NRC had not previously approved this approach for maintaining acceptable temperatures in the control building. The applicant proposed acceptance criteria for the evaluation of the control building structure's thermal performance based on industry and NRC guidelines. The applicant incorporated by reference an analysis of the control building structure's thermal performance as described in Tier 2, Sections 3H, 6.4, and 9.4. The applicant also proposed ITAACs to confirm that an updated analysis of the as-built structure continues to meet the thermal performance acceptance criteria. For the reasons set forth in Section 6.4.3 of the FSER, the NRC finds that the applicant's acceptance criteria are consistent with the advanced light water reactor control room envelope atmosphere temperature limits in NUREG–1242, ''NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document,'' and the use of the wet bulb globe temperature index in evaluation of heat stress conditions as described in NUREG–0700, ''Human-System Interface Design Review Guidelines.'' For the reasons set forth in Section 9.4.1 of the FSER, the NRC finds the control building structure thermal performance analysis and ITAACs acceptable based on the analysis using bounding environmental assumptions. Accordingly, the NRC finds that the acceptance criteria, control building structure thermal performance analysis, and the ITAACs, provide reasonable assurance that acceptable temperatures will be maintained in the control building for 72 hours. Therefore, the NRC finds that the control building design in regard to thermal performance conforms to the guidelines of SRP Section 6.4 and complies with the requirements of the GDC 19.

D. Feedwater Temperature Operating Domain

In operating BWRs, the recirculation pumps are used in combination with the control rods to control and maneuver reactor power level during normal power operation. The ESBWR design is unique in that the core is cooled by natural circulation during normal operation, and there are no recirculation pumps. In Chapter 15 of the DCD, GEH references licensing topical report (LTR) NEDO–33338, Revision 1, ''ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis.'' This LTR describes a broadening of the ESBWR operating domain, which allows for increased flexibility of operation by adjusting the feedwater temperature. This increased flexibility reduces the duty (mechanical stress) to the fuel and minimizes the probability of pellet-clad interactions and associated fuel failures.

By adjusting the feedwater temperature, the operator can control the reactor power level without control blade motion and with minimum impact on the fuel duty. Control blade maneuvering can also be performed at lower power levels.

To control the feedwater temperature, the ESBWR design includes a seventh feedwater heater with high-pressure steam. Feedwater temperature is controlled by either manipulating the main steam flow to the No. 7 feedwater heater to increase feedwater temperature above the temperature normally provided by the feedwater heaters with turbine extraction steam (normal feedwater temperature) or by directing a portion of the feedwater flow around the high-pressure feedwater heaters to decrease feedwater temperature below the normal feedwater temperature. An increase in feedwater temperature decreases reactor power, and a decrease in feedwater temperature increases reactor power. As described in Section 15.1.6 of the FSER, the applicant provided analyses that demonstrated ample margin to acceptance criteria. For the reasons set forth in Section 15.1.6 of the FSER, the NRC concludes that the applicant has adequately accounted for the effects of the proposed feedwater temperature operating domain extension on the nuclear design. Further, the applicant has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients and that the effects of postulated transients and accidents will not impair the capability to cool the core. Based on the evaluation documented in Section 15.1.6 of the FSER, the NRC concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable regulatory requirements.

E. Steam Dryer Analysis Methodology

As a result of RPV steam dryer issues at operating BWRs, the NRC issued revised guidance in Regulatory Guide (RG) 1.20, ''Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,'' and SRP Sections 3.9.2, ''Dynamic Testing and Analysis of Systems, Structures, and Components,'' and 3.9.5, ''Reactor Pressure Vessel Internals,'' for the evaluation of the structural integrity of steam dryers in BWR nuclear power plants. The guidance requested that applicants for BWR nuclear power plant design certifications, licenses, or license amendments perform analyses to demonstrate that the steam dryer will maintain its structural integrity during plant operation when experiencing acoustic and hydrodynamic fluctuating pressure loads. This demonstration of RPV steam dryer structural integrity consists of three general steps:

(1) Predict the fluctuating pressure loads on the steam dryer,

(2) Use these fluctuating pressure loads in a structural analysis to demonstrate the adequacy of the steam dryer design, and

(3) Implement a steam dryer monitoring program for confirming the steam dryer design analysis results during the initial plant power ascension testing and periodic steam dryer inspections.

In its March 2011 FSER, the NRC staff described its review of the GEH methodology used to demonstrate the steam dryer structural integrity as described in Revision 9 of the ESBWR DCD and four referenced topical reports on which the NRC staff had issued separate SERs. The NRC staff concluded that the methodology was technically sound and provided a conservative analytical approach for definition of flow-induced acoustic pressure loading on the steam dryer, and that the design provided assurance of the structural integrity of the steam dryer and demonstrated conformance with GDCs 1, ''Quality Standards and Records,'' 2 ''Design Bases for Protection Against Natural Phenomena,'' and 4, ''Environmental and Dynamic Effects Design Bases.'' The NRC received no public comments on the proposed rule with respect to the steam dryer analysis methodology.

Following the publication of the proposed rule, the NRC staff identified safety issues applicable to the ESBWR steam dryer structural analysis based on information obtained during the NRC's review of a license amendment request for a power uprate at an operating BWR nuclear power plant. Consequently, the NRC staff communicated to GEH in a letter dated January 19, 2012 (ADAMS Accession No. ML120170304), that it was concerned that the bases for its FSER on the ESBWR DCD and its SERs on several applicable GEH topical reports were no longer valid. Specifically, errors were identified in the benchmarking GEH used as a basis for determining fluctuating pressure loading on the steam dryer and errors were identified in a number of GEH's modeling parameters. The NRC staff subsequently issued requests for additional information (RAIs) and held multiple public meetings and non-public meetings (in which the NRC staff and GEH discussed GEH proprietary information) to clarify and discuss the safety issues with the ESBWR steam dryer analysis methodology. The NRC staff also conducted an audit of the GEH steam dryer analysis methodology at the GEH facility in Wilmington, North Carolina, in March 2012, and a vendor inspection, at that facility, of the quality assurance program for GEH engineering methods in April 2012.

To document the resolution of those issues, GEH revised the ESBWR DCD by removing references to its 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 following four LTRs were removed by GEH (public and proprietary versions cited):

- NEDE–33313 and NEDE–33313P, ''ESBWR Steam Dryer Structural Evaluation,'' all revisions
- NEDE–33312 and NEDE–33312P, ''ESBWR Steam Dryer Acoustic Load Definition,'' all revisions
- NEDC–33408 and NEDC–33408P, ''ESBWR Steam Dryer—Plant Based Load Evaluation Methodology,'' all revisions
- NEDC–33408, Supplement 1, and NEDC–33408P, Supplement 1, ''ESBWR Steam Dryer—Plant Based Load Evaluation Methodology Supplement 1,'' all revisions

To replace the information formerly provided by the four LTRs, GEH revised the ESBWR DCD to reference three new engineering reports (public and proprietary versions cited):

- NEDO–33312 and NEDE–33312P, Rev. 5, December 2013, ''ESBWR Steam Dryer Acoustic Load Definition''
- NEDO–33408 and NEDE–33408P, Rev. 5, December 2013, ''ESBWR Steam Dryer—Plant Based Load Evaluation Methodology—PBLE01 Model Description''
- NEDO–33313 and NEDE–33313P, Rev. 5, December 2013, ''ESBWR Steam Dryer Structural Evaluation''

GEH revised the following DCD sections to correct errors and provide additional information related to the design and evaluation of the structural integrity of the ESBWR steam dryer:

- Tier 1, Chapter 2, Section 2.1, ''Nuclear Steam Supply''
- Tier 1, Chapter 2, Section 2.1.1, ''Reactor Pressure Vessel and Internals''
- Tier 2, Chapter 1, Tables 1.6–1, 1.9–21, and 1D–1
- Tier 2, Chapter 3, Section 3.9.2, ''Dynamic Testing and Analysis of Systems, Components and Equipment''
- Tier 2, Chapter 3, Section 3.9.5, ''Reactor Pressure Vessel Internals''
- Tier 2, Chapter 3, Section 3.9.9, ''COL Information''
- Tier 2, Chapter 3, Section 3.9.10, ''References''
- Tier 2, Chapter 3, Appendix 3L, ''Reactor Internals Flow Induced Vibration Program''

The revisions to these documents enhance the detailed design and evaluation process related to the structural integrity of the ESBWR steam dryer in several ways. For example, the source of data used to benchmark the analysis methodology was modified in Revision 10 to the ESBWR DCD to a different operating nuclear power plant for which the NRC recently authorized an extended power uprate. In addition, the details of the design methodology were made more restrictive in several respects, including limiting the analysis methods for fillet welds and using more conservative data and assumptions. The changes also designate additional information as Tier 2* and clarify regulatory process steps for completing the detailed design and startup testing of the ESBWR steam dryer, including COL information items to be satisfied by a COL applicant, ITAACs to be met by a COL licensee, and model license conditions that may be proposed by a COL applicant.

The NRC staff reviewed the revised ESBWR DCD sections, new GEH engineering reports, and RAI responses and prepared an advanced supplemental SER to replace Section 3.9.5, ''Reactor Pressure Vessel Internals,'' of the original FSER. To maintain the description of the regulatory evaluation of all ESBWR reactor vessel internals in the same location, the advanced supplemental SER replaced the entire Section 3.9.5 in the original FSER, although only the ESBWR steam dryer discussion has been modified in the advanced supplemental SER in any significant respect. The advanced supplemental SER documents the NRC staff conclusion that Revision 10 to the ESBWR DCD and the referenced engineering reports provide sufficient information to support the adequacy of the design basis for the ESBWR reactor vessel internals. The advanced supplemental SER also documents the NRC staff conclusion that the design process for the ESBWR reactor vessel internals is acceptable and meets the requirements of 10 CFR part 50, appendix A, GDC 1, 2, 4, and 10; 10 CFR 50.55a; and 10 CFR part 52. Finally, the advanced supplemental SER documents the NRC staff conclusion that the ESBWR design documentation for the reactor vessel internals in Revision 10 to the ESBWR DCD is acceptable and provides the bases for the NRC staff conclusion that GEH's application for the ESBWR design certification meets the requirements of 10 CFR part 52, subpart B, that are applicable and technically relevant to the ESBWR standard plant design. The NRC adopts the above conclusions and finds, based on the application materials discussed in the FSER as modified by the advanced supplemental SER, that the ESBWR steam dryer design meets all applicable NRC requirements and may be incorporated by reference in a COL application.

The changes to the ESBWR steam dryer description in the DCD and supporting documentation may be regarded as significant changes which do not represent a ''logical outgrowth'' of the proposed rule and would therefore require an opportunity for public comment. To preclude any procedural challenges to the ESBWR final design certification rule in this area, the NRC staff published a supplemental proposed rule to provide an opportunity for public comment on these changes. The proposed rule and the supplemental proposed rule both provided an opportunity for public comment on the GEH evaluation methodology supporting the ESBWR steam dryer design. The NRC did not receive any comments on the proposed rule or the supplemental proposed rule related to the ESBWR steam dryer analysis methodology.

The NRC staff briefed the Advisory Committee for Reactor Safeguards (ACRS) Subcommittee on the ESBWR Design Certification on March 5, 2014, and the ACRS Full Committee on April 10, 2014, on its detailed review of the ESBWR steam dryer analysis methodology, including the significant improvements to the GEH Plant-Based Load Evaluation (PBLE01) methodology for the ESBWR steam dryer to resolve the technical issues with the reliability of the methodology. During the ACRS Subcommittee briefing, the Committee suggested that the NRC staff change the advanced supplemental SER to clarify the description of the steam dryer analysis methodology. Following the Full Committee meeting, the ACRS provided a letter to the Commission on April 17, 2014, that found that the ESBWR steam dryer design is adequate, and the associated structural analysis and planned startup test program are acceptable. In its letter, the ACRS noted that, ''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.''

In preparing the supplemental FSER referenced in this final rule (Supplement No. 1 to NUREG–1966), the NRC staff modified the advanced supplemental SER referenced in the supplemental proposed rule to reflect the changes suggested during the March 5, 2014, ACRS subcommittee meeting. These changes include: (1) Clarifying an inconsistency in referring to steam flow rates, (2) clarifying the acceptable methods for the analysis of the stress in the fillet welds in the ESBWR steam dryer caused by acoustic and hydrodynamic fluctuating pressure loads, and for the three allowable methods proposed by GEH to analyze the stress in fillet welds in the ESBWR steam dryer, clarifying the description of (a) the test problem used by GEH to demonstrate the adequacy of those methods, (b) the limitations in the specific GEH engineering report for application of those methods, and (c) the results of the test problem in demonstrating the acceptability of each of the three fillet weld analysis methods. In addition, the supplemental FSER includes a new section that provides the conclusion of the review by the ACRS of the ESBWR steam dryer analysis methodology. The NRC's regulatory basis for the acceptance of the ESBWR steam dryer analysis methodology remains the same in the supplemental FSER as provided in the advanced supplemental SER referenced in the supplemental proposed rule. In addition, the NRC staff corrected a variety of typographical, grammatical, and format errors in the advanced supplemental SER. The NRC staff also added appendices to the supplemental SER, each of which correspond to and augment the appendices in the FSER.

F. Aircraft Impact Assessment (AIA)

Under 10 CFR 50.150, which became effective on July 13, 2009, designers of new nuclear power reactors are required to perform an assessment of the effects on the designed facility of the impact of a large, commercial aircraft. An applicant for a new DCR is required to submit a description of the design features and functional capabilities identified as a result of the assessment (key design features) in its DCD together with a description of how the identified design features and functional capabilities show that the acceptance criteria in 10 CFR 50.150(a)(1) are met.

To address the requirements of 10 CFR 50.150, GEH completed an assessment of the effects on the designed facility of the impact of a large, commercial aircraft. GEH also added Appendix 19D to DCD Tier 2 to describe the design features and functional capabilities of the ESBWR identified as a result of the assessment that ensure the reactor core remains cooled and the SFP integrity is maintained. These design features and their functional capabilities are summarized as follows:

- The isolation condenser system provides core cooling.
- The emergency core cooling system provides core cooling.
- The main steam isolation system maintains high pressure for core cooling with the isolation condenser system.
- The CRD system inserts control rods to shut down the reactor. This enables core cooling with the systems described above.
- The digital control and instrumentation system actuates the CRD system to shut down the reactor and enable core cooling and initiates the automatic depressurization system and gravity-driven cooling system for core cooling at low pressure.
- The reinforced concrete containment vessel protects key design features located inside the vessel from structural and fire damage.
- The location and design of the reactor building structure, including exterior walls, interior walls, intervening structures inside the building and barriers on large openings in the exterior walls protect the reinforced concrete containment vessel from impact.
- The location and design of the turbine building structure protect the adjacent wall of the reactor building from impact.
- The location and design of the fuel building structure protect the adjacent wall of the reactor building from impact.
- The location and design of fire barriers inside the reactor building protect credited core cooling equipment from fire damage.
- The location (below grade) and design of SFP structure protect the SFP from impact.

The acceptance criteria in 10 CFR 50.150(a)(1) are: 1) the reactor core will remain cooled or the containment will remain intact; and 2) spent fuel pool cooling or spent fuel pool integrity is maintained. For the reasons set forth in Section 19.2.7 of the FSER, the NRC finds that the applicant has performed an aircraft impact assessment using an NRC-endorsed methodology that is reasonably formulated to identify design features and functional capabilities to show, with reduced use of operator action, that the acceptance criteria in 10 CFR 50.150(a)(1) are met. For the same reasons, the NRC finds that the applicant adequately described the key design features and functional capabilities credited to meet 10 CFR 50.150, including descriptions of how the key design features and functional capabilities show that the acceptance criteria in 10 CFR 50.150(a)(1) are met. Therefore, the NRC finds that the applicant meets the applicable requirements of 10 CFR 50.150(b).

G. ASME Code Case N–782

Under 10 CFR 50.55a(a)(3), GEH requested NRC approval for the use of ASME Code Case N–782, ''Use of Code Editions, Addenda, and Cases Section III, Division 1,'' as a proposed alternative to the rules of Section III, Subsection NCA–1140 regarding applied Code Editions and Addenda required by 10 CFR 50.55a(c), (d), and (e). ASME Code Case N–782 provides that the Code Edition and Addenda endorsed in a certified design or licensed by the regulatory authority may be used for systems and components subject to ASME Code, Section III requirements. These alternative requirements are in lieu of the requirements that base the Edition and Addenda solely on the date of an application for a construction permit and were issued to address new reactors licensed under 10 CFR part 52. Reference to ASME Code Case N–782 will be included in component and system design specifications and design reports to permit certification of these specifications and reports to the Code Edition and Addenda cited in the DCD. For the reasons set forth in Section 5.2.1.1.3 of the FSER, the NRC finds the use of ASME Code Case N–782 as a proposed alternative to the requirements of Section III, Subsection NCA–1140 under 10 CFR 50.55a(a)(3) acceptable for the ESBWR.

H. Exemption for the Safety Parameter Display System

The NRC is approving an exemption from 10 CFR 50.34(f)(2)(iv) as it relates to the safety parameter display system. This provision requires an applicant to provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, and is capable of displaying a full range of important plant parameters and data trends on demand and indicating when process limits are being approached or exceeded. The ESBWR design integrates the safety parameter display system into the design of the nonsafety-related distribution control and information system, rather than using a stand-alone console. For the reasons set forth in Section 18.8.3.2 of the FSER, the NRC finds that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that application of 10 CFR 50.34(f)(2)(iv) is not necessary to serve the underlying purpose of that rule in the context of the ESBWR design because the applicant has provided an acceptable alternative that accomplishes the purpose of the regulation. For the ESBWR, this purpose is accomplished by the plant alarm and display systems. In addition, the NRC finds that the proposed exemption is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security.

I. Hurricane-Generated Winds and Missiles

Nuclear power plants must be designed to withstand the effects of natural phenomena, including those that could result in the most severe wind events (tornadoes and hurricanes). The design bases for plant structures, systems, and components must reflect consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin to account for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. Initially, the U.S. Atomic Energy Commission, the predecessor to the NRC, considered tornadoes to be the bounding extreme wind events and issued RG 1.76, ''Design-Basis Tornado for Nuclear Power Plants,'' in April 1974, which reflected this technical position. RG 1.76 describes a design-basis tornado that a nuclear power plant should be designed to withstand without undue risk to the health and safety of the public. The design-basis tornado wind speeds were chosen so that the probability that a tornado exceeding the design-basis would occur was on the order of 10^{-7} per year per nuclear power plant.

In March 2007, the NRC issued Revision 1 of RG 1.76. Revision 1 of RG 1.76 relies on the Enhanced Fujita Scale, which was implemented by the National Weather Service in February 2007. The Enhanced Fujita Scale is a revised assessment relating tornado damage to wind speed, which resulted in a decrease in design-basis tornado wind speed criteria in Revision 1 of RG 1.76, although the probability that a tornado would exceed this reduced wind speed remained on the order of 10^{-7} per year per nuclear power plant. Because design-basis tornado wind speeds were decreased as a result of the analysis performed to update RG 1.76, it could no longer be assumed that the revised tornado design-basis wind speeds would bound design-basis hurricane wind speeds in all areas of the U.S. This prompted the NRC to research extreme wind gusts during hurricanes and their relationship to design-basis hurricane wind speeds, which resulted in the NRC developing a new regulatory guide, RG 1.221, ''Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants.''

RG 1.221 evaluates missile velocities associated with several types of missiles considered for different hurricane wind speeds. The hurricane missile analyses presented in RG 1.221 are based on missile aerodynamic and initial condition assumptions that are similar to those used for the analyses of tornado-borne missile velocities adopted for Revision 1 to RG 1.76. However, the assumed hurricane wind field differs from the assumed tornado wind field in that the hurricane wind field does not change spatially during the missile's flight time, but does vary with height above the ground. Because the size of the hurricane zone with the highest winds is large relative to the size of the missile trajectory, the hurricane missile is subjected to the highest wind speeds throughout its trajectory. In contrast, the tornado wind field is smaller, so the tornado missile is subject to the strongest winds only at the beginning of its flight. This results in the same missile having a higher maximum velocity in a hurricane wind field than in a tornado wind field with the same maximum (3-second gust) wind speed.

RG 1.221 was issued in final form in October 2011 (76 FR 63541). Thus, formal NRC adoption of RG 1.221 occurred after the June 7, 2011, close of the public comment period for the proposed ESBWR DCR, and well after completion of the NRC's review of the ESBWR DCD and the FSER for the ESBWR design in March 2011.

Tornado loads on SSCs are addressed in Section 3.3.2 of the ESBWR DCD. However, Section 3.3.2 of the ESBWR DCD does not explicitly state whether the loads that would be experienced during a hurricane would be bounded under the load analysis for tornadoes. Tornado-generated missiles are addressed in Section 3.5.1.4 of the ESBWR DCD. Section 3.5.1.4 of the ESBWR DCD states that ''tornado generated missiles are determined to be the limiting natural phenomena hazard in the design of all structures required for safe shutdown of the nuclear power plant. Because tornado missiles are used in the design basis, they envelop missiles generated by less intense phenomena such as extreme winds.'' The DCD also provides the design-basis tornado and missile spectrum in Tier 1, Table 5.1–1 and Tier 2, Table 2.0–1, and states its conformance with certain positions in RGs 1.13, 1.27, 1.76, and 1.117.

Thus, the ESBWR applicant has not addressed, and the NRC has not specifically determined, whether the

ESBWR design is in conformance with GDCs 2 and 4 for hurricane wind and missile loads that are not bounded by the total tornado loads analyzed in the DCD. For these reasons, the NRC is only making a final safety determination on the acceptability of the ESBWR design with respect to loads on the applicable SSCs from hurricane winds and hurricane-generated missiles that are bounded by other loads analyzed in the DCD.

Accordingly, the NRC is excluding two issues from issue finality and issue resolution in the ESBWR DCD. First, with respect to the scope of the design in Section 3.3.2 of the ESBWR DCD, the NRC is excluding from finality the narrow issue of loads on applicable SSCs from hurricanes, but only to the extent that such loads are not bounded by other loads analyzed in the ESBWR DCD. Second, with respect to the scope of the design in Section 3.5.1.4 of the ESBWR DCD, the NRC is excluding from finality the narrow issue of loads on applicable SSCs from hurricane-generated missiles, but only to the extent that such loads are not bounded by other loads analyzed in the ESBWR DCD. This is accomplished in paragraph A.2.g of Section IV, ''Additional Requirements and Restrictions,'' and paragraph B.1 of Section VI, ''Issue Resolution,'' of the new appendix E to 10 CFR part 52, by excluding loads from hurricane winds and hurricane-generated missiles on the applicable SSCs from the finality accorded to the ESBWR design if they are not bounded as described. Under the exclusion, a COL applicant referencing the ESBWR DCR must demonstrate that loads from site-specific hurricane winds and hurricane-generated missiles are bounded by the total tornado load as analyzed in the ESBWR DCD. If the total tornado load analyses are not bounding, the COL applicant has several ways of addressing the exclusion, for example, demonstrating that the design can withstand the hurricane wind loads and hurricane-generated missile loads.

The NRC's narrow exclusion with respect to issue finality, as reflected in the ESBWR DCR language, does not require any change to the ESBWR design, the ESBWR DCD, or the NRC's EA supporting the ESBWR rulemaking. Nor are any changes required to the associated analyses for total tornado loads as described in the ESBWR DCD.

J. Loss of One or More Phases of Offsite Power

Bulletin 2012–01, ''Design Vulnerability in Electric Power System,'' as applied to passive plant designs such as the ESBWR, addresses the need for electric power system designs to be able to detect the loss of one or more of the three phases of an offsite power circuit connected to the plant electrical systems and provide an alarm in the control room. Bulletin 2012–01 was issued after the proposed rule was issued and the public comment period closed. In its response to Bulletin 2012–01, GEH provided additional details on the monitoring and alarm functions for all three phases of the offsite power circuits and included applicable information in Revision 10 to the DCD. GEH also added new ITAACs to ensure implementation of these design features by a COL holder. The NRC staff reviewed the ESBWR design features that can detect and provide an alarm for the loss of one or more of the three phases of an offsite power circuit. For the reasons set forth in Section 8.2.3, ''Staff Evaluation,'' of the supplemental FSER, the NRC concludes that no design vulnerability identified in Bulletin 2012–01 exists in the ESBWR electric power system.

K. Spent Fuel Assembly Integrity in Spent Fuel Racks

Prior to publishing the proposed rule, the NRC performed its review of the integrity of spent fuel racks based on SRP Section 9.1.2, ''New and Spent Fuel Storage.'' This section states that ''Designing the storage pool and fuel storage racks to meet seismic Category I requirements provides reasonable assurance that earthquakes will not cause a substantial coolant loss, a reduction in margin to criticality, or damage to the fuel assemblies.'' This section supports the NRC's requirements in GDC 2, which requires that nuclear power plant SSCs important to safety be designed to withstand the effects of natural phenomena, such as an earthquake without loss of capability to perform their safety functions. The ESBWR FSER concluded that the design of the SFP, the buffer pool, and the fuel storage racks complied with the requirements of GDC 2 and met the guidance of SRP Section 9.1.2.

After publication of the proposed rule, the NRC recognized that Appendix D, ''Guidance on Spent Fuel Racks,'' to SRP Section 3.8.4, ''Other Seismic Category I Structures,'' states that, ''It should be demonstrated that the consequent loads on the fuel assembly do not lead to damage of the fuel.'' In other words, though the spent fuel rack may have remained intact during a seismic event, because there are gaps between the rack and the fuel assemblies, the applicant should demonstrate that the spent fuel assemblies in the rack have not sustained damage during that seismic event. During the NRC staff's review of the ESBWR design and prior to its publication of its FSER, the NRC staff did not specifically review the design of the spent fuel in the spent fuel racks against this guidance, but only against that of SRP Section 9.1.2 as described above.

To confirm the structural integrity of the fuel in the spent fuel racks, the NRC staff conducted an audit on August 5 and September 8, 2011. The audit summary is available under ADAMS Accession No. ML112860614. GEH subsequently submitted additional information (ADAMS Accession No. ML11269A093) to address whether the consequent loads on the fuel assembly that result from the design-basis seismic event would lead to fuel damage. For the reasons set forth in Section 3.8.4 of the supplemental FSER, the NRC finds that the fuel assemblies maintain structural integrity when subject to the design-basis seismic loads, the fuel assemblies in the fuel storage racks are structurally adequate to withstand the design-basis seismic loads, and the fuel assemblies are in compliance with GDC 2.

L. Turbine Building Offgas System Design Requirements

Regulatory Guide (RG) 1.143, ''Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,'' provides guidance on classifying and designing radioactive waste management systems (RWMSs). The Offgas System (OGS), which is part of the Gaseous Waste Management System, is classified as a Category RW–IIa (High Hazard) RWMS in accordance with RG 1.143. Following publication of the proposed rule, the NRC staff identified that while it had evaluated the OGS against the guidelines of RG 1.143, the NRC staff had not evaluated the structure housing the OGS (i.e., the turbine building), against the guidelines of RG 1.143. Subsequently, the NRC staff reviewed the information included in various sections of the ESBWR DCD regarding protection of the OGS. For the reasons set forth in Section 3.8.4.3 of the supplemental FSER, the NRC finds that the turbine building structure provides adequate protection for the OGS components to meet the design criteria in RG 1.143 for Category RW–IIa.

Because the NRC staff's evaluation of the turbine building structure came after completion of the FSER, issuance of the final SDA, and publication of the proposed rule, the NRC decided to document the NRC staff's review on this issue in the supplemental FSER. The evaluation was performed using information already included in Revision 9 of the ESBWR DCD and that information did not change in Revision 10 of the DCD. Further, the NRC determined that no changes were required to the ESBWR DCD, the proposed rule text, or the EA supporting this rulemaking.

M. ASME BPV Code Statement in Chapter 1 of the ESBWR DCD

In Revision 10 to the ESBWR DCD, Tier 1, Section 1.1.1, ''Definitions,'' the applicant added a definition of ''ASME Code'' to its Tier 1 definitions. This addition addressed compliance with the ASME BPV Code and the use of alternatives to the ASME BPV Code requirements as permitted in 10 CFR 50.55a(a)(3). For the ESBWR DCR, several ITAACs in the ESBWR Tier 1 are required to verify that ASME BPV Code, Section III construction requirements have been met. During actual construction of a nuclear power plant, it is inevitable that departures from the ASME BPV Code construction requirements will be needed. These departures occur for various reasons such as unavailability of material, hardship in implementing fabrication sequences required by the Code, and the availability of newer and more effective construction techniques. As such, the regulations in 10 CFR 50.55a, ''Codes and standards,'' provide for the use of alternatives to Section III construction requirements to overcome such hardships and allow a degree of flexibility in constructing nuclear power plants without compromising safety requirements. Pursuant to 10 CFR 50.55a(a)(3), proposed alternatives to Section III requirements may be used when authorized by the NRC. Before using these alternatives, the applicant or licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements of 10 CFR 50.55a would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

During the construction of two nuclear power plants licensed under 10 CFR part 52 (Vogtle Electric Generating Plant, Units 3 and 4, and V.C. Summer Nuclear Station, Units 2 and 3), the question arose whether changes to ASME BPV Code requirements, such as the use of alternatives in accordance with 10 CFR 50.55a(a)(3), are permitted without the need to submit an exemption from the regulations pursuant to 10 CFR 50.12, ''Specific exemptions.'' The NRC staff found that this issue was previously discussed in the **SUPPLEMENTARY INFORMATION** section of a final rule dated August 28, 2007, amending the regulations to address 10 CFR part 52 requirements (72 FR 49352). Therein, the NRC stated in Section VI, ''Section-by-Section Analysis,'' for Section 52.7, ''Specific Exemptions,'' (at 72 FR 49438) that, ''§ 52.7 does not supersede the applicability of more specific dispensation provisions in other parts of Chapter I. For example, a holder of a COL would not require a separate part 52 exemption in order to obtain approval of an alternative to a provision of an applicable ASME Code provision that is otherwise required under 10 CFR 50.55a; the licensee need only satisfy the criteria in § 50.55a(a)(3) . . .'' The 2007 10 CFR part 52 final rule **SUPPLEMENTARY INFORMATION** clarified that using alternatives to ASME Code requirements authorized in accordance with 10 CFR 50.55a is sufficient and does not require a COL holder to submit an exemption when changes involve a departure from only ASME Code requirements.

To clarify the use of alternatives when verifying compliance with ASME BPV Code ITAACs, GEH proposed to clarify in its Tier 1 definitions in Revision 10 to the ESBWR DCD, Section 1.1.1, ''Definitions,'' that ''ASME Code'' means ASME BPV Code requirements or any alternative authorized by the NRC pursuant to 10 CFR 50.55a(a)(3). This change does not affect previous NRC safety findings in the FSER or change the status of how the ESBWR standard design complies with ASME BPV Code requirements. For the reasons set forth in Section 14.3 of the supplemental FSER, the NRC finds that these changes to the definition of ASME Code are acceptable.

N. Clarification of ASME Component Design ITAACs

Following the publication of the proposed rule, the NRC staff reviewed ITAACs for inspectability and consistency across several design certifications. This review identified the potential issue that the ITAACs related to verification of component design, as written in Revision 9 of the ESBWR DCD, might be viewed as requiring design verification of as-designed ASME BPV Code components, rather than as-built ASME BPV Code components, as originally intended. Verifying interim ASME BPV Code design reports at the design stage would result in an unnecessary regulatory burden with no benefit to safety. In Revision 10 of the ESBWR DCD, the ASME BPV Code component ITAACs were revised to clarify that the activities needed to satisfy the ITAACs are performed at the as-built stage. For the reasons set forth in Section 14.3.3 of the supplemental FSER, the NRC concludes that this clarification promotes efficient ITAAC closure and reduces potential confusion while having no effect on previous NRC safety findings.

O. Corrections, Editorial, and Conforming Changes

GEH made corrections and editorial changes in Revision 10 of the DCD. The NRC corrected typographical errors, made other editorial changes, and added units of measurements to the advanced supplemental SER. The NRC also revised the advanced supplemental SER after publication of the supplemental proposed rule to include conforming changes such as adding appendices that augment the appendices in the FSER.

V. Rulemaking Procedure

A. Exclusions From Issue Finality and Issue Resolution for Spent Fuel Pool Instrumentation

As described in Section III of the **SUPPLEMENTARY INFORMATION** section of this document related to how the ESBWR design addresses Fukushima NTTF recommendations, the NRC is changing the ESBWR DCR language to exclude from finality the safety-related SFP level instruments: (1) Being designed to allow the connection of an independent power source, and (2) maintaining its design accuracy following a power interruption or change in power source without recalibration. There was no change to the ESBWR design, as described in the DCD, the NRC's EA supporting the ESBWR rulemaking (and in particular, the SAMDA analysis), or the ESBWR FSER. In addition, the final rule is more conservative than the proposed rule because it is more limiting both as to what is certified and to the scope of issue finality. The NRC is not aware of any entity other than the applicant, GEH, who would be adversely affected by this change. With respect to the exclusions, GEH voluntarily declined to submit additional information that would avoid the need for exclusions from issue finality and issue resolution on this matter. The NRC did not receive any public comments in the area of spent fuel pool instrumentation (which otherwise would suggest public interest in this matter). For these reasons, the NRC staff concluded that a supplemental opportunity for public comment was not warranted for these exclusions from issue finality and issue resolution.

B. Incorporation by Reference of Public Documents

The change to the ESBWR DCR language related to approval for incorporation by reference by the Office of the Federal Register of 20 publicly-available documents is described in Section III of the **SUPPLEMENTARY INFORMATION** section of this document. The supplemental proposed rule discussed the changes to the ESBWR DCR language but deferred the discussion of why a public comment opportunity was not provided to the final rule. The NRC did not offer a supplemental opportunity for public comment on this matter for the following reasons. First, the text of the DCD—when discussing each of the 20 publicly-available documents—makes clear that these are intended to be requirements. Thus, a member of the public could have discerned and commented on the failure of Tables 1.6–1 and 1.6–2 of the Revision 9 of the DCD to differentiate between documents intended to be requirements (given the information presented throughout DCD Revision 9) and documents which were intended only to be references (*i.e.,* ''for information only''). The public could also have commented on the discrepancy between the language of Revision 9 of the DCD (which regards these documents as being incorporated by reference into the DCD) and the failure of the proposed ESBWR design certification rule to list the publicly-available referenced documents as being approved by the Office of the Federal Register for incorporation by reference. Finally, the NRC did not receive any comments on the proposed rule with respect to Tables 1.6–1 and 1.6–2 in Revision 9 of the DCD, or the incorporation by reference language in Section III of proposed Appendix E to part 52 (which otherwise would suggest public interest in this matter). For these reasons, the NRC staff concluded that a supplemental opportunity for public comment was not warranted with respect to the status of the 20 documents as requirements and their incorporation by reference into the ESBWR design certification rule.

C. Changes to Tier 2 Information*

The final rule includes three changes from the proposed rule regarding Tier 2* matters under Section VIII of the ESBWR rule language as described in Section III of the **SUPPLEMENTARY INFORMATION** section of this document. Because one of those changes was related to the steam dryer, and for the same reasons as the steam dryer analysis methodology being offered a supplemental opportunity for public comment, the related Tier 2* change was included in the supplemental proposed rule and no public comments were received on this topic. The other two Tier 2* changes—related to the specific subsections of ASME BPV Code and a correction to the type of valves used in the ESBWR design—were included for consistency with the ESBWR design as described in the DCD. First, paragraph VIII.B.6.c.(1) is changed from ''ASME Boiler and Pressure Vessel Code, Section III'' to ''ASME Boiler and Pressure Vessel Code, Section III, Subsections NE (Division 1) and CC (Division 2) for containment vessel design.'' The NRC determined that no changes were required to the ESBWR design or the DCD; rather, the change to the rule text is needed to make the rule consistent with Revisions 9 and 10 of the ESBWR DCD. Further, the change represents a restriction as compared to the proposed rule language. That is, the proposed rule would allow the larger scope of Tier 2* information with respect to ASME BPV Code, Section III to revert to Tier 2 after full power, whereas the change to the final rule does not allow containment vessel design information subject to Subsection NE., Division 1, and Subsection CC, Division 2, to revert to Tier 2 after the plant first achieves full power following the finding required by 10 CFR 52.103(g). Therefore, the NRC concludes that a supplemental opportunity for public comment on these changes to the rule is not warranted.

Second, paragraph VIII.B.6.c.(3) is changed from ''Motor-operated valves'' to ''Power-operated valves.'' The NRC determined that no changes were required to the ESBWR design or the DCD; rather, the change to the rule text is needed to make the rule consistent with Revisions 9 and 10 of the ESBWR DCD. Further, the change to the rule text is corrective in nature and does not represent a substantive change to the nature of Tier 2* matters. Therefore, the NRC concludes that a supplemental opportunity for public comment on these changes to the rule is not warranted.

D. Other Changes to the ESBWR Rule Language and Difference From Other DCRs

The ESBWR final rule language differs from the proposed rule language in several areas that are administrative or clarifying and do not involve any substantive change. Those differences, and the rationale for the differences, are as follows. Paragraph III.A, which describes the document being incorporated by reference and how to examine or obtain copies of that document, was revised to conform to other recently issued DCRs and to the Office of the Federal Register's guidance. Paragraphs III.D and V.A were revised to include the NUREG number for the FSER; the NUREG was not available when the NRC published the ESBWR proposed rule. Paragraphs IV.A.3, VI.E, and X.A.1 were administratively revised to remove acronyms for SUNSI and SGI but retain the terms that these acronyms represent for consistency with other DCRs. For paragraph VI.E, footnoted text was moved into the body of the regulation where these terms were noted. Paragraph V.B.1 was revised to clarify that, similar to the regulations that apply to the ESBWR design in Paragraph V.A, the regulations that the ESBWR design is exempt from are those codified as of the date the final rule is signed by the Secretary of the Commission. Because these changes are administrative in nature, the NRC concluded that a supplemental opportunity for public comment was not warranted for these matters.

ESBWR final rule language differs from the rule language of other DCRs in several areas that are not otherwise explained in the preceding paragraph. Those differences, and the rationale for the differences, are as follows. Paragraph II.B was administratively revised to include the term ''generic TS,'' similar to that of ''generic DCD'' in Paragraph II.A, as it is used in appendix E. Paragraph II.C was revised to clarify the actual content of a plant-specific DCD. Paragraph IV.A.2.a was revised to provide flexibility to COL applicants by updating the process by which a COL applicant can reference information in the generic DCD—either by including that information or incorporating it by reference; current DCRs are silent as to how to include this information. Paragraphs IV.A.2.d and VI.B.7 were revised to conform to other NRC regulations regarding site characteristics for a COL, postulated site parameters for a certified design, and the interface requirements. Finally, paragraph IX was reserved for future use because the substantive requirements in this paragraph (for other DCRs) has since been incorporated into 10 CFR part 52 in a 2007 rulemaking (72 FR 49352; August 28, 2007) and thus are no longer needed in the four existing DCR appendices. The NRC intends to remove these requirements from Section IX of the four existing DCR appendices in future amendment(s) separate from this rulemaking. Because these are administrative in nature, the NRC concluded that a supplemental opportunity for public comment was not warranted for these matters.

E. Exclusions From Issue Finality and Issue Resolution for Hurricane-Generated Winds and Missiles

As described in Section IV of the **SUPPLEMENTARY INFORMATION** section of this document, the final rule contains exclusions from issue finality and issue resolution related to hurricane-generated winds and missiles. The ESBWR design, as described in the DCD, the NRC's EA supporting the ESBWR rulemaking (and in particular, the SAMDA analysis), and the ESBWR FSER did not change. In addition, the change to the final rule is more conservative than the proposed rule because it is more limiting as to what is certified and the scope of issue finality. The NRC is not aware of any entity other than the applicant, GEH, who would be adversely affected by this change. With respect to the exclusions, GEH voluntarily declined to submit additional information which would avoid the need for exclusions from issue finality and issue resolution on this matter. The NRC did not receive any public comments on hurricane winds or hurricane missiles (which otherwise would suggest public interest in this matter). For these reasons, the NRC staff concluded that a supplemental opportunity for public comment was not warranted for these exclusions from issue finality and issue resolution.

F. Loss of One or More Phases of Offsite Power

The changes that GEH made to the DCD and the NRC staff conclusions in its supplemental FSER to clarify how the ESBWR design addresses the loss of one or more phases of offsite power in order to demonstrate compliance with GDC 17, ''Electric Power Systems,'' are described in Section IV of the **SUPPLEMENTARY INFORMATION** section of this document. These changes did not require a change to the rule text or to the EA supporting this rulemaking. The NRC did not receive any public comments on the proposed rule with respect to the adequacy of the offsite power system (which would otherwise suggest public interest in this matter). For these reasons, the NRC staff concluded that a supplemental opportunity for public comment was not warranted for this matter.

G. Spent Fuel Assembly Integrity in Spent Fuel Racks

The discussion in the supplemental FSER related to spent fuel assembly integrity in spent fuel racks is described in Section IV of the **SUPPLEMENTARY INFORMATION** section of this document. The NRC staff determined that the additional information provided by GEH did not require a change to the design of the fuel or the spent fuel racks as described in Revision 9 of the ESBWR DCD or new design commitments in the DCD. No changes were required to the ESBWR DCD, the rule text, or the EA supporting this rulemaking. The NRC did not receive any public comments on the proposed rule with respect to spent fuel pool assembly integrity (which otherwise would suggest public interest in this matter). For these reasons, the NRC staff concluded that a supplemental opportunity for public comment was not warranted for this matter, including the supplemental FSER.

H. Turbine Building Offgas System Design Requirements

The NRC staff's evaluation of the turbine building structure relative to the Turbine Building Offgas System design requirements, as documented in a supplemental FSER, is described in Section IV of the **SUPPLEMENTARY INFORMATION** section of this document. The staff's evaluation, which was not documented in the March 2011 FSER, was performed using information in Revision 9 of the ESBWR DCD that did not change in Revision 10 of the DCD. Further, there were no changes required to the ESBWR DCD, the rule text, or the EA supporting this rulemaking. The NRC did not receive any public comments on the proposed rule with respect to the Turbine Building Offgas System (which otherwise would suggest public interest in this matter). For these reasons, the NRC staff concluded that a supplemental opportunity for public comment was not warranted for this matter.

I. ASME BPV Code Statement in Chapter 1 of the ESBWR DCD

The technical clarification to the DCD and supplemental FSER related to the ASME BPV Code statement in Chapter 1 of the ESBWR DCD is described in Section IV of the **SUPPLEMENTARY INFORMATION** section of this document. This clarification does not affect previous NRC safety findings in the FSER, change the ESBWR's compliance with Code requirements, or require changes to the rule text for this rulemaking. For these reasons, the NRC staff concluded that a supplemental opportunity for public comment was not warranted for this matter.

J. Clarification of ASME Component Design ITAACs

The technical clarifications that GEH made to the DCD and the staff's conclusions in its supplemental FSER regarding the ASME component design ITAACs are described in Section IV of the **SUPPLEMENTARY INFORMATION** section of this document. This clarification does not affect previous NRC safety findings in the FSER, nor does it require changes to the rule text for this rulemaking. For these reasons, the NRC staff concluded that a supplemental opportunity for public comment was not warranted for this matter.

K. Changes to the Supplemental FSER After Publication of the Supplemental Proposed Rule

The advanced supplemental SER was issued on April 17, 2014 (ADAMS Accession No. ML14043A134). After the supplemental proposed rule was issued, and to reflect the changes suggested during the March 5, 2014, ACRS subcommittee meeting, the NRC revised the advanced supplemental SER and prepared it as a supplement to the FSER. In this revision the NRC clarified the discussion of the ESBWR steam dryer analysis methodology regarding Methods 1, 2, and 3 in Section 3.9.5.3.3.5.2.3. In addition, the supplemental FSER includes a new section that provides the conclusion of the review by the ACRS of the ESBWR steam dryer analysis methodology. The NRC staff's regulatory basis for the acceptance of the ESBWR steam dryer analysis methodology remains the same in the supplemental FSER as provided in the advanced supplemental SER referenced in the supplemental proposed rule. For this reason, the NRC staff concluded that a supplemental opportunity for public comment was not warranted for this matter. The supplemental FSER (ADAMS Accession No. ML14155A333) will be published as Supplement No. 1 to NUREG 1966. NUREG–1966 was published in April 2014 (ADAMS Accession No. ML14100A304).

L. Corrections, Editorial, and Conforming Changes

GEH made editorial changes in Revision 10 of the DCD. The NRC corrected typographical errors, made other editorial changes, and added units of measurements to the advanced supplemental SER. The NRC staff also revised the advanced supplemental SER after publication of the supplemental proposed rule to include conforming changes such as adding appendices that augment the appendices in the FSER. Because these changes are administrative in nature, the NRC staff concluded that a supplemental opportunity for public comment was not warranted for these matters.

VI. Planned Withdrawal of the ESBWR SDA

In its application (ADAMS Accession No. ML052450245), GEH requested the NRC provide its design approval for the ESBWR design. The SDA for the ESBWR design was issued in March 2011 (ADAMS Accession No. ML110540310) after the completion of the FSER. In a letter dated June 3, 2014 (ADAMS Accession No. ML14154A094), GEH requested that the NRC retire the SDA at the time of issuance of the final ESBWR DCR. In accordance with GEH's request, the NRC plans to issue a **Federal Register** notice announcing the withdrawal of the ESBWR SDA after the effective date of the final ESBWR design certification rule.

VII. Section-by-Section Analysis

The following discussion sets forth the purpose and key aspects of each section and paragraph of the final ESBWR DCR. All section and paragraph references are to the provisions in appendix E to 10 CFR part 52 unless otherwise noted. The NRC has modeled the ESBWR DCR on the existing DCRs, with certain modifications where necessary to account for differences in the ESBWR design documentation, design features, and EA (including SAMDAs). As a result, the DCRs are standardized to the extent practical.

A. Introduction (Section I)

The purpose of Section I of appendix E to 10 CFR part 52 (this appendix) is to identify the standard plant design that would be approved by this DCR and the applicant for certification of the standard design. Identification of the design certification applicant is necessary to implement this appendix for two reasons. First, the implementation of 10 CFR 52.63(c) depends on whether an applicant for a COL contracts with the design certification applicant to provide the generic DCD and supporting design information. If the COL applicant does not use the design certification applicant to provide the design information and instead uses an alternate nuclear plant vendor, then the COL applicant must meet the requirements in 10 CFR 52.73. The COL applicant must demonstrate that the alternate supplier is qualified to provide the standard plant design information. Second, paragraph X.A.1 requires the design certification applicant to maintain the generic DCD throughout the time this appendix may be referenced. Thus, it is necessary to identify the entity to which the requirement in paragraph X.A.1 applies.

B. Definitions (Section II)

During development of the first two DCRs, the NRC decided that there would be both generic (master) DCDs maintained by the NRC and the design certification applicant, as well as individual plant-specific DCDs maintained by each applicant and licensee that reference this appendix. This distinction is necessary in order to specify the relevant plant-specific requirements to applicants and licensees referencing the appendix. In order to facilitate the maintenance of the master DCDs, the NRC requires that each application for a standard design certification be updated to include an electronic copy of the final version of the DCD. The final version is required to incorporate all amendments to the DCD submitted since the original application, as well as any changes directed by the NRC as a result of its review of the original DCD or as a result of public comments. This final version is the master DCD incorporated by reference in the DCR. The master DCD would be revised as needed to include generic changes to the version of the DCD approved in this design certification rulemaking. These changes would occur as the result of generic rulemaking by the Commission, under the change criteria in Section VIII.

The NRC also requires each applicant and licensee referencing this appendix to submit and maintain a plant-specific DCD as part of the COL FSAR. This plant-specific DCD must either include or incorporate by reference the information in the generic DCD. The plant-specific DCD would be updated as necessary to reflect the generic changes to the DCD that the Commission may adopt through rulemaking, plant-specific departures from the generic DCD that the Commission imposed on the licensee by order, and any plant-specific departures that the licensee chooses to make in accordance with the relevant processes in Section VIII. Thus, the plant-specific DCD functions like an updated FSAR because it would provide the most complete and accurate information on a plant's design-basis for that part of the plant within the scope of this appendix. Therefore, this appendix defines both a generic DCD and a plant-specific DCD.

Also, the NRC is treating the TS in Chapter 16 of the generic DCD as a special category of information and designating them as generic TS in order to facilitate the special treatment of this information under this appendix. A COL applicant must submit plant-specific TS that consist of the generic TS, which may be modified under paragraph VIII.C, and the remaining plant-specific information needed to complete the TS. The FSAR that is required by 10 CFR 52.79 will consist of the plant-specific DCD, the site-specific portion of the FSAR, and the plant-specific TS.

The terms Tier 1, Tier 2, Tier 2*, and COL action items (license information) are defined in this appendix because these concepts were not envisioned when 10 CFR part 52 was developed. The design certification applicants and the NRC used these terms in implementing the two-tiered rule structure that was proposed by representatives of the nuclear industry after issuance of 10 CFR part 52. Therefore, appropriate definitions for these additional terms are included in this appendix. The nuclear industry representatives requested a two-tiered structure for the DCRs to achieve issue preclusion for a greater amount of information than was originally planned for the DCRs, while retaining flexibility for design implementation. The Commission approved the use of a two-tiered rule structure in its SRM, dated February 14, 1991, on SECY–90–377, ''Requirements for Design Certification under 10 CFR Part 52,'' dated November 8, 1990. This document and others are available in the Regulatory History of Design Certification (see Section VII of this document).

The Tier 1 portion of the design-related information contained in the DCD is *certified* by this appendix and, therefore, subject to the special backfit provisions in paragraph VIII.A. An applicant who references this appendix is required to include or incorporate by reference and comply with Tier 1, under paragraphs III.B and IV.A.1. This information consists of an introduction to Tier 1, the system based and non-system based design descriptions and corresponding ITAACs, significant interface requirements, and significant site parameters for the design (refer to Section C.I.1.8 of RG 1.206 for guidance on significant interface requirements and site parameters). The design descriptions, interface requirements, and site parameters in Tier 1 were derived from Tier 2, but may be more general than the Tier 2 information. The NRC staff's evaluation of the Tier 1 information is provided in Section 14.3

of the FSER. Changes to or departures from the Tier 1 information must comply with Section VIII.A.

The Tier 1 design descriptions serve as requirements for the lifetime of a facility license referencing the design certification. The ITAACs verify that the as-built facility conforms to the approved design and applicable regulations. Under 10 CFR 52.103(g), the Commission must find that the acceptance criteria in the ITAACs are met before authorizing operation. After the Commission has made the finding required by 10 CFR 52.103(g), the ITAACs do not constitute regulatory requirements for licensees or for renewal of the COL. However, subsequent modifications to the facility within the scope of the design certification must comply with the design descriptions in the plant-specific DCD unless changes are made under the change process in Section VIII. The Tier 1 interface requirements are the most significant of the interface requirements for systems that are wholly or partially outside the scope of the standard design. Tier 1 interface requirements must be met by the site-specific design features of a facility that references this appendix. An application that references this appendix must demonstrate that the site characteristics at the proposed site fall within the site parameters (both Tier 1 and Tier 2) (refer to paragraph V.D of this document).

Tier 2 is the portion of the design-related information contained in the DCD that is *approved* by this appendix but not certified. Tier 2 information is subject to the backfit provisions in paragraph VIII.B. Tier 2 includes the information required by 10 CFR 52.47(a) and 52.47(c) (with the exception of generic TS and conceptual design information) and the supporting information on inspections, tests, and analyses that will be performed to demonstrate that the acceptance criteria in the ITAACs have been met. As with Tier 1, paragraphs III.B and IV.A.1 require an applicant who references this appendix to include or incorporate by reference Tier 2 and to comply with Tier 2, except for the COL action items, including the availability controls in Appendix 19ACM of the generic DCD. The definition of Tier 2 makes clear that Tier 2 information has been determined by the NRC, by virtue of its inclusion in this appendix and its designation as Tier 2 information, to be an approved sufficient method for meeting Tier 1 requirements. However, there may be other acceptable ways of complying with Tier 1 requirements. The appropriate criteria for departing from Tier 2 information are specified in paragraph VIII.B. Departures from Tier 2 information do not negate the requirement in paragraph III.B to incorporate by reference Tier 2 information.

A definition of ''combined license action items'' (COL information), which is part of the Tier 2 information, has been added to clarify that COL applicants who reference this appendix are required to address COL action items in their license application. However, the COL action items are not the only acceptable set of information. An applicant may depart from or omit COL action items, provided that the departure or omission is identified and justified in the FSAR. After issuance of a construction permit or COL, these items are not requirements for the licensee unless they are restated in the FSAR. For additional discussion, see Section V.D of this document.

The availability controls, which are set forth in Appendix 19ACM of the generic DCD, were added to the information that is part of Tier 2 to clarify that the availability controls are not operational requirements for the purposes of paragraph VIII.C. Rather, the availability controls are associated with specific design features. The availability controls may be changed if the associated design feature is changed under paragraph VIII.B. For additional discussion, see Section V.C of this document.

Certain Tier 2 information has been designated in the generic DCD with brackets and italicized text as ''Tier 2*'' information and, as discussed in greater detail in the section-by-section analysis for Section H, a plant-specific departure from Tier 2* information requires prior NRC approval. However, the Tier 2* designation expires for some of this information when the facility first achieves full power after the finding required by 10 CFR 52.103(g). The process for changing Tier 2* information and the time at which its status as Tier 2* expires is set forth in paragraph VIII.B.6. Some Tier 2* requirements concerning special preoperational tests are designated to be performed only for the first plant or first three plants referencing the ESBWR DCR. The Tier 2* designation for these selected tests will expire after the first plant or first three plants complete the specified tests. However, a COL action item requires that subsequent plants also perform the tests or justify that the results of the first-plant-only or first-three-plants-only tests are applicable to the subsequent plant.

The regulations at 10 CFR 50.59 set forth thresholds for permitting changes to a plant as described in the FSAR without NRC approval. Inasmuch as 10 CFR 50.59 is the primary change mechanism for operating nuclear plants, the NRC has determined that future plants referencing the ESBWR DCR should use thresholds as close to 10 CFR 50.59, as is practicable and appropriate for new reactors. Because of some differences in how the change control requirements are structured in the DCRs, certain definitions contained in 10 CFR 50.59 are not applicable to 10 CFR part 52 and are not being included in this rule. The NRC is including a definition for a ''departure from a method of evaluation'' (paragraph II.G), which is appropriate to include in this rulemaking so that the eight criteria in paragraph VIII.B.5.b will be implemented for new reactors as intended.

C. Scope and Contents (Section III)

The purpose of Section III is to describe and define the scope and contents of this design certification and to set forth how documentation discrepancies or inconsistencies are to be resolved. Paragraph III.A is the required statement of the OFR for approval of the incorporation by reference of Tier 1, Tier 2, and the generic TS in Revision 10 of the ESBWR DCD, as well as the 20 documents listed in Table 1 of paragraph III.A. Paragraph III.B requires COL applicants and licensees to comply with the requirements of this appendix. The legal effect of incorporation by reference is that the incorporated material has the same legal status as if it were published in the *Code of Federal Regulations.* This material, like any other properly-issued regulation, has the force and effect of law. Tier 1 and Tier 2 information, as well as the generic TS, have been combined into a single document called the generic DCD, in order to effectively control this information and facilitate its incorporation by reference into the rule. The generic DCD was prepared to meet the technical information contents of application requirements for design certifications under 10 CFR 52.47(a) and the requirements of the OFR for incorporation by reference under 1 CFR part 51. One of the requirements of the OFR for incorporation by reference is that the design certification applicant must make the documents incorporated by reference available upon request after the final rule becomes effective. Therefore, paragraph III.A identifies a GEH representative to be contacted in order to obtain a copy of the DCD and the 20 documents incorporated by reference into the ESBWR design certification rule.

Paragraphs III.A and III.B also identify the availability controls in Appendix 19ACM of the generic DCD as part of the Tier 2 information. During its review of the ESBWR design, the NRC determined that residual uncertainties associated with passive safety system performance increased the importance of nonsafety-related active systems in providing defense-in-depth functions that back-up the passive systems. As a result, GEH developed administrative controls to provide a high level of confidence that active systems having a significant safety role are available when challenged. GEH named these additional controls ''availability controls.'' The NRC included this characterization in Section III to ensure that these availability controls are binding on applicants and licensees that reference this appendix and will be enforceable by the NRC. The NRC's evaluation of the availability controls is provided in Chapter 22 of the FSER.

The generic DCD (master copy) and the 20 publicly-available documents listed in Table 1 of paragraph III.A are electronically accessible under the ADAMS Accession Nos. provided in paragraph III.A and at the OFR. Copies of these documents are also available at the NRC's PDR and from GEH as described in paragraph III.A. Questions concerning the accuracy of information in an application that references this appendix will be resolved by checking the master copy of the generic DCD or its referenced documents in ADAMS. If the design certification applicant makes a generic change (rulemaking) to the DCD under 10 CFR 52.63 and the change process provided in Section VIII, then at the completion of the rulemaking the NRC would request approval of the Director, OFR, for the revised master DCD. The NRC is requiring that the design certification applicant maintain an up-to-date copy of the master DCD that includes any generic changes it has made under paragraph X.A.1 because it is likely that most applicants intending to reference the standard design would obtain the generic DCD from the design certification applicant. Plant-specific changes to and departures from the generic DCD will be maintained by the applicant or licensee that references this appendix in a plant-specific DCD under paragraph X.A.2.

In addition to requiring compliance with this appendix, paragraph III.B clarifies that the conceptual design information and GEH's evaluation of SAMDAs are not considered to be part of this appendix. The conceptual design information is for those portions of the plant that are outside the scope of the standard design and are contained in Tier 2 information. As provided by 10 CFR 52.47(a)(24), these conceptual designs are not part of this appendix and, therefore, are not applicable to an application that references this appendix. Therefore, the applicant is not required to conform to the conceptual design information that was provided by the design certification applicant. The conceptual design information, which consists of site-specific design features, was required to facilitate the design certification review. Conceptual design information is neither Tier 1 nor Tier 2. Section 1.8.2 of Tier 2 identifies the location of the conceptual design information. GEH's evaluation of various design alternatives to prevent and mitigate severe accidents does not constitute design requirements. The NRC's assessment of this information is discussed in Section IX of this document.

Paragraphs III.C and III.D set forth the way potential conflicts are to be resolved. Paragraph III.C establishes the Tier 1 description in the DCD as controlling in the event of an inconsistency between the Tier 1 and Tier 2 information in the DCD. Paragraph III.D establishes the generic DCD as the controlling document in the event of an inconsistency between the DCD and the FSER (including Supplement No. 1) for the certified standard design.

Paragraph III.E makes it clear that design activities that are wholly outside the scope of this design certification may be performed using actual site characteristics, provided the design activities do not affect Tier 1 or Tier 2, or conflict with the interface requirements in the DCD. This provision applies to site-specific portions of the plant, such as the administration building. Because this statement is not a definition, this provision has been located in Section III.

D. Additional Requirements and Restrictions (Section IV)

Section IV sets forth additional requirements and restrictions imposed upon an applicant who references this appendix. Paragraph IV.A sets forth the information requirements for these applicants. This paragraph distinguishes between information and/or documents which must actually be *included* in the application or the DCD, versus those which may be *incorporated by reference* (i.e., referenced in the application as if the information or documents were included in the application). Any incorporation by reference in the application should be clear and should specify the title, date, edition, or version of a document, the page number(s), and table(s) containing the relevant information to be incorporated.

Paragraph IV.A.1 requires an applicant who references this appendix to incorporate by reference this appendix in its application. The legal effect of such an incorporation by reference into the application is that this appendix is legally binding on the applicant or licensee. Paragraph IV.A.2.a requires that a plant-specific DCD be included in the initial application to ensure that the applicant commits to complying with the DCD. This paragraph also requires the plant-specific DCD to either include or incorporate by reference the generic DCD information. Further, this paragraph also requires the plant-specific DCD to use the same format as the generic DCD and reflect the applicant's proposed exemptions and departures from the generic DCD as of the time of submission of the application. The plant-specific DCD will be part of the plant's FSAR, along with information for the portions of the plant outside the scope of the referenced design. Paragraph IV.A.2.a also requires that the initial application include the reports on departures and exemptions as of the time of submission of the application.

Paragraph IV.A.2.b requires that an application referencing this appendix include the reports required by paragraph X.B for exemptions and departures proposed by the applicant as of the date of submission of its application. Paragraph IV.A.2.c requires submission of plant-specific TS for the plant that consists of the generic TS from Chapter 16 of the DCD, with any changes made under paragraph VIII.C, and the TS for the site-specific portions of the plant that are either partially or wholly outside the scope of this design certification. The applicant must also provide the plant-specific information designated in the generic TS, such as bracketed values (refer to guidance provided in Interim Staff Guidance (ISG) DC/COL–ISG–8, ''Necessary Content of Plant-Specific Technical Specifications,'' ADAMS Accession No. ML083310259).

Paragraph IV.A.2.d requires the applicant referencing this appendix to provide information demonstrating that the proposed site characteristics fall within the site parameters for this appendix and that the plant-specific interface requirements have been met as required by 10 CFR 52.79(d). If the proposed site has a characteristic that does not fall within one or more of the site parameters in the DCD, then the proposed site is unacceptable for this design unless the applicant seeks an exemption under Section VIII and provides adequate justification for locating the certified design on the proposed site. Paragraph IV.A.2.e requires submission of information addressing COL action items, identified in the generic DCD as COL information in the application. The COL information identifies matters that need to be addressed by an applicant who references this appendix, as required by subpart C of 10 CFR part 52. An applicant may differ from or omit these items, provided that the difference or omission is identified and justified in its application. Based on the applicant's difference or omission, the NRC may impose additional licensing requirement(s) on the COL applicant as appropriate. Paragraph IV.A.2.f requires that the application include the information specified by 10 CFR 52.47(a) that is not within the scope of this rule, such as generic issues that must be addressed or operational issues not addressed by a design certification, in whole or in part, by an applicant that references this appendix. Paragraph IV.A.2.g requires that the application include information demonstrating that hurricane loads on those SSCs described in Section 3.3.2 of the generic DCD are either bounded by the total tornado loads analyzed in Section 3.3.2 of the generic DCD or will meet applicable NRC requirements with consideration of hurricane loads in excess of the total tornado loads. Paragraph IV.A.2.g further requires that hurricane-generated missile loads on those SSCs described in Section 3.5.2 of the generic DCD are either bounded by tornado-generated missile loads analyzed in Section 3.5.1.4 of the generic DCD or will meet applicable NRC requirements with consideration of hurricane-generated missile loads in excess of the tornado-generated missile loads. Paragraph IV.A.2.h requires that the application include information demonstrating that SFP level instrumentation is designed to allow the connection of an independent power source and that the instrumentation will maintain its design accuracy following a power interruption or change in power source without recalibration. Paragraph IV.A.3 requires the applicant to physically include, not simply reference, the SUNSI (including proprietary information and security-related information) and SGI referenced in the DCD, or its equivalent, to ensure that the applicant has actual notice of these requirements.

Paragraph IV.A.4 indicates requirements that must be met in cases where the COL applicant is not using the entity that was the original applicant for the design certification (or amendment) to supply the design for the applicant's use. Paragraph IV.A.4 requires that a COL applicant referencing this appendix include, as part of its application, a demonstration that an entity other than GEH Nuclear Energy is qualified to supply the ESBWR certified design unless GEH Nuclear Energy supplies the design for the applicant's use. This includes the non-public versions (or their equivalents) of the documents listed in Table 3 under section III.B of the **SUPPLEMENTARY INFORMATION** section of this document. In cases where a COL applicant is not using GEH Nuclear Energy to supply the ESBWR certified design, the required information would be used to support any NRC finding under 10 CFR 52.73(a) that an entity other than the one originally sponsoring the design certification or design certification amendment is qualified to supply the certified design.

Paragraph IV.B reserves to the Commission the right to determine in what manner this appendix may be referenced by an applicant for a construction permit or operating license under 10 CFR part 50. This determination may occur in the context of a subsequent rulemaking modifying 10 CFR part 52 or this DCR, or on a case-by-case basis in the context of a specific application for a 10 CFR part 50 construction permit or operating license. This provision is necessary because the previous DCRs were not implemented in the manner that was originally envisioned at the time that 10 CFR part 52 was promulgated. The NRC's concern is with the way ITAACs were developed and the lack of experience with design certifications in license proceedings. Therefore, it is appropriate that the Commission retain some discretion regarding the way this appendix could be referenced in a 10 CFR part 50 licensing proceeding.

E. Applicable Regulations (Section V)

The purpose of Section V is to specify the regulations that were applicable and in effect at the time this design certification was approved (i.e., as of the date specified in paragraph V.A, which would be the date that this appendix is approved by the Commission and signed by the Secretary of the Commission). These regulations consist of the technically relevant regulations identified in paragraph V.A, except for the regulations in paragraph V.B that are not applicable to this certified design.

In paragraph V.B, the NRC identifies the regulations that do not apply to the ESBWR design. The Commission has determined that the ESBWR design should be exempt from portions of 10 CFR 50.34 as described in the FSER (NUREG–1966) and/or summarized below:

Paragraph (f)(2)(iv) of 10 CFR 50.34—Contents of Construction Permit and Operating License Applications: Technical Information.

This paragraph requires an applicant to provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. The ESBWR design integrates the safety parameter display system into the design of the nonsafety-related distribution control and information system, rather than uses a stand-alone console. The safety parameter display system is described in Section 7.1.5 of the DCD.

The NRC has also determined that the ESBWR design is approved to use the following alternative. Under 10 CFR 50.55a(a)(3), GEH requested NRC approval for the use of ASME Code Case N–782 as a proposed alternative to the rules of Section III, Subsection NCA–1140, regarding applied Code Editions and Addenda required by 10 CFR 50.55a(c), (d), and (e). ASME Code Case N–782 provides that the Code Edition and Addenda endorsed in a certified design or licensed by the regulatory authority may be used for systems and components constructed to ASME Code, Section III requirements. These alternative requirements are in lieu of the requirements that base the Edition and Addenda on the construction permit date. Reference to ASME Code Case N–782 will be included in component and system design specifications and design reports to permit certification of these specifications and reports to the Code Edition and Addenda cited in the DCD. The NRC's bases for approving the use of ASME Code Case N–782 as a proposed alternative to the requirements of ASME Section III Subsection NCA–1140 under 10 CFR 50.55a(a)(3) for ESBWR are described in Section 5.2.1.1.3 of the FSER.

F. Issue Resolution (Section VI)

The purpose of Section VI is to identify the scope of issues that are resolved by the NRC in this rulemaking and, therefore, are ''matters resolved'' within the meaning and intent of 10 CFR 52.63(a)(5). The section is divided into five parts: Paragraph A identifies the NRC's safety findings in adopting this appendix, paragraph B identifies the scope and nature of issues which are resolved by this rulemaking, paragraph C identifies issues that are not resolved by this rulemaking, paragraph D identifies the backfit restrictions applicable to the Commission with respect to this appendix, and paragraph E identifies the availability of secondary references.

Paragraph VI.A describes the nature of the Commission's findings in general terms and makes the findings required by 10 CFR 52.54 for the Commission's approval of this DCR. Furthermore, paragraph VI.A explicitly states the Commission's determination that this design provides adequate protection of the public health and safety.

Paragraph VI.B sets forth the scope of issues that may not be challenged as a matter of right in subsequent proceedings. The introductory phrase of paragraph VI.B clarifies that issue resolution as described in the remainder of the paragraph extends to the delineated NRC proceedings referencing this appendix. The remainder of paragraph VI.B describes the categories of information for which there is issue resolution. Specifically, paragraph VI.B.1 provides that all nuclear safety issues arising from the Atomic Energy Act of 1954, as amended, that are associated with the information in the NRC staff's FSER (NUREG–1966 and Supplement No. 1), the Tier 1 and Tier 2 information (including the availability controls in Appendix 19ACM of the generic DCD), the 20 documents referenced in Table 1 of paragraph III.A, and the rulemaking record for this appendix are resolved within the meaning of 10 CFR 52.63(a)(5). These resolved issues include the information referenced in the DCD that are requirements (i.e., ''secondary references''), as well as all issues arising from SUNSI (including proprietary information and security-related information) and SGI that are intended to be requirements. However, paragraph VI.B.1 expressly excludes from issue resolution: The HFE procedure development and training program development identified in Sections 18.9 and 18.10 of the generic DCD; hurricane loads on those SSCs described in Section 3.3.2 of the generic DCD that are not bounded by the total tornado loads analyzed in Section 3.3.2 of the generic DCD; hurricane-generated missile loads on those SSCs described in Section 3.5.2 of the generic DCD that are not bounded by tornado-generated missile loads analyzed in Section 3.5.1.4 of the generic DCD; or that SFP level instrumentation is designed to allow the connection of an independent power source, and that the instrumentation will maintain its design accuracy following a power interruption or change in power source without recalibration.

Paragraph VI.B.2 provides for issue preclusion of SUNSI (including proprietary information and security-related information) and SGI, consisting of the fifty (50) non-publicly available documents listed in Tables 1.6–1 and 1.6–2 of Tier 2 of the ESBWR DCD, Revision 10.

Paragraphs VI.B.3, VI.B.4, VI.B.5, and VI.B.6 clarify that approved changes to and departures from the DCD, which are accomplished in compliance with the relevant procedures and criteria in Section VIII, continue to be matters resolved in connection with this rulemaking. Paragraphs VI.B.4, VI.B.5, and VI.B.6, which characterize the scope of issue resolution in three situations, use the phrase ''but only for that plant.'' Paragraph VI.B.4 describes how issues associated with a DCR are resolved when an exemption has been granted for a plant referencing the DCR. Paragraph VI.B.5 describes how issues are resolved when a plant referencing the DCR obtains a license amendment for a departure from Tier 2 information. Paragraph VI.B.6 describes how issues are resolved when the applicant or licensee departs from the Tier 2 information on the basis of paragraph VIII.B.5, which will waive the requirement for NRC approval. In all three situations, after a matter (e.g., an exemption in the case of paragraph VI.B.4) is addressed for a specific plant referencing a DCR, the adequacy of that matter *for that plant* is resolved and will constitute part of the licensing basis for that plant. Therefore, that matter will not ordinarily be subject to challenge in any subsequent proceeding or action for that plant (e.g., an enforcement action) listed in the introductory portion of paragraph IV.B. By contrast, there will be no legally binding issue resolution on that subject matter *for any other plant,* or in a subsequent rulemaking amending the applicable DCR. However, the NRC's consideration of the safety, regulatory or policy issues necessary to the determination of the exemption or license amendment may, in appropriate circumstances, be relied upon as part of the basis for NRC action in other licensing proceedings or rulemaking.

Paragraph VI.B.7 provides that, for those plants located on sites whose site characteristics fall within the site parameters assumed in the GEH evaluation of SAMDAs, all issues with respect to SAMDAs arising under the NEPA, associated with the information in the EA for this design and the information regarding SAMDAs in NEDO–33306, Revision 4, ''ESBWR Severe Accident Mitigation Design Alternatives'' are also resolved within the meaning and intent of 10 CFR 52.63(a)(5). If a deviation from a site parameter is granted, the deviation applicant has the initial burden of demonstrating that the original SAMDA analysis still applies to the actual site characteristics; however, if the deviation is approved, requests for litigation at the COL stage must meet the requirements of 10 CFR 2.309 and present sufficient information to create a genuine controversy in order to obtain a hearing on the site parameter deviation.

Paragraph VI.C reserves the right of the Commission to impose operational requirements on applicants that reference this appendix. This provision reflects the fact that only some operational requirements, including portions of the generic TS in Chapter 16 of the DCD, and no operational programs, such as operational quality assurance (QA), were completely or comprehensively reviewed by the NRC in this design certification rulemaking proceeding. Therefore, the special backfit and finality provisions of 10 CFR 52.63 apply only to those operational requirements that either the NRC completely reviewed and approved, or formed the basis for an NRC safety finding of the adequacy of the ESBWR, as documented in the NRC's FSER and Supplement No. 1 for the ESBWR. This is consistent with the currently approved design certifications in 10 CFR part 52, appendices A through D. Although information on operational matters is included in the DCDs of each of these currently approved designs, for the most part these design certifications do not provide approval for operational information, and none provide approval for operational ''programs'' (e.g., emergency preparedness programs, operational QA programs). Most operational information in the DCD simply serves as ''contextual information'' (i.e., information necessary to understand the design of certain SSCs and how they would be used in the overall context of the facility). The NRC did not use contextual information to support the NRC's safety conclusions and such information does not constitute the underlying safety bases for the adequacy of those SSCs. Thus, contextual operational information on any particular topic does not constitute one of the ''matters resolved'' under paragraph VI.B.

The NRC notes that operational requirements may be imposed on licensees referencing this design certification through the inclusion of license conditions in the license, or inclusion of a description of the operational requirement in the plant-specific FSAR.[5] The NRC's choice of the regulatory vehicle for imposing the operational requirements will depend upon, among other things: (1) Whether the development and/or implementation of these requirements must occur prior to either the issuance of the COL or the Commission finding under 10 CFR 52.103(g), and (2) the nature of the change controls that are appropriate given the regulatory, safety, and security significance of each operational requirement.

Paragraph VI.C allows the NRC to impose future operational requirements (distinct from design matters) on applicants who reference this design certification. Also, license conditions for portions of the plant within the scope of this design certification (e.g., start-up and power ascension testing) are not restricted by 10 CFR 52.63. The requirement to perform these testing programs is contained in Tier 1 information. However, ITAACs cannot be specified for these subjects because the matters to be addressed in these license conditions cannot be verified prior to fuel load and operation, when the ITAACs are satisfied. Therefore, another regulatory vehicle is necessary to ensure that licensees comply with the matters contained in the license conditions. License conditions for these areas cannot be developed now because this requires the type of detailed design information that will be developed during a COL review. In the absence of detailed design information to evaluate the need for and develop specific post-fuel load verifications for these matters, the Commission is reserving in this rule the right to impose, at the time of COL issuance, license conditions addressing post-fuel load verification activities for portions of the plant within the scope of this design certification.

Paragraph VI.D reiterates the restrictions (contained in Section VIII) placed upon the Commission when ordering generic or plant-specific modifications, changes or additions to SSCs, design features, design criteria, and ITAACs (paragraph VI.D.3 addresses ITAACs) within the scope of the certified design.

[5] Certain activities, ordinarily conducted following fuel load and therefore considered ''operational requirements,'' but which may be relied upon to support a Commission finding under 10 CFR 52.103(g), may themselves be the subject of ITAAC to ensure their implementation prior to the 10 CFR 52.103(g) finding.

Paragraph VI.E provides that the NRC will specify at an appropriate time the procedures for interested persons to obtain access to SUNSI (including proprietary information and security-related information) and SGI information for the ESBWR DCR. Access to such information would be for the sole purpose of requesting or participating in certain specified hearings, such as: (1) The hearing required by 10 CFR 52.85 where the underlying application references this appendix; (2) any hearing provided under 10 CFR 52.103 where the underlying COL references this appendix; and (3) any other hearing relating to this appendix in which interested persons have the right to request an adjudicatory hearing.

For proceedings where the notice of hearing was published before the effective date of the final rule, the Commission's order governing access to SUNSI and SGI shall be used to govern access to such information within the scope of the rulemaking. For proceedings in which the notice of hearing or opportunity for hearing is published after the effective date of the final rule, paragraph VI.E applies and governs access to SUNSI and SGI. For these proceedings, as stated in paragraph VI.E, the NRC will specify the access procedures at an appropriate time.

For both a hearing required by 10 CFR 52.85 where the underlying application references this appendix, and in any hearing on ITAACs completion under 10 CFR 52.103, the NRC expects to follow its current practice of establishing the procedures by order at the time that the notice of hearing is published in the **Federal Register**. See, for example, Florida Power and Light Co., Combined License Application for the Turkey Point Units 6 & 7, Notice of Hearing, Opportunity To Petition for Leave To Intervene and Associated Order Imposing Procedures for Access to SUNSI and Safeguards Information for Contention Preparation (75 FR 34777; June 18, 2010); Notice of Receipt of Application for License; Notice of Consideration of Issuance of License; Notice of Hearing and Commission Order and Order Imposing Procedures for Access to SUNSI and Safeguards Information for Contention Preparation; In the Matter of AREVA Enrichment Services, LLC (Eagle Rock Enrichment Facility) (74 FR 38052; July 30, 2009).

G. Duration of This Appendix (Section VII)

The purpose of Section VII is, in part, to specify the period during which this design certification may be referenced by an applicant for a COL, under 10 CFR 52.55. This section also states that the design certification remains valid for an applicant or licensee that references the design certification until the application is withdrawn or the license expires. Therefore, if an application references this design certification during the 15-year period, then the design certification will be effective until the application is withdrawn or the license issued on that application expires. Also, the design certification will be effective for the referencing licensee if the license is renewed. The NRC intends this appendix to remain valid for the life of the plant that references the design certification to achieve the benefits of standardization and licensing stability. This means that changes to, or plant-specific departures from, information in the plant-specific DCD must be made under the change processes in Section VIII for the life of the plant.

H. Processes for Changes and Departures (Section VIII)

The purpose of Section VIII is to set forth the processes for generic changes to, or plant-specific departures (including exemptions) from, the DCD. The Commission adopted this restrictive change process in order to achieve a more stable licensing process for applicants and licensees that reference DCRs. Section VIII is divided into three paragraphs, which correspond to Tier 1, Tier 2, and operational requirements. The language of Section VIII distinguishes between generic *changes to* the DCD versus plant-specific *departures from* the DCD. Generic *changes* must be accomplished by rulemaking because the intended subject of the change is this DCR itself, as is contemplated by 10 CFR 52.63(a)(1). Consistent with 10 CFR 52.63(a)(3), any generic rulemaking changes are applicable to all plants, absent circumstances which render the change [''modification'' in the language of 10 CFR 52.63(a)(3)] ''technically irrelevant.'' By contrast, plant-specific *departures* could be either a Commission-issued order to one or more applicants or licensees; or an applicant or licensee-initiated departure applicable only to that applicant's or licensee's plant(s), similar to a 10 CFR 50.59 departure or an exemption. Because these plant-specific departures will result in a DCD that is unique for that plant, Section X requires an applicant or licensee to maintain a plant-specific DCD. For purposes of brevity, the following discussion refers to both generic changes and plant-specific departures as ''change processes.''

Section VIII refers to an exemption from one or more requirements of this appendix and the criteria for granting an exemption. The NRC cautions that when the exemption involves an underlying substantive requirement (applicable regulation), then the applicant or licensee requesting the exemption must also show that an exemption from the underlying applicable requirement meets the criteria of 10 CFR 52.7.

Tier 1 Information

The change processes for Tier 1 information are covered in paragraph VIII.A. Generic changes to Tier 1 are accomplished by rulemakings that amend the generic DCD and are governed by the standards in 10 CFR 52.63(a)(1) and 10 CFR 52.63(a)(2). No matter who proposes it, a generic change under 10 CFR 52.63(a)(1) will not be made to a certified design while it is in effect unless the change: (1) Is necessary for compliance with Commission regulations applicable and in effect at the time the certification was issued; (2) is necessary to provide adequate protection of the public health and safety or common defense and security; (3) reduces unnecessary regulatory burden and maintains protection to public health and safety and common defense and security; (4) provides the detailed design information necessary to resolve selected design acceptance criteria; (5) corrects material errors in the certification information; (6) substantially increases overall safety, reliability, or security of a facility and the costs of the change are justified; or (7) contributes to increased standardization of the certification information. The rulemakings must provide for notice and opportunity for public comment on the proposed change, as required by 10 CFR 52.63(a)(2). The Commission will give consideration to whether the benefits justify the costs for plants that are already licensed or for which an application for a permit or license is under consideration.

Departures from Tier 1 may occur in two ways: (1) The Commission may *order* a licensee to depart from Tier 1, as provided in paragraph VIII.A.3; or (2) an applicant or licensee may request an *exemption* from Tier 1, as provided in paragraph VIII.A.4. If the Commission seeks to order a licensee to depart from Tier 1, paragraph VIII.A.3 requires that the Commission find both that the departure is necessary for adequate protection or for compliance and that special circumstances are present. Paragraph VIII.A.4 provides that exemptions from Tier 1 requested by an applicant or licensee are governed by the requirements of 10 CFR 52.63(b)(1) and 52.98(f), which provide an opportunity for a hearing. In addition, the Commission will not grant requests for exemptions that may result in a significant decrease in the level of safety otherwise provided by the design.

Tier 2 Information

The change processes for the three different categories of Tier 2 information, namely, Tier 2, Tier 2*, and Tier 2* with a time of expiration, are set forth in paragraph VIII.B. The change process for Tier 2 has the same elements as the Tier 1 change process, but some of the standards for plant-specific orders and exemptions are different.

The process for generic Tier 2 changes (including changes to Tier 2* and Tier 2* with a time of expiration) tracks the process for generic Tier 1 changes. As set forth in paragraph VIII.B.1, generic Tier 2 changes are accomplished by rulemaking amending the generic DCD and are governed by the standards in 10 CFR 52.63(a)(1). No matter who proposes it, a generic change under 10 CFR 52.63(a)(1) will not be made to a certified design while it is in effect unless the change: (1) Is necessary for compliance with NRC regulations applicable and in effect at the time the certification was issued; (2) is necessary to provide adequate protection of the public health and safety or common defense and security; (3) reduces unnecessary regulatory burden and maintains protection to public health and safety and common defense and security; (4) provides the detailed design information necessary to resolve selected design acceptance criteria; (5) corrects material errors in the certification information; (6) substantially increases overall safety, reliability, or security of a facility and the costs of the change are justified; or (7) contributes to increased standardization of the certification information. If a generic change is made to Tier 2* information, then the category and expiration, if necessary, of the new information will also be determined in the rulemaking and the appropriate change process for that new information would apply.

Departures from Tier 2 may occur in five ways: (1) The Commission may order a plant-specific departure, as set forth in paragraph VIII.B.3; (2) an applicant or licensee may request an exemption from a Tier 2 requirement as set forth in paragraph VIII.B.4; (3) a licensee may make a departure without prior NRC approval under paragraph VIII.B.5; (4) the licensee may request NRC approval for proposed departures which do not meet the requirements in paragraph VIII.B.5 as provided in paragraph VIII.B.5.d; and (5) the licensee may request NRC approval for a departure from Tier 2* information under paragraph VIII.B.6.

Similar to Commission-ordered Tier 1 departures and generic Tier 2 changes, Commission-ordered Tier 2 departures cannot be imposed except when necessary either to bring the certification into compliance with the NRC's regulations applicable and in effect at the time of approval of the design certification or to ensure adequate protection of the public health and safety or common defense and security, as set forth in paragraph VIII.B.3. However, the special circumstances for the Commission-ordered Tier 2 departures do not have to outweigh any decrease in safety that may result from the reduction in standardization caused by the plant-specific order, as required by 10 CFR 52.63(a)(4). The Commission determined that it was not necessary to impose an additional limitation similar to that imposed on Tier 1 departures by 10 CFR 52.63(a)(4) and (b)(1). This type of additional limitation for standardization would unnecessarily restrict the flexibility of applicants and licensees with respect to Tier 2 information.

An applicant or licensee may request an exemption from Tier 2 information as set forth in paragraph VIII.B.4. The applicant or licensee must demonstrate that the exemption complies with one of the special circumstances in 10 CFR 50.12(a). In addition, the Commission will not grant requests for exemptions that may result in a significant decrease in the level of safety otherwise provided by the design. However, the special circumstances for the exemption do not have to outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. If the exemption is requested by an applicant for a license, the exemption is subject to litigation in the same manner as other issues in the license hearing, consistent with 10 CFR 52.63(b)(1). If the exemption is requested by a licensee, then the exemption is subject to litigation in the same manner as a license amendment.

Paragraph VIII.B.5 allows an applicant or licensee to depart from Tier 2 information, without prior NRC approval, if the proposed departure does not involve a change to, or departure from, Tier 1 or Tier 2* information, TS, or does not require a license amendment under paragraphs VIII.B.5.b or VIII.B.5.c. The TS referred to in

VIII.B.5.a of this paragraph are the TS in Chapter 16 of the generic DCD, including bases, for departures made prior to issuance of the COL. After issuance of the COL, the plant-specific TS are controlling under paragraph VIII.B.5. The bases for the plant-specific TS will be controlled by the bases control program, which is specified in the plant-specific TS administrative controls section. The requirement for a license amendment in paragraph VIII.B.5.b will be similar to the requirement in 10 CFR 50.59 and apply to all information in Tier 2 except for the information that resolves the severe accident issues.

The NRC concludes that the resolution of ex-vessel severe accident design features should be preserved and maintained in the same fashion as all other safety issues that were resolved during the design certification review (refer to SRM on SECY–90–377, ''Requirements for Design Certification Under 10 CFR Part 52,'' dated February 15, 1991, ADAMS Accession No. ML003707892). However, because of the increased uncertainty in ex-vessel severe accident issue resolutions, the NRC has adopted separate criteria in paragraph VIII.B.5.c for determining if a departure from information that resolves ex-vessel severe accident design features would require a license amendment. For purposes of applying the special criteria in paragraph VIII.B.5.c, ex-vessel severe accident resolutions are limited to design features where the intended function of the design feature is relied upon to resolve postulated accidents when the reactor core has melted and exited the reactor vessel, and the containment is being challenged. These design features are identified in Sections 19.2.3, 19.3.2, 19.3.3, 19.3.4, and Appendices 19A and 19B of the DCD, with other issues, and are described in other sections of the DCD. Therefore, the location of design information in the DCD is not important to the application of this special procedure for ex-vessel severe accident design features. However, the special procedure in paragraph VIII.B.5.c does not apply to design features that resolve so-called ''beyond design-basis accidents'' or other low-probability events. The important aspect of this special procedure is that it is limited to ex-vessel severe accident design features, as defined above. Some design features may have intended functions to meet ''design basis'' requirements and to resolve ''severe accidents.'' If these design features are reviewed under paragraph VIII.B.5, then the appropriate criteria from either paragraphs VIII.B.5.b or VIII.B.5.c are selected depending upon the function being changed.

An applicant or licensee that plans to depart from Tier 2 information, under paragraph VIII.B.5, is required to prepare an evaluation that provides the bases for the determination that the proposed change does not require a license amendment or involve a change to Tier 1 or Tier 2* information, or a change to the TS, as explained above. In order to achieve the NRC's goals for design certification, the evaluation needs to consider all of the matters that were resolved in the DCD, such as generic issue resolutions that are relevant to the proposed departure. The benefits of the early resolution of safety issues would be lost if departures from the DCD were made that violated these resolutions without appropriate review.

The evaluation of the relevant matters needs to consider the proposed departure over the full range of power operation from startup to shutdown, as it relates to anticipated operational occurrences, transients, DBAs, and severe accidents. The evaluation must also include a review of all relevant secondary references from the DCD because Tier 2 information, which is intended to be treated as a requirement, is contained in the secondary references. The evaluation should consider Tables 14.3–1a through 14.3–1c and 19.2–3 of the generic DCD to ensure that the proposed change does not impact Tier 1 information. These tables contain cross-references from the safety analyses and probabilistic risk assessment (PRA) in Tier 2 to the important parameters that were included in Tier 1.

Paragraph VIII.B.5.d addresses information described in the DCD to address aircraft impacts, in accordance with 10 CFR 52.47(a)(28). Under 10 CFR 52.47(a)(28), applicants are required to include the information required by 10 CFR 50.150(b) in their DCD. Under 10 CFR 50.150(b), applications for standard design certifications are required to include:

1. A description of the design features and functional capabilities identified as a result of the AIA required by 10 CFR 50.150(a)(1); and

2. A description of how such design features and functional capabilities meet the assessment requirements in 10 CFR 50.150(a)(1).

An applicant or licensee who changes this information is required to consider the effect of the changed design feature or functional capability on the original AIA required by 10 CFR 50.150(a). The applicant or licensee is also required to describe in the plant-specific DCD how the modified design features and functional capabilities continue to meet the assessment requirements in 10 CFR 50.150(a)(1). Submittal of this updated information is governed by the reporting requirements in Section X.B.

In an adjudicatory proceeding (e.g., for issuance of a COL), a person who believes that an applicant or licensee has not complied with paragraph VIII.B.5 when departing from Tier 2 information is permitted to petition to admit such a contention into the proceeding under paragraph VIII.B.5.f. This provision was included because an incorrect departure from the requirements of this appendix essentially places the departure outside of the scope of the Commission's safety finding in the design certification rulemaking. Therefore, it follows that properly founded contentions alleging such incorrectly implemented departures cannot be considered ''resolved'' by this rulemaking. As set forth in paragraph VIII.B.5.f, the petition must comply with the requirements of 10 CFR 2.309 and show that the departure does not comply with paragraph VIII.B.5. Other persons may file a response to the petition under 10 CFR 2.309. If, on the basis of the petition and any responses, the presiding officer in the proceeding determines that the required showing has been made, the matter shall be certified to the Commission for its final determination. In the absence of a proceeding, petitions alleging nonconformance with paragraph VIII.B.5 requirements applicable to Tier 2 departures will be treated as petitions for enforcement action under 10 CFR 2.206.

Paragraph VIII.B.6 provides a process for departing from Tier 2* information. The creation of and restrictions on changing Tier 2* information resulted from the development of the Tier 1 information for the Advanced Boiling Water Reactor design certification (appendix A to 10 CFR part 52) and the System 80+ design certification (appendix B to 10 CFR part 52). During this development process, these applicants requested that the amount of information in Tier 1 be minimized to provide additional flexibility for an applicant or licensee who references these appendices. Also, many codes, standards, and design processes that were not specified in Tier 1 as acceptable for meeting ITAACs were specified in Tier 2. The result of these departures is that certain significant information exists only in Tier 2 and the Commission does not want this significant information to be changed without prior NRC approval. This Tier 2* information is identified in the generic DCD with italicized text and brackets (see Table 1D–1 in Appendix 1D of the ESBWR DCD).

Although the Tier 2* designation was originally intended to last for the lifetime of the facility, like Tier 1 information, the NRC determined that some of the Tier 2* information could expire when the plant first achieves full (100 percent) power, after the finding required by 10 CFR 52.103(g), while other Tier 2* information must remain in effect throughout the life of the facility. The factors determining whether Tier 2* information could expire after full power is first achieved (first full power) were whether the Tier 1 information would govern these areas after first full power and the NRC's determination that prior approval was required before implementation of the change due to the significance of the information. Therefore, certain Tier 2* information listed in paragraph VIII.B.6.c ceases to retain its Tier 2* designation after full power operation is first achieved following the Commission finding under 10 CFR 52.103(g). Thereafter, that information is deemed to be Tier 2 information that is subject to the departure requirements in paragraph VIII.B.5. By contrast, the Tier 2* information identified in paragraph VIII.B.6.b retains its Tier 2* designation throughout the duration of the license, including any period of license renewal.

Certain preoperational tests in paragraph VIII.B.6.c are designated to be performed only for the first plant that references this appendix. GEH's basis for performing these ''first-plant-only'' preoperational tests is provided in Section 14.2.8 of the DCD. The NRC found GEH's basis for performing these tests and its justification for only performing the tests on the first plant acceptable. The NRC's decision was based on the need to verify that plant-specific manufacturing and/or construction variations do not adversely impact the predicted performance of certain passive safety systems, while recognizing that these special tests will result in significant thermal transients being applied to critical plant components. The NRC concludes that the range of manufacturing or construction variations that could adversely affect the relevant passive safety systems would be adequately disclosed after performing the designated tests on the first plant. The Tier 2* designation for these tests will expire after the first plant completes these tests, as indicated in paragraph VIII.B.6.c.

If Tier 2* information is changed in a generic rulemaking, the designation of the new information (Tier 1, 2*, or 2) will also be determined in the rulemaking and the appropriate process for future changes will apply. If a plant-specific departure is made from Tier 2* information, then the new designation will apply only to that plant. If an applicant who references this design certification makes a departure from Tier 2* information, the new information will be subject to litigation in the same manner as other plant-specific issues in the licensing hearing. If a licensee makes a departure from Tier 2* information, it will be treated as a license amendment under 10 CFR 50.90 and the finality will be determined under paragraph VI.B.5. Any requests for departures from Tier 2* information that affects Tier 1 must also comply with the requirements in paragraph VIII.A.

Operational Requirements

The change process for TS and other operational requirements in the DCD is set forth in paragraph VIII.C. This change process has elements similar to the Tier 1 and Tier 2 change processes in paragraphs VIII.A and VIII.B, but with significantly different change standards. Because of the different finality status for TS and other operational requirements (refer to paragraph V.F of this document), the Commission designated a special category of information, consisting of the TS and other operational requirements, with its own change process in proposed paragraph VIII.C. The key to using the change processes proposed in Section VIII is to determine if the proposed change or departure requires a change to a design feature described in the generic DCD. If a design change is required, then the appropriate change process in paragraph VIII.A or VIII.B applies. However, if a proposed change to the TS or other operational requirements does not require a change to a design feature in the generic DCD, then paragraph VIII.C applies. The language in paragraph VIII.C also distinguishes between generic (Chapter 16 of the DCD) and plant-specific TS to account for the different treatment and finality accorded TS before and after a license is issued.

The process in paragraph VIII.C.1 for making generic changes to the generic TS in Chapter 16 of the DCD or other operational requirements in the generic DCD is accomplished by rulemaking and governed by the backfit standards in 10 CFR 50.109. The determination of whether the generic TS and other operational requirements were completely reviewed and approved in the design certification rulemaking is based upon the extent to which the NRC reached a safety conclusion in the FSER on this matter. If it cannot be determined, in the absence of a specific statement, that the TS or operational requirement was comprehensively reviewed and finalized in the design certification rulemaking, then there is no backfit restriction under 10 CFR 50.109 because no prior position, consistent with paragraph VI.B, was taken on this safety matter. Generic changes made under paragraph VIII.C.1 are applicable to all applicants or licensees (refer to paragraph VIII.C.2), unless the change is irrelevant because of a plant-specific departure.

Some generic TS and availability controls contain values in brackets [ ]. The brackets are placeholders indicating that the NRC's review is not complete and represent a requirement that the applicant for a COL referencing the ESBWR DCR must replace the values in brackets with final plant-specific values (refer to guidance provided in Interim Staff Guidance DC/COL–ISG–8, ''Necessary Content of Plant-Specific Technical Specifications''). The values in brackets are neither part of the DCR nor are they binding. Therefore, the replacement of bracketed values with final plant-specific values does not require an exemption from the generic TS or availability controls.

Plant-specific departures may occur by either a Commission order under paragraph VIII.C.3 or an applicant's exemption request under paragraph VIII.C.4. The basis for determining if the TS or operational requirement was completely reviewed and approved for these processes is the same as for paragraph VIII.C.1 above. If the TS or operational requirement is comprehensively reviewed and finalized in the design certification rulemaking, then the Commission must demonstrate that special circumstances are present before ordering a plant-specific departure. If not, there is no restriction on plant-specific changes to the TS or operational requirements, prior to the issuance of a license, provided a design change is not required. Although the generic TS were reviewed and approved by the NRC staff in support of the design certification review, the Commission intends to consider the lessons learned from subsequent operating experience during its licensing review of the plant-specific TS. The process for petitioning to intervene on a TS or operational requirement contained in paragraph VIII.C.5 is similar to other issues in a licensing hearing, except that the petitioner must also demonstrate why special circumstances are present pursuant to 10 CFR 2.335.

Finally, the generic TS will have no further effect on the plant-specific TS after the issuance of a license that references this appendix. The bases for the generic TS will be controlled by the change process in paragraph VIII.C. After a license is issued, the bases will be controlled by the bases change provision set forth in the administrative controls section of the plant-specific TS.

I. [RESERVED] (Section IX)

This section is reserved for future use. As discussed in Section IV of the **SUPPLEMENTARY INFORMATION** section of this document, the matters discussed in this section of earlier design certification rules—inspections, tests, analyses, and acceptance criteria—are now addressed in the substantive provisions of 10 CFR part 52. Accordingly, there is no need to repeat these regulatory provisions in the ESBWR design certification rule.

J. Records and Reporting (Section X)

The purpose of Section X is to set forth the requirements that will apply to maintaining records of changes to and departures from the generic DCD, which are to be reflected in the plant-specific DCD. Section X also sets forth the requirements for submitting reports (including updates to the plant-specific DCD) to the NRC. This section of the appendix is similar to the requirements for records and reports in 10 CFR part 50, except for minor differences in information collection and reporting requirements.

Paragraph X.A.1 requires that a generic DCD and the SUNSI (including proprietary information and security-related information) and SGI referenced in the generic DCD be maintained by the applicant for this rule. The generic DCD concept was developed, in part, to meet the OFR requirements for incorporation by reference, including public availability of documents incorporated by reference. However, the SUNSI (including proprietary information and security-related information) and SGI could not be included in the generic DCD because they are not publicly available. Nonetheless, the SUNSI (including proprietary information and security-related information) and SGI was reviewed by the NRC and, as stated in paragraph VI.B.2, the NRC considers the information to be resolved within the meaning of 10 CFR 52.63(a)(5). Because this information is not in the generic DCD, this information, or its equivalent, is required to be provided by an applicant for a license referencing this DCR. Paragraph X.A.1 requires the design certification applicant to maintain the SUNSI (including proprietary information and security-related information) and SGI, which it developed and used to support its design certification application. This ensures that the referencing applicant has direct access to this information from the design certification applicant, if it has contracted with the applicant to provide the SUNSI (including proprietary information and security-related information) and SGI to support its license application. The NRC may also inspect this information if it was not submitted to the NRC (e.g., the AIA required by 10 CFR 50.150). Only the generic DCD and 20 publicly-available documents referenced in the DCD are identified and incorporated by reference into this rule. The generic DCD and the NRC-approved version of the SUNSI (including proprietary information and security-related information) and SGI must be maintained by the applicant (GEH) for the period of time that this appendix may be referenced.

Paragraphs X.A.2 and X.A.3 place recordkeeping requirements on the applicant or licensee who references this design certification so that its plant-specific DCD accurately reflects both generic changes to the generic DCD and plant-specific departures made under Section VIII. The term ''plant-specific'' is used in paragraph X.A.2 and other sections of this appendix to distinguish between the generic DCD that is incorporated by reference into this appendix and the plant-specific DCD that the applicant is required to submit under paragraph IV.A. The requirement to maintain changes to the generic DCD is explicitly stated to ensure that these changes are not only reflected in the generic DCD, which will be maintained by the applicant for design certification, but also in the plant-specific DCD. Therefore, records of generic changes to the DCD will be required to be maintained by both entities to ensure that both entities have up-to-date DCDs.

Paragraph X.A.4.a requires the applicant to maintain a copy of the AIA performed to comply with the requirements of 10 CFR 50.150(a) for the term of the certification (including any period of renewal). This provision, which is consistent with 10 CFR 50.150(c)(3), will facilitate any NRC inspections of the assessment that the NRC decides to conduct. Similarly, paragraph X.A.4.b requires an applicant or licensee who references this appendix to maintain a copy of the AIA performed to comply with the requirements of 10 CFR 50.150(a) throughout the pendency of the application and for the term of the license (including any period of renewal). This provision is consistent with 10 CFR 50.150(c)(4). For all applicants and licensees, the supporting documentation retained onsite should describe the methodology used in performing the assessment, including the identification of potential design features and functional capabilities to show that the acceptance criteria in 10 CFR 50.150(a)(1) will be met.

Paragraph X.A does not place recordkeeping requirements on site-specific information that is outside the scope of this rule. As discussed in paragraph V.D of this document, the FSAR required by 10 CFR 52.79 will contain the plant-specific DCD and the site-specific information for a facility that references this rule. The phrase ''site-specific portion of the final safety analysis report'' in paragraph X.B.3.c refers to the information that is contained in the FSAR for a facility (required by 10 CFR 52.79) but is not part of the plant-specific DCD (required by paragraph IV.A). Therefore, this rule does not require that duplicate documentation be maintained by an applicant or licensee that references this rule because the plant-specific DCD is part of the FSAR for the facility.

Paragraph X.B.1 requires applicants or licensees that reference this rule to submit reports, which describe departures from the DCD and include a summary of the written evaluations. The requirement for the written evaluations is set forth in paragraph X.A.1. The frequency of the report submittals is set forth in paragraph X.B.3. The requirement for submitting a summary of the evaluations is similar to the requirement in 10 CFR 50.59(d)(2).

Paragraph X.B.2 requires applicants or licensees that reference this rule to submit updates to the DCD, which include both generic changes and plant-specific departures. The frequency for submitting updates is set forth in paragraph X.B.3. The requirements in paragraph X.B.3 for submitting the reports and updates will vary according to certain time periods during a facility's lifetime. If a potential applicant for a COL who references this rule decides to depart from the generic DCD prior to submission of the application, then paragraph X.B.3.a will require that the updated DCD be submitted as part of the initial application for a license. Under paragraph X.B.3.b, the applicant may submit any subsequent updates to its plant-specific DCD along with its amendments to the application provided that the submittals are made at least once per year. Because amendments to an application are typically made more frequently than once a year, this should not be an excessive burden on the applicant.

Paragraph X.B.3.b also requires semi-annual submission of the reports required by paragraph X.B.1 throughout the period of application review and construction. The NRC will use the information in the reports to help plan the NRC's inspection and oversight during this phase when the licensee is conducting detailed design, procurement of components and equipment, construction, and preoperational testing. In addition, the NRC will use the information in making its finding on ITAACs under 10 CFR 52.103(g), as well as any finding on interim operation under Section 189.a(1)(B)(iii) of the AEA. Once a facility begins operation (for a COL under 10 CFR part 52, after the Commission has made a finding under 10 CFR 52.103(g)), the frequency of reporting will be governed by the requirements in paragraph X.B.3.c.

VIII. Agreement State Compatibility

Under the ''Policy Statement on Adequacy and Compatibility of Agreement States Programs,'' approved by the Commission on June 20, 1997, and published in the **Federal Register** (62 FR 46517; September 3, 1997), this rule is classified as compatibility ''NRC.'' Compatibility is not required for Category ''NRC'' regulations. The NRC program elements in this category are those that relate directly to areas of regulation reserved to the NRC by the AEA or the provisions of Title 10 of the *Code of Federal Regulations,* and although an Agreement State may not adopt program elements reserved to the NRC, it may wish to inform its licensees of certain requirements by a mechanism that is consistent with a particular State's administrative procedure laws, but does not confer regulatory authority on the State.

IX. Availability of Documents

The documents identified in the following table are available to interested persons through one or more of the following methods, as indicated.

Document	ADAMS Accession No./ web link/ **Federal Register** citation
Proposed Rule Documents:	
SECY–11–0006, ''Proposed Rule—ESBWR Design Certification''	ML102220172
Staff Requirements Memorandum for SECY–11–0006, ''Proposed Rule—ESBWR Design Certification''	ML110670047
General Electric Company Application for Final Design Approval and Design Certification of ESBWR Standard Plant Design.	ML052450245
ESBWR Design Control Document, Revision 9	ML103440266
ESBWR Final Safety Evaluation Report (NUREG–1966)	ML14100A304
ESBWR FSER Final Chapters	ML103470210
Final Design Approval for the Economic Simplified Boiling Water Reactor	ML110540310
ESBWR Draft Environmental Assessment	ML102220247
ESBWR Proposed Rule **Federal Register** Notice, 76 FR 16549, March 24, 2011	ML110610353
Public Comments on the March 2011 Proposed Rule:	
Comment (1) from Farouk D. Baxter on Environmental Impact Statement for Two AP1000 Units at Levy County Site	ML102350160
Comment submission S1 from Paul C. Daugherty	ML110880057
Comment submission S2 from Farouk D. Baxter	ML110880315
Comment submission S3 from Patricia T. Birnie, Chair, General Electric Stockholders' Alliance	ML11158A088
Comment submission S4 from anonymous	ML11187A303
Comment submission P1, Emergency Petition To Suspend All Pending Reactor Licensing Decisions and Related Rulemaking Decisions Pending Investigation of Lessons Learned From Fukushima Daiichi Nuclear Power Station Accident (initial).	ML111040472
Comment submission P2, Emergency Petition To Suspend All Pending Reactor Licensing Decisions and Related Rulemaking Decisions Pending Investigation of Lessons Learned From Fukushima Daiichi Nuclear Power Station Accident (amended).	ML111080855
Comment submission P3, Declaration of Dr. Arjun Makhijani in Support of Emergency Petition To Suspend All Pending Reactor Licensing Decisions and Relating Rulemaking Decisions Pending Investigation of Lessons Learned From Fukushima Daiichi Nuclear Power Station Accident.	ML111100618
Comment submission P4, Comment of Jerald Head on Behalf of GE–Hitachi Nuclear Energy Opposing Petition To Suspend All Pending Reactor Licensing Decisions and Related Rulemaking Decisions Pending Investigation of Lessons Learned From Fukushima Daiichi Nuclear Power Station Accident.	ML11124A103
Comment submission P5, Petitioners' Reply to Responses to Emergency Petition To Suspend All Pending Reactor Licensing Decisions and Related Rulemaking Decisions Pending Investigation of Lessons Learned From Fukushima Daiichi Nuclear Power Station Accident.	ML111260637
Comment submission P6, Comments of Terry J. Lodge on PR 52, NEPA Requirement To Address Safety and Environmental Implications of the Fukushima Task Force Report From ESBWR, Fermi 3 Intervenors.	ML112430118
Public Comments Compilation—Final Rule—ESBWR Design Certification (RIN 3150–AI85)	ML113130141
Supplemental Safety Evaluation for the ESBWR Design Certification:	
Advanced Supplemental Safety Evaluation Report for the Economic Simplified Boiling-Water Reactor Standard Plant Design.	ML14043A134
Supplemental Safety Evaluation Report for the Economic Simplified Boiling-Water Reactor Standard Plant Design	ML14155A333
Supplemental Proposed Rule Documents:	
ESBWR Design Control Document, Rev. 10	ML14104A929
ESBWR Supplemental Proposed Rule **Federal Register** Notice, 79 FR 25715, May 6, 2014	ML14043A508
Final Rule Documents:	
SECY–14–0081, ''Final Rule—ESBWR Design Certification''	ML111730346
Staff Requirements Memorandum for SECY–14–0081, ''Final Rule—ESBWR Design Certification''	ML14259A545
ESBWR Final Environmental Assessment	ML111730382
Other Documents Relevant to the ESBWR Rulemaking:	
NEDO–33306, Revision 4, ''ESBWR Severe Accident Mitigation Design Alternatives''	ML102990433
NEDO–33312, Rev. 5, ''ESBWR Steam Dryer Acoustic Load Definition''	ML13344B157

Document	ADAMS Accession No./ web link/ **Federal Register** citation
NEDO–33313, Rev. 5, ''ESBWR Steam Dryer Structural Evaluation''	ML13344B158
NEDO–33338, Revision 1, ''ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis''	ML091380173
NEDO–33408P, Revision 5, ''ESBWR Steam Dryer—Plant-Based Load Evaluation Methodology, PBLE01 Model Description''.	ML13344B159
Commission Memorandum and Order (CLI–11–05), September 9, 2011 (available on the NRC Web site in Volume 74 at *http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0750/*).	ML112521106
Commission Order, ''Scheduling Order of the Secretary Regarding Petitions To Suspend Adjudicatory, Licensing and Rulemaking Activities (PR 52 re ESBWR Design Certification)''.	ML111101277
Order EA–12–049, ''Order Modifying Licenses With Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events''.	ML12054A735
Order EA 12–051, ''Order Modifying Licenses With Regard to Reliable Spent Fuel Pool Instrumentation''	ML12054A679
Staff Requirements Memorandum for SECY–90–377, ''Requirements for Design Certification Under 10 CFR Part 52''	ML003707892
SECY–94–084, ''Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs''.	ML003708068
Staff Requirements Memorandum for SECY–96–077, ''Certification of Two Evolutionary Designs''	ML003754873
SECY–96–077, ''Certification of Two Evolutionary Designs''	ML003708129
Staff Requirements Memorandum for SECY–11–0093, ''Near-Team Report and Recommendations for Agency Actions Following the Events in Japan''.	ML112310021
SECY–11–0093, ''Enclosure: The Near-Term Task Force Review of Insights From the Fukushima Dai-ichi Accident''	ML111861807
Staff Requirements Memorandum for SECY–11–0117, ''Proposed Charter for the Longer-Term Review of Lessons Learned From the March 11, 2011, Japanese Earthquake and Tsunami''.	ML112920034
SECY–11–0117, ''Proposed Charter for the Longer-Term Review of Lessons Learned From the March 11, 2011, Japanese Earthquake and Tsunami''.	ML11231A723
SECY–11–0124, ''Recommended Actions To Be Taken Without Delay From The Near-Term Task Force Report''	ML11245A127
SECY–11–0137, ''Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned''	ML11269A204
Staff Requirements Memorandum for SECY–12–0025, ''Proposed Orders and Requests for Information in Response to Lessons Learned From Japan's March 11, 2011, Great Tōhoku Earthquake and Tsunami''.	ML120690347
SECY–12–0025, ''Proposed Orders and Requests for Information in Response to Lessons Learned From Japan's March 11, 2011, Great Tōhoku Earthquake and Tsunami''.	ML12039A103
SECY–14–0046, ''Fifth 6-Month Status Update on Response to Lessons Learned From Japan's March 11, 2011, Great Tōhoku Earthquake and Subsequent Tsunami''.	ML14064A523
Regulatory Guide 1.13, ''Spent Fuel Storage Facility Design Basis''	ML070310035
Regulatory Guide 1.20, ''Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing''.	ML070260376
Regulatory Guide 1.27, ''Ultimate Heat Sink for Nuclear Power Plants (for Comment)''	ML003739996
Regulatory Guide 1.76, ''Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants''	ML070360253
Regulatory Guide 1.117, ''Tornado Design Classification''	ML003739346
Regulatory Guide 1.143, ''Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants''.	ML003740200
Regulatory Guide 1.206, Section C.I.1, ''Standard Format and Content of Combined License Applications—Introduction and General Description of the Plant''.	ML070630005
Regulatory Guide 1.221, ''Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants''	ML110940303
NUREG–0700, Revision 2, ''Human-Systems Interface Design Review Guidelines'' (three volumes)	ML021700337 ML021700342 ML021700371
NUREG–0711, Revision 2, ''Human Factors Engineering Program Review Model''	ML040770540
NUREG–0711, Revision 3, ''Human Factors Engineering Program Review Model''	ML12324A013
NUREG–0800, Section 3.8.4, Revision 2, ''Other Seismic Category I Structures,'' Appendix D, ''Guidance on Spent Fuel Pool Racks''.	ML070550054
NUREG–0800, Section 3.9.2, Revision 3, ''Dynamic Testing and Analysis of Systems, Structures, and Components''	ML070230008
NUREG–0800, Section 3.9.5, Revision 3, ''Reactor	ML070230009
Pressure Vessel Internals''	
NUREG–0800, SRP Section 6.4, Revision 3, ''Control Room Habitability System''	ML070550069
NUREG–0800, SRP Section 9.1.2, Revision 4, ''New and Spent Fuel Storage''	ML070550057
NUREG–0800, SRP Section 13.4, Revision 3, ''Operational Programs''	ML070470463
NUREG–0800, SRP Section 13.5.2.1, Revision 2, ''Operating and Emergency Operating Procedures''	ML070100635
NUREG–0800, SRP Section 18, Revision 2, ''Human Factors Engineering''	ML070670253
NUREG–1242, ''NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document, Evolutionary Plant Designs'' (five volumes).	ML100610048 ML100430013 ML063620331 ML070600372 ML070600373
NRC Bulletin 2012–01: Design Vulnerability in Electric Power System	ML12074A115
Interim Staff Guidance DC/COL–ISG–8, ''Necessary Content of Plant-Specific Technical Specifications''	ML083310259
JLD–ISG–2012–03 Revision 0, ''Compliance With Order EA–12–051, Reliable Spent Fuel Pool Instrumentation,''	ML12221A339
NEI 12–02, Revision 1, ''Industry Guidance for Compliance With NRC Order EA–12–051, To Modify Licenses With Regard to Reliable Spent Fuel Pool Instrumentation''.	ML122400399
''Clarifications Requested by NRC Staff on Economic Simplified Boiling Water Reactor Fuel Design''	ML11269A093
Audit Report, ''ESBWR Fuel Seismic Audit Summary''	ML112860614

Document	ADAMS Accession No./ web link/ **Federal Register** citation
Notice of Violation, "ESBWR AIA Inspection Report Inspection, NRC Inspection Report No. 0520000/10/2010–201 and Notice of Violation".	ML102740292
Reply to Notice of Violation, NRC Inspection Report 052000010–10–201	ML103010047
GE–Hitachi Nuclear Energy Americas, LLC, Reply to Notice of Violation, NRC IR 052000010–10–201	ML103400150
ACRS Memorandum—Final Rule—ESBWR Design Certification (RIN 3150–AI85)	ML113120076
ACRS Memorandum—ESBWR Design Certification Rulemaking and Supplemental Final Safety Evaluation Report	ML11340A043
ACRS Memorandum—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.	ML14107A263
ACRS Memorandum—Final Rule—ESBWR Design Certification (RIN 3150–AI85)	ML14196A207
Regulatory History of Design Certification [6]	ML003761550

X. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995 (Act), Pub. L. 104–113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this final rule, the NRC is approving the ESBWR standard plant design for use in nuclear power plant licensing under 10 CFR part 50 or part 52. Design certifications are not generic rulemakings establishing a generally applicable standard with which all 10 CFR parts 50 and 52 nuclear power plant licensees or applicants for SDAs, design certifications, or manufacturing licenses must comply. Design certifications are NRC approvals of specific nuclear power plant designs by rulemaking. Furthermore, design certifications are initiated by an applicant for rulemaking, rather than by the NRC. For these reasons, the NRC concludes that the Act does not apply to this final rule.

XI. Finding of No Significant Environmental Impact: Availability

The NRC has determined under NEPA, and the NRC's regulations in subpart A, ''National Environmental Policy Act; Regulations Implementing Section 102(2),'' of 10 CFR part 51, ''Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,'' that this DCR is not a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement (EIS) is not required. The NRC's generic determination in this regard is reflected in 10 CFR 51.32(b)(1). The basis for the NRC's categorical exclusion in this regard, as discussed in the 2007 final rule amending 10 CFR parts 51 and 52 (August 28, 2007; 72 FR 49352–49566), is based upon the following considerations. A DCR does not authorize the siting, construction, or operation of a facility referencing any particular design; it only codifies the ESBWR design in a rule. The NRC will evaluate the environmental impacts and issue an EIS as appropriate under NEPA as part of the application for the construction and operation of a facility referencing any particular DCR.

In addition, consistent with 10 CFR 51.30(d) and 10 CFR 51.32(b), the NRC has prepared a final EA (ADAMS Accession No. ML111730382) for the ESBWR design addressing various design alternatives to prevent and mitigate severe accidents. The EA is based, in part, upon the NRC's review of GEH's evaluation of various design alternatives to prevent and mitigate severe accidents in NEDO–33306, Revision 4, ''ESBWR Severe Accident Mitigation Design Alternatives.'' Based upon review of GEH's evaluation, the Commission concludes that: (1) GEH identified a reasonably complete set of potential design alternatives to prevent and mitigate severe accidents for the ESBWR design; (2) none of the potential design alternatives are justified on the basis of cost-benefit considerations; and (3) it is unlikely that other design changes would be identified and justified during the term of the design certification on the basis of cost-benefit considerations because the estimated core damage frequencies for the ESBWR are very low on an absolute scale. These issues are considered resolved for the ESBWR design.

The NRC requested comments on the draft EA but the comments received did not include anything to suggest that: (i) A rule certifying the ESBWR standard design would be a major Federal action, or (ii) the SAMDA evaluation omitted a design alternative that should have been considered or incorrectly considered the costs and benefits of the alternatives it did consider. Therefore, no change to the EA was warranted. All environmental issues concerning SAMDAs associated with the information in the final EA and NEDO–33306 are considered resolved for facility applications referencing the ESBWR design if the site characteristics at the site proposed in the facility application fall within the site parameters specified in NEDO–33306.

The final EA, upon which the Commission's finding of no significant impact is based, and the ESBWR DCD are available for examination and copying at the NRC's PDR, One White Flint North, Room O–1 F21, 11555 Rockville Pike, Rockville, Maryland 20852.

XII. Paperwork Reduction Act

This rule contains new or amended information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501, *et seq.*). These requirements were approved by the Office of Management and Budget (OMB), control number 3150–0151. The burden to the public for these information collections is estimated to average 15 hours per response.

Send comments on any aspect of these information collections, including suggestions for reducing the burden, to the Records and FOIA/Privacy Services Branch (T–5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or by Internet electronic mail to *INFOCOLLECTS.RESOURCE@NRC.GOV;* and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB–10202, (3150–0151), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond

[6] The regulatory history of the NRC's design certification reviews is a package of documents that is available in NRC's PDR and Electronic Reading Room. This history spans the period during which the NRC simultaneously developed the regulatory standards for reviewing these designs and the form and content of the rules that certified the designs.

to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

XIII. Regulatory Analysis

The NRC has not prepared a regulatory analysis for this final rule. The NRC prepares regulatory analyses for rulemakings that establish generic regulatory requirements applicable to all licensees. Design certifications are not generic rulemakings in the sense that design certifications do not establish standards or requirements with which all licensees must comply. Rather, design certifications are NRC approvals of specific nuclear power plant designs by rulemaking, which then may be voluntarily referenced by applicants for COLs. Furthermore, design certification rulemakings are initiated by an applicant for a design certification, rather than the NRC. Preparation of a regulatory analysis in this circumstance would not be useful because the design to be certified is proposed by the applicant rather than the NRC. For these reasons, the NRC concludes that preparation of a regulatory analysis is neither required nor appropriate.

XIV. Regulatory Flexibility Certification

Under the Regulatory Flexibility Act (5 U.S.C. 605(b)), the NRC certifies that this rule does not have a significant economic impact on a substantial number of small entities. This final rule provides for certification of a nuclear power plant design. Neither the design certification applicant, nor prospective nuclear power plant licensees who reference this DCR, fall within the scope of the definition of ''small entities'' set forth in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810). Thus, this rule does not fall within the purview of the Regulatory Flexibility Act.

XV. Backfitting and Issue Finality

The NRC has determined that this final rule does not constitute a backfit as defined in the backfit rule (10 CFR 50.109) and that it is not inconsistent with any applicable issue finality provision in 10 CFR part 52.

This initial DCR does not constitute backfitting as defined in the backfit rule (10 CFR 50.109) because there are no operating licenses under 10 CFR part 50 referencing this DCR.

This initial DCR is not inconsistent with any applicable issue finality provision in 10 CFR part 52 because it does not impose new or changed requirements on existing DCRs in appendices A through D to 10 CFR part 52, and no COLs or manufacturing licenses issued by the NRC at this time reference a final ESBWR DCR. Although there are several COL applications referencing the *application* for the ESBWR DCR, there is no issue finality protection accorded to such a COL applicant under either 10 CFR 52.63 or 10 CFR 52.83.

For these reasons, neither a backfit analysis nor a discussion addressing the issue finality provisions in 10 CFR part 52 was prepared for this rule.

XVI. Congressional Review Act

In accordance with the Congressional Review Act of 1996 (5 U.S.C. 801–808), the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of the Office of Management and Budget.

XVII. Plain Writing

The Plain Writing Act of 2010 (Pub. L. 111–274) requires Federal agencies to write documents in a clear, concise, and well-organized manner. The NRC has written this document to be consistent with the Plain Writing Act as well as the Presidential Memorandum, ''Plain Language in Government Writing,'' published June 10, 1998 (63 FR 31883).

XVIII. Availability of Guidance

The NRC will not be issuing guidance for this rulemaking. The NRC has previously published relevant guidance in RG 1.206, ''Combined License Applications for Nuclear Power Plants (LWR Edition).'' This RG provides guidance for preparing an application for a COL under 10 CFR part 52, including guidance related to referencing a design certification in that application. Each DCR is similar in its content and structure. Therefore, the existing guidance in RG 1.206 is adequate to support this DCR.

List of Subjects in 10 CFR Part 52

Administrative practice and procedure, Antitrust, Backfitting, Combined license, Early site permit, Emergency planning, Fees, Incorporation by reference, Inspection, Limited work authorization, Nuclear power plants and reactors, Probabilistic risk assessment, Prototype, Reactor siting criteria, Redress of site, Reporting and recordkeeping requirements, Standard design, Standard design certification.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 552 and 553; the NRC is adopting the following amendments to 10 CFR part 52.

PART 52—LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

■ 1. The authority citation for 10 CFR part 52 continues to read as follows:

Authority: Atomic Energy Act secs. 103, 104, 147, 149, 161, 181, 182, 183, 185, 186, 189, 223, 234 (42 U.S.C. 2133, 2201, 2167, 2169, 2232, 2233, 2235, 2236, 2239, 2282); Energy Reorganization Act secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); Government Paperwork Elimination Act sec. 1704 (44 U.S.C. 3504 note); Energy Policy Act of 2005, Pub. L. 109–58, 119 Stat. 594 (2005).

■ 2. In § 52.11, paragraph (b) is revised to read as follows:

§ 52.11 Information collection requirements: OMB approval.

* * * * *

(b) The approved information collection requirements contained in this part appear in §§ 52.7, 52.15, 52.16, 52.17, 52.29, 52.35, 52.39, 52.45, 52.46, 52.47, 52.57, 52.63, 52.75, 52.77, 52.79, 52.80, 52.93, 52.99, 52.110, 52.135, 52.136, 52.137, 52.155, 52.156, 52.157, 52.158, 52.171, 52.177, and appendices A, B, C, D, E, and N of this part.

■ 3. A new Appendix E to 10 CFR part 52 is added to read as follows:

Appendix E to Part 52—Design Certification Rule for the ESBWR Design

I. Introduction

Appendix E constitutes the standard design certification for the Economic Simplified Boiling-Water Reactor (ESBWR) design, in accordance with 10 CFR part 52, subpart B. The applicant for certification of the ESBWR design is GE-Hitachi Nuclear Energy.

II. Definitions

A. *Generic design control document (generic DCD)* means the document containing the Tier 1 and Tier 2 information and generic technical specifications that is incorporated by reference into this appendix.

B. *Generic technical specifications (generic TS)* means the information required by 10 CFR 50.36 and 50.36a for the portion of the plant that is within the scope of this appendix.

C. *Plant-specific DCD* means that portion of the combined license (COL) final safety analysis report (FSAR) that sets forth both the generic DCD information and any plant-specific changes to generic DCD information.

D. *Tier 1* means the portion of the design-related information contained in the generic DCD that is approved and certified by this appendix (Tier 1 information). The design descriptions, interface requirements, and site parameters are derived from Tier 2 information. Tier 1 information includes:

1. Definitions and general provisions;

2. Design descriptions;

3. Inspections, tests, analyses, and acceptance criteria (ITAACs);

4. Significant site parameters; and

5. Significant interface requirements.

E. *Tier 2* means the portion of the design-related information contained in the generic DCD that is approved but not certified by this appendix (Tier 2 information). Compliance with Tier 2 is required, but generic changes to and plant-specific departures from Tier 2 are governed by Section VIII of this appendix. Compliance with Tier 2 provides a sufficient, but not the only acceptable, method for complying with Tier 1. Compliance methods differing from Tier 2 must satisfy the change process in Section VIII of this appendix. Regardless of these differences, an applicant or licensee must meet the requirement in paragraph III.B of this appendix to reference Tier 2 when referencing Tier 1. Tier 2 information includes:

1. Information required by §§ 52.47(a) and 52.47(c), with the exception of generic TS and conceptual design information;

2. Supporting information on the inspections, tests, and analyses that will be performed to demonstrate that the acceptance criteria in the ITAACs have been met;

3. COL action items (COL license information), which identify certain matters that must be addressed in the site-specific portion of the FSAR by an applicant who references this appendix. These items constitute information requirements but are not the only acceptable set of information in the FSAR. An applicant may depart from or omit these items, provided that the departure or omission is identified and justified in the FSAR. After issuance of a construction permit or COL, these items are not requirements for the licensee unless such items are restated in the FSAR; and

4. The availability controls in Appendix 19ACM of the DCD.

F. *Tier 2** means the portion of the Tier 2 information, designated as such in the generic DCD, which is subject to the change process in paragraph VIII.B.6 of this appendix. This designation expires for some Tier 2* information under paragraph VIII.B.6 of this appendix.

G. *Departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses* means:

1. Changing any of the elements of the method described in the plant-specific DCD unless the results of the analysis are conservative or essentially the same; or

2. Changing from a method described in the plant-specific DCD to another method unless that method has been approved by the NRC for the intended application.

H. All other terms in this appendix have the meaning set out in 10 CFR 50.2, 10 CFR 52.1, or Section 11 of the Atomic Energy Act of 1954, as amended, as applicable.

III. Scope and Contents

A. Incorporation by reference approval. The documents in Table 1 are approved for incorporation by reference by the Director of the Office of the Federal Register under 5 U.S.C. 552(a) and 1 CFR part 51. You may obtain copies of the generic DCD from Jerald G. Head, Senior Vice President, Regulatory Affairs, GE-Hitachi Nuclear Energy, 3901 Castle Hayne Road, MC A–18, Wilmington, NC 28401, telephone: 1–910–819–5692. You can view the generic DCD online in the NRC Library at *http://www.nrc.gov/reading-rm/adams.html.* In ADAMS, search under the ADAMS Accession No. listed in Table 1. If you do not have access to ADAMS or if you have problems accessing documents located in ADAMS, contact the NRC's Public Document Room (PDR) reference staff at 1–800–397–4209, 1–301–415–3747, or by email at *PDR.Resource@nrc.gov.* These documents can also be viewed at the Federal rulemaking Web site, *http://www.regulations.gov,* by searching for documents filed under Docket ID NRC–2010–0135. Copies of these documents are available for examination and copying at the NRC's PDR located at Room O–1F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852. Copies are also available for examination at the NRC Library located at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852, telephone: 301–415–5610, email: *Library.Resource@nrc.gov.* All approved material is available for inspection at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 1–202–741–6030 or go to *http://www.archives.gov/federal-register/cfr/ibrlocations.html.*

TABLE 1—DOCUMENTS APPROVED FOR INCORPORATION BY REFERENCE

Document No.	Document title	ADAMS Accession No.
GE Hitachi:		
26A6642AB Rev. 10	ESBWR Design Control Document, Revision 10, Tier 1, dated April 2014	ML14104A929 (package)
26A6642AB Rev. 10	ESBWR Design Control Document, Revision 10, Tier 2, dated April 2014	ML14104A929 (package)
Bechtel Power Corporation:		
BC–TOP–3–A	"Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," Topical Report, Revision 3, August 1974.	ML14093A218
BC–TOP–9A	"Design of Structures for Missile Impact," Topical Report, Revision 2, September 1974.	ML14093A217
General Electric:		
GEZ–4982A	General Electric Large Steam Turbine Generator Quality Control Program, The STG Global Supply Chain Quality Management System (MFGGLO–GEZ–0010) Revision 1.2, February 7, 2006.	ML14093A215
GE Nuclear Energy:		
NEDO–11209–04A	"GE Nuclear Energy Quality Assurance Program Description," Class 1, Revision 8, March 31, 1989.	ML14093A209
NEDO–31960–A	"BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," Class I, November 1995.	ML14093A212
NEDO–31960–A—Supplement 1	"BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," Class I, November 1995.	ML14093A211
NEDO–32465–A	GE Nuclear Energy and BWR Owners' Group, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," Class I, August 1996.	ML14093A210
GE-Hitachi Nuclear Energy:		
NEDO–33181	"NP–2010 COL Demonstration Project Quality Assurance Plan," Revision 6, August 2009.	ML14248A297
NEDO–33219	"ESBWR Human Factors Engineering Functional Requirements Analysis Implementation Plan," Revision 4, Class I, February 2010.	ML100350104
NEDO–33260	"Quality Assurance Requirements for Suppliers of Equipment and Services to the GEH ESBWR Project," Revision 5, Class I, April 2008.	ML14248A648
NEDO–33262	"ESBWR Human Factors Engineering Operating Experience Review Implementation Plan," Revision 3, Class I, January 2010.	ML100340030
NEDO–33266	"ESBWR Human Factors Engineering Staffing and Qualifications Implementation Plan," Revision 3, Class I, January 2010.	ML100350167

TABLE 1—DOCUMENTS APPROVED FOR INCORPORATION BY REFERENCE—Continued

Document No.	Document title	ADAMS Accession No.
NEDO–33267	"ESBWR Human Factors Engineering Human Reliability Analysis Implementation Plan," Revision 4, Class I, January 2010.	ML100330609
NEDO–33277	"ESBWR Human Factors Engineering Human Performance Monitoring Implementation Plan," Revision 4, Class I, January 2010.	ML100270770
NEDO–33278	"ESBWR Human Factors Engineering Design Implementation Plan," Revision 4, Class I, January 2010.	ML100270468
NEDO–33289	"ESBWR Reliability Assurance Program," Revision 2, Class II, September 2008.	ML14248A662
NEDO–33337	"ESBWR Initial Core Transient Analyses," Revision 1, Class I, April 2009	ML091130628
NEDO–33338	"ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," Revision 1, Class I, May 2009.	ML091380173
NEDO–33373–A	"Dynamic, Load-Drop, and Thermal-Hydraulic Analyses for ESBWR Fuel Racks," Revision 5, Class I, October 2010.	ML102990226 (part 1) ML102990228 (part 2)
NEDO–33411	"Risk Significance of Structures, Systems and Components for the Design Phase of the ESBWR," Revision 2, Class I, February 2010.	ML100610417

B. An applicant or licensee referencing this appendix, in accordance with Section IV of this appendix, shall incorporate by reference and comply with the requirements of this appendix, including Tier 1, Tier 2 (including the availability controls in Appendix 19ACM of the DCD), and the generic TS except as otherwise provided in this appendix. Conceptual design information in the generic DCD and the evaluation of severe accident mitigation design alternatives in NEDO–33306, Revision 4, ''ESBWR Severe Accident Mitigation Design Alternatives,'' are not part of this appendix.

C. If there is a conflict between Tier 1 and Tier 2 of the DCD, then Tier 1 controls.

D. If there is a conflict between the generic DCD and either the application for design certification of the ESBWR design or NUREG–1966, ''Final Safety Evaluation Report Related to Certification of the ESBWR Standard Design,'' (FSER) and Supplement No. 1 to NUREG–1966, then the generic DCD controls.

E. Design activities for structures, systems, and components that are wholly outside the scope of this appendix may be performed using site characteristics, provided the design activities do not affect the DCD or conflict with the interface requirements.

IV. Additional Requirements and Restrictions

A. An applicant for a COL who references this appendix shall, in addition to complying with the requirements of §§ 52.77, 52.79, and 52.80, comply with the following requirements:

1. Incorporate by reference, as part of its application, this appendix.

2. Include, as part of its application:

a. A plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for the ESBWR design, either by including or incorporating by reference the generic DCD information, and as modified and supplemented by the applicant's exemptions and departures;

b. The reports on departures from and updates to the plant-specific DCD required by paragraph X.B of this appendix;

c. Plant-specific TS, consisting of the generic and site-specific TS that are required by 10 CFR 50.36 and 50.36a;

d. Information demonstrating that the site characteristics fall within the site parameters and that the interface requirements have been met;

e. Information that addresses the COL action items;

f. Information required by § 52.47(a) that is not within the scope of this appendix;

g. Information demonstrating that hurricane loads on those structures, systems, and components described in Section 3.3.2 of the generic DCD are either bounded by the total tornado loads analyzed in Section 3.3.2 of the generic DCD or will meet applicable NRC requirements with consideration of hurricane loads in excess of the total tornado loads; and hurricane-generated missile loads on those structures, systems, and components described in Section 3.5.2 of the generic DCD are either bounded by tornado-generated missile loads analyzed in Section 3.5.1.4 of the generic DCD or will meet applicable NRC requirements with consideration of hurricane-generated missile loads in excess of the tornado-generated missile loads; and

h. Information demonstrating that the spent fuel pool level instrumentation is designed to allow the connection of an independent power source, and that the instrumentation will maintain its design accuracy following a power interruption or change in power source without requiring recalibration.

3. Include, in the plant-specific DCD, the sensitive, unclassified, non-safeguards information (including proprietary information and security-related information) and safeguards information referenced in the ESBWR generic DCD.

4. Include, as part of its application, a demonstration that an entity other than GE-Hitachi Nuclear Energy is qualified to supply the ESBWR design unless GE-Hitachi Nuclear Energy supplies the design for the applicant's use.

B. The Commission reserves the right to determine in what manner this appendix may be referenced by an applicant for a construction permit or operating license under 10 CFR part 50.

V. Applicable Regulations

A. Except as indicated in paragraph B of this section, the regulations that apply to the ESBWR design are in 10 CFR parts 20, 50, 73, and 100, codified as of October 6, 2014, that are applicable and technically relevant, as described in the FSER (NUREG–1966) and Supplement No. 1.

B. The ESBWR design is exempt from portions of the following regulations:

1. Paragraph (f)(2)(iv) of 10 CFR 50.34—Contents of Applications: Technical Information—codified as of October 6, 2014.

VI. Issue Resolution

A. The Commission has determined that the structures, systems, components, and design features of the ESBWR design comply with the provisions of the Atomic Energy Act of 1954, as amended, and the applicable regulations identified in Section V of this appendix; and therefore, provide adequate protection to the health and safety of the public. A conclusion that a matter is resolved includes the finding that additional or alternative structures, systems, components, design features, design criteria, testing, analyses, acceptance criteria, or justifications are not necessary for the ESBWR design.

B. The Commission considers the following matters resolved within the meaning of § 52.63(a)(5) in subsequent proceedings for issuance of a COL, amendment of a COL, or renewal of a COL, proceedings held under § 52.103, and enforcement proceedings involving plants referencing this appendix:

1. All nuclear safety issues associated with the information in the FSER and Supplement No. 1; Tier 1, Tier 2 (including referenced information, which the context indicates is intended as requirements, and the availability controls in Appendix 19ACM of the DCD), the 20 documents referenced in Table 1 of paragraph III.A, and the rulemaking record for certification of the ESBWR design, with the exception of: generic TS and other operational requirements such as human factors engineering procedure development and training program development in Sections 18.9 and 18.10 of the generic DCD; hurricane loads on those structures, systems, and components described in Section 3.3.2 of the generic DCD that are not bounded by the total tornado loads analyzed in Section 3.3.2 of the generic DCD; hurricane-generated missile loads on those structures, systems, and

components described in Section 3.5.2 of the generic DCD that are not bounded by tornado-generated missile loads analyzed in Section 3.5.1.4 of the generic DCD; and spent fuel pool level instrumentation design in regard to the connection of an independent power source, and how the instrumentation will maintain its design accuracy following a power interruption or change in power source without recalibration;

2. All nuclear safety and safeguards issues associated with the referenced information in the 50 non-public documents in Tables 1.6–1 and 1.6–2 of Tier 2 of the DCD which contain sensitive unclassified non-safeguards information (including proprietary information and security-related information) and safeguards information and which, in context, are intended as requirements in the generic DCD for the ESBWR design, with the exception of human factors engineering procedure development and training program development in Chapters 18.9 and 18.10 of the generic DCD;

3. All generic changes to the DCD under and in compliance with the change processes in paragraphs VIII.A.1 and VIII.B.1 of this appendix;

4. All exemptions from the DCD under and in compliance with the change processes in paragraphs VIII.A.4 and VIII.B.4 of this appendix, but only for that plant;

5. All departures from the DCD that are approved by license amendment, but only for that plant;

6. Except as provided in paragraph VIII.B.5.f of this appendix, all departures from Tier 2 under and in compliance with the change processes in paragraph VIII.B.5 of this appendix that do not require prior NRC approval, but only for that plant;

7. All environmental issues concerning severe accident mitigation design alternatives associated with the information in the NRC's Environmental Assessment for the ESBWR design (ADAMS Accession No. ML111730382) and NEDO–33306, Revision 4, "ESBWR Severe Accident Mitigation Design Alternatives," (ADAMS Accession No. ML102990433) for plants referencing this appendix whose site characteristics fall within those site parameters specified in NEDO–33306.

C. The Commission does not consider operational requirements for an applicant or licensee who references this appendix to be matters resolved within the meaning of § 52.63(a)(5). The Commission reserves the right to require operational requirements for an applicant or licensee who references this appendix by rule, regulation, order, or license condition.

D. Except under the change processes in Section VIII of this appendix, the Commission may not require an applicant or licensee who references this appendix to:

1. Modify structures, systems, components, or design features as described in the generic DCD;

2. Provide additional or alternative structures, systems, components, or design features not discussed in the generic DCD; or

3. Provide additional or alternative design criteria, testing, analyses, acceptance criteria, or justification for structures, systems, components, or design features discussed in the generic DCD.

E. The NRC will specify at an appropriate time the procedures to be used by an interested person who seeks to review portions of the design certification or references containing safeguards information or sensitive unclassified non-safeguards information (including proprietary information, such as trade secrets and commercial or financial information obtained from a person that are privileged or confidential (10 CFR 2.390 and 10 CFR part 9), and security-related information), for the purpose of participating in the hearing required by § 52.85, the hearing provided under § 52.103, or in any other proceeding relating to this appendix in which interested persons have a right to request an adjudicatory hearing.

VII. Duration of This Appendix

This appendix may be referenced for a period of 15 years from November 14, 2014, except as provided for in §§ 52.55(b) and 52.57(b). This appendix remains valid for an applicant or licensee who references this appendix until the application is withdrawn or the license expires, including any period of extended operation under a renewed license.

VIII. Processes for Changes and Departures

A. Tier 1 information

1. Generic changes to Tier 1 information are governed by the requirements in § 52.63(a)(1).

2. Generic changes to Tier 1 information are applicable to all applicants or licensees who reference this appendix, except those for which the change has been rendered technically irrelevant by action taken under paragraphs A.3 or A.4 of this section.

3. Departures from Tier 1 information that are required by the Commission through plant-specific orders are governed by the requirements in § 52.63(a)(4).

4. Exemptions from Tier 1 information are governed by the requirements in §§ 52.63(b)(1) and 52.98(f). The Commission will deny a request for an exemption from Tier 1, if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design.

B. Tier 2 information

1. Generic changes to Tier 2 information are governed by the requirements in 10 CFR 52.63(a)(1).

2. Generic changes to Tier 2 information are applicable to all applicants or licensees who reference this appendix, except those for which the change has been rendered technically irrelevant by action taken under paragraphs B.3, B.4, B.5, or B.6 of this section.

3. The Commission may not require new requirements on Tier 2 information by plant-specific order while this appendix is in effect under 10 CFR 52.55 or 52.61, unless:

a. A modification is necessary to secure compliance with the Commission's regulations applicable and in effect at the time this appendix was approved, as set forth in Section V of this appendix, or to ensure adequate protection of the public health and safety or the common defense and security; and

b. Special circumstances as defined in 10 CFR 50.12(a) are present.

4. An applicant or licensee who references this appendix may request an exemption from Tier 2 information. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of 10 CFR 50.12(a). The Commission will deny a request for an exemption from Tier 2, if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design. The grant of an exemption to an applicant must be subject to litigation in the same manner as other issues material to the license hearing. The grant of an exemption to a licensee must be subject to an opportunity for a hearing in the same manner as license amendments.

5.a. An applicant or licensee who references this appendix may depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the TS, or requires a license amendment under paragraph B.5.b or B.5.c of this section. When evaluating the proposed departure, an applicant or licensee shall consider all matters described in the plant-specific DCD.

b. A proposed departure from Tier 2, other than one affecting resolution of a severe accident issue identified in the plant-specific DCD or one affecting information required by § 52.47(a)(28) to address aircraft impacts, requires a license amendment if it would:

(1) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD;

(2) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety and previously evaluated in the plant-specific DCD;

(3) Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD;

(4) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD;

(5) Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD;

(6) Create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD;

(7) Result in a design-basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered; or

(8) Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

c. A proposed departure from Tier 2 affecting resolution of an ex-vessel severe accident design feature identified in the plant-specific DCD, requires a license amendment if:

(1) There is a substantial increase in the probability of an ex-vessel severe accident

such that a particular ex-vessel severe accident previously reviewed and determined to be not credible could become credible; or

(2) There is a substantial increase in the consequences to the public of a particular ex-vessel severe accident previously reviewed.

d. A proposed departure from Tier 2 information required by § 52.47(a)(28) to address aircraft impacts shall consider the effect of the changed design feature or functional capability on the original aircraft impact assessment required by 10 CFR 50.150(a). The applicant or licensee shall describe in the plant-specific DCD how the modified design features and functional capabilities continue to meet the aircraft impact assessment requirements in 10 CFR 50.150(a)(1).

e. If a departure requires a license amendment under paragraph B.5.b or B.5.c of this section, it is governed by 10 CFR 50.90.

f. A departure from Tier 2 information that is made under paragraph B.5 of this section does not require an exemption from this appendix.

g. A party to an adjudicatory proceeding for either the issuance, amendment, or renewal of a license or for operation under § 52.103(a), who believes that an applicant or licensee who references this appendix has not complied with paragraph VIII.B.5 of this appendix when departing from Tier 2 information, may petition to admit into the proceeding such a contention. In addition to compliance with the general requirements of 10 CFR 2.309, the petition must demonstrate that the departure does not comply with paragraph VIII.B.5 of this appendix. Further, the petition must demonstrate that the change bears on an asserted noncompliance with an ITAAC acceptance criterion in the case of a § 52.103 preoperational hearing, or that the change bears directly on the amendment request in the case of a hearing on a license amendment. Any other party may file a response. If, on the basis of the petition and any response, the presiding officer determines that a sufficient showing has been made, the presiding officer shall certify the matter directly to the Commission for determination of the admissibility of the contention. The Commission may admit such a contention if it determines the petition raises a genuine issue of material fact regarding compliance with paragraph VIII.B.5 of this appendix.

6.a. An applicant who references this appendix may not depart from Tier 2* information, which is designated with italicized text or brackets and an asterisk in the generic DCD, without NRC approval. The departure will not be considered a resolved issue, within the meaning of Section VI of this appendix and § 52.63(a)(5).

b. A licensee who references this appendix may not depart from the following Tier 2* matters without prior NRC approval. A request for a departure will be treated as a request for a license amendment under 10 CFR 50.90.

(1) Fuel mechanical and thermal-mechanical design evaluation reports, including fuel burnup limits.

(2) Control rod mechanical and nuclear design reports.

(3) Fuel nuclear design report.

(4) Critical power correlation.

(5) Fuel licensing acceptance criteria.

(6) Control rod licensing acceptance criteria.

(7) Mechanical and structural design of spent fuel storage racks.

(8) Steam dryer pressure load analysis methodology.

c. A licensee who references this appendix may not, before the plant first achieves full power following the finding required by § 52.103(g), depart from the following Tier 2* matters except under paragraph B.6.b of this section. After the plant first achieves full power, the following Tier 2* matters revert to Tier 2 status and are subject to the departure provisions in paragraph B.5 of this section.

(1) ASME Boiler and Pressure Vessel Code, Section III, Subsections NE (Division 1) and CC (Division 2) for containment vessel design.

(2) American Concrete Institute 349 and American National Standards Institute/American Institute of Steel Construction—N690.

(3) Power-operated valves.

(4) Equipment seismic qualification methods.

(5) Piping design acceptance criteria.

(6) Instrument setpoint methodology.

(7) Safety-Related Distribution Control and Information System performance specification and architecture.

(8) Safety System Logic and Control hardware and software.

(9) Human factors engineering design and implementation.

(10) First of a kind testing for reactor stability (first plant only).

(11) Reactor precritical heatup with reactor water cleanup/shutdown cooling (first plant only).

(12) Isolation condenser system heatup and steady state operation (first plant only).

(13) Power maneuvering in the feedwater temperature operating domain (first plant only).

(14) Load maneuvering capability (first plant only).

(15) Defense-in-depth stability solution evaluation test (first plant only).

d. Departures from Tier 2* information that are made under paragraph B.6 of this section do not require an exemption from this appendix.

C. Operational requirements.

1. Generic changes to generic TS and other operational requirements that were completely reviewed and approved in the design certification rulemaking and do not require a change to a design feature in the generic DCD are governed by the requirements in 10 CFR 50.109. Generic changes that require a change to a design feature in the generic DCD are governed by the requirements in paragraphs A or B of this section.

2. Generic changes to generic TS and other operational requirements are applicable to all applicants who reference this appendix, except those for which the change has been rendered technically irrelevant by action taken under paragraphs C.3 or C.4 of this section.

3. The Commission may require plant-specific departures on generic TS and other operational requirements that were completely reviewed and approved, provided a change to a design feature in the generic DCD is not required and special circumstances as defined in 10 CFR 2.335 are present. The Commission may modify or supplement generic TS and other operational requirements that were not completely reviewed and approved or require additional TS and other operational requirements on a plant-specific basis, provided a change to a design feature in the generic DCD is not required.

4. An applicant who references this appendix may request an exemption from the generic TS or other operational requirements. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of § 52.7. The grant of an exemption must be subject to litigation in the same manner as other issues material to the license hearing.

5. A party to an adjudicatory proceeding for the issuance, amendment, or renewal of a license, or for operation under § 52.103(a), who believes that an operational requirement approved in the DCD or a TS derived from the generic TS must be changed may petition to admit such a contention into the proceeding. The petition must comply with the general requirements of 10 CFR 2.309 and must demonstrate why special circumstances as defined in 10 CFR 2.335 are present, or demonstrate compliance with the Commission's regulations in effect at the time this appendix was approved, as set forth in Section V of this appendix. Any other party may file a response to the petition. If, on the basis of the petition and any response, the presiding officer determines that a sufficient showing has been made, the presiding officer shall certify the matter directly to the Commission for determination of the admissibility of the contention. All other issues with respect to the plant-specific TS or other operational requirements are subject to a hearing as part of the license proceeding.

6. After issuance of a license, the generic TS have no further effect on the plant-specific TS. Changes to the plant-specific TS will be treated as license amendments under 10 CFR 50.90.

IX. [Reserved]

X. Records and Reporting

A. Records

1. The applicant for this appendix shall maintain a copy of the generic DCD that includes all generic changes it makes to Tier 1 and Tier 2, and the generic TS and other operational requirements. The applicant shall maintain the sensitive unclassified non-safeguards information (including proprietary information and security-related information) and safeguards information referenced in the generic DCD for the period that this appendix may be referenced, as specified in Section VII of this appendix.

2. An applicant or licensee who references this appendix shall maintain the plant-specific DCD to accurately reflect both generic changes to the generic DCD and plant-specific departures made under Section

VIII of this appendix throughout the period of application and for the term of the license (including any period of renewal).

3. An applicant or licensee who references this appendix shall prepare and maintain written evaluations that provide the bases for the determinations required by Section VIII of this appendix. These evaluations must be retained throughout the period of application and for the term of the license (including any period of renewal).

4.a. The applicant for the ESBWR design shall maintain a copy of the aircraft impact assessment performed to comply with the requirements of 10 CFR 50.150(a) for the term of the certification (including any period of renewal).

b. An applicant or licensee who references this appendix shall maintain a copy of the aircraft impact assessment performed to comply with the requirements of 10 CFR 50.150(a) throughout the pendency of the application and for the term of the license (including any period of renewal).

B. Reporting

1. An applicant or licensee who references this appendix shall submit a report to the NRC containing a brief description of any plant-specific departures from the DCD, including a summary of the evaluation of each. This report must be filed in accordance with the filing requirements applicable to reports in § 52.3.

2. An applicant or licensee who references this appendix shall submit updates to its plant-specific DCD that reflect the generic changes to and plant-specific departures from the generic DCD made under Section VIII of this appendix. These updates shall be filed under the filing requirements applicable to final safety analysis report updates in 10 CFR 52.3 and 50.71(e).

3. The reports and updates required by paragraphs X.B.1 and X.B.2 of this appendix must be submitted as follows:

a. On the date that an application for a license referencing this appendix is submitted, the application must include the report and any updates to the generic DCD.

b. During the interval from the date of application for a license to the date the Commission makes its finding required by § 52.103(g), the report must be submitted semi-annually. Updates to the plant-specific DCD must be submitted annually and may be submitted along with amendments to the application.

c. After the Commission makes the finding required by § 52.103(g), the reports and updates to the plant-specific DCD must be submitted, along with updates to the site-specific portion of the final safety analysis report for the facility, at the intervals required by 10 CFR 50.59(d)(2) and 50.71(e)(4), respectively, or at shorter intervals as specified in the license.

Dated at Rockville, Maryland, this 6th day of October, 2014.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,

Secretary of the Commission.

[FR Doc. 2014–24362 Filed 10–14–14; 8:45 am]

BILLING CODE 7590–01–P