IN RESPONSE, PLEASE
REFER TO: M140916A

September 16, 2014

MEMORANDUM FOR: Mark A. Satorius
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - AFFIRMATION SESSION, 9:50 A.M., TUESDAY, SEPTEMBER 16, 2014, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

I. SECY-14-0081 – Final Rule: Economic Simplified Boiling-Water Reactor Design Certification

The Commission approved a final rule adding Appendix E to10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," to certify the Economic Simplified Boiling-Water Reactor (ESBWR) standard design, subject to the attached changes, and the exclusion of the following matters from issue finality and issue resolution:

a) Human Factors Engineering procedures and training.
b) Loads on applicable structures, systems and components (SSCs) from hurricanes that are not bounded by other loads analyzed in the ESBWR Design Control Document (DCD).
c) Loads on applicable SSCs from hurricane-generated missiles to the extent that such loads are not bounded by other loads analyzed in the ESBWR DCD.
d) Spent fuel pool instrumentation design allows the connection of an independent power source.
e) Spent fuel pool instrumentation maintains its design accuracy following a power interruption or change in power source without recalibration.

The Commission has approved the changes to the advanced supplemental Safety Evaluation Report that were incorporated into the draft supplemental Final Safety Evaluation Report and certified that this rule will not have a negative economic impact on a substantial number of small entities under the Regulatory Flexibility Act. The Commission has approved the determination that neither the backfit rule (10 CFR 50.109, "Backfitting"), nor any of the issue finality provisions in 10 CFR Part 52, apply to the issuance of this final Design Certification Rule.

Following incorporation of these changes, the Federal Register notice should be reviewed by the Rulemaking, Directives, and Editing Branch in the Office of Administration and forwarded to the Office of the Secretary for signature and publication.

Attachment: Changes to the Final Rule in SECY-14-0081

cc: Chairman Macfarlane
Commissioner Svinicki
Commissioner Ostendorff
OGC
CFO
OCAA
OCA
OIG
OPA
Office Directors, Regions, ACRS, ASLBP (via E-Mail)
PDR

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; e-mail: 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 e-mail 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, e-mail: George.Tartal@nrc.gov; or David Misenhimer, Office of New Reactors, telephone: 301-415-6590, e-mail: 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

A. Regulatory Treatment of Nonsafety Systems (RTNSS)

B. Containment Performance

C. Control Room Cooling

D. Feedwater Temperature Operating Domain

E. Steam Dryer Analysis Methodology

F. Aircraft Impact Assessment (AIA)

G. American Society of Mechanical Engineers (ASME) Code Case N-782

H. Exemption for the Safety Parameter Display System

I. Hurricane-Generated Winds and Missiles

J. Loss of One or More Phases of Offsite Power

K. Spent Fuel Assembly Integrity in Spent Fuel Racks

L. Turbine Building Offgas System Design Requirements

M. ASME Boiler and Pressure Vessel Code (BPV Code) Statement in Chapter 1 of the ESBWR Design Control Document (DCD)

N. Clarification of ASME Component Design ~~Component Design~~ Inspections, Tests, Analyses, and Acceptance Criteria (ITAACs)

O. Corrections, Editorial, and Conforming Changes

V. Rulemaking Procedure

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

B. Incorporation by Reference of Public Documents

C. Changes to Tier 2* Information

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

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

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 website 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. The NRC did not receive any comment submissions after the August 31, 2011 practicality date. ← Define this term to distinguish it from the close of public comment. How is it arrived at?

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 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 ~~concludes~~ 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 result of this comment.

<u>Electrical Systems</u>

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 the design-basis, to 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

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 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 ~~which~~ reference the ESBWR also 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. 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 ~~which~~ reference the ESBWR also make any necessary design changes.

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 ~~to~~ provide ~~for~~ 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 ~~which~~ 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)

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~~s~~ 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.

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) provides 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 ~~which~~ 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 notice addresses the regulatory and policy issues that were addressed in the March 2011 proposed rule, the May 2014 supplemental proposed rule, and ~~thus~~ 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]

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

[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.

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 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 7 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

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, 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 ~~are~~ 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 ~~which~~ 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 both design certification rulemakings and in licensing.

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

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disclosure under 10 CFR 2.390(a)(7)(vi).

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

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 accidents 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 features – and not just ex-vessel severe accident design features – are considered.

design

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

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 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 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).

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 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, ~~foe~~ 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.

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 ~~of~~ the NRC staff's evaluation of the turbine building structure ~~was~~ ~~occurred~~ 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

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 ITAACs 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 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 ~~are~~ 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.

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 ~~to designate~~ 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 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.

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 ~~which~~ are not resolved by

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 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

[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.

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 ~~which~~ 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, ~~which~~ that were not specified in Tier 1 ~~that are~~ as acceptable for meeting ITAACs, were specified in Tier 2. The result of these departures is that certain significant information only exists 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 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 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 ~~that~~ 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.

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 ~~using~~ 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

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 ~~set~~ 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.

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 ~~that wishes to~~ 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;

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 ~~wishes~~ 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), 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 ~~which~~ 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~~, which~~ 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.