SECRETARY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 30, 2019

COMMISSION VOTING RECORD

DECISION ITEM: SECY-19-0020

TITLE: DIRECT FINAL RULE: ADVANCED POWER REACTOR 1400 DESIGN CERTIFICATION (RIN 3150-AJ67; NRC-2015-0224)

The Commission acted on the subject paper in an Affirmation Session as recorded in the Staff Requirements Memorandum (SRM) of April 30, 2019.

This Record contains a summary of voting on this matter together with the individual vote sheets, views and comments of the Commission.

Annette Vietti-Cook

Annette L. Vietti-Cook
Secretary of the Commission

Enclosures:
1. Voting Summary
2. Commissioner Vote Sheets

cc: Chairman Svinicki
Commissioner Baran
Commissioner Burns
Commissioner Caputo
Commissioner Wright
OGC
EDO
PDR

VOTING SUMMARY – SECY-19-0020

RECORDED VOTES

	APPROVED	DISAPPROVED	ABSTAIN	NOT PARTICIPATING	COMMENTS	DATE
Chrm. Svinicki	X				X	03/20/19
Cmr. Baran	X				X	03/11/19
Cmr. Burns	X				X	03/19/19
Cmr. Caputo	X				X	04/17/19
Cmr. Wright	X				X	03/20/19

AFFIRMATION ITEM

RESPONSE SHEET

TO: Annette Vietti-Cook, Secretary

FROM: CHAIRMAN SVINICKI

SUBJECT: SECY-19-0020: Direct Final Rule - Advanced Power Reactor 1400 Design Certification

Approved _XX_ Disapproved ____ Abstain ____ Not Participating ____

COMMENTS: Below _XX_ Attached _XX_ None ____

I approve the direct final rule and companion proposed rule for publication in the *Federal Register*, subject to the attached edits. I approve the use of the direct final rule process with regard to this action. In light of this process, I am confident that the General Counsel will review the Commission-directed changes to the text of the direct final rule and promptly advise the Commission of any issues, prior to affirmation of the direct final rule. I approve the environmental assessment, subject to the attached edits.

SIGNATURE

03/20/19

DATE

Entered on "STARS" Yes ✓ No ___

[7590-01-P]

KLS Edits

NUCLEAR REGULATORY COMMISSION

10 CFR Part 52

[NRC-2015-0224]

RIN 3150-AJ67

Advanced Power Reactor 1400 (APR1400) Design Certification

AGENCY: Nuclear Regulatory Commission.

ACTION: Direct final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to certify the Advanced Power Reactor 1400 (APR1400) standard ~~plant~~ design. ~~This action is necessary so that a~~Applicants or licensees intending to construct and operate an APR1400 standard ~~plant~~ design may do so by referencing this design certification (DC) rule. The applicant for the certification of the APR1400 standard ~~plant~~ design is Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. (KEPCO/KHNP).

DATES: The final rule is effective **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, unless significant adverse comments are received by **[INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. If the direct final rule is withdrawn as a result of such comments, timely notice of the withdrawal will be published in the *Federal Register*. The incorporation by reference of certain publications listed in this regulation is approved by

the Director of the Office of the Federal Register as of **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE FEDERAL REGISTER]**.

ADDRESSES: You may submit comments by any of the following methods (unless this document describes a different method for submitting comments on a specific subject):

- **Federal Rulemaking Web Site:** Go to https://www.regulations.gov/ and search for Docket ID NRC-2015-0224. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical questions contact the individuals listed in the FOR FURTHER INFORMATION CONTACT section of this document.
- **E-mail comments to:** Rulemaking.Comments@nrc.gov. If you do not receive an automatic e-mail reply confirming receipt, then contact us at 301-415-1677.
- **Fax comments to:** Secretary, U.S. Nuclear Regulatory Commission at 301-415-1101.
- **Mail comments to:** Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff.
- **Hand deliver comments to:** 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. (Eastern Time) Federal workdays; telephone: 301-415-1677.

For additional direction on obtaining information and submitting comments, see "Obtaining Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

FOR FURTHER INFORMATION CONTACT: Yanely Malave-Velez, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-1519; e-mail:

Yanely.Malave@nrc.gov, or William Ward, Office of New Reactors, telephone: 301-415-7038,; e-mail: William.Ward@nrc.gov. Both are staff of the U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

TABLE OF CONTENTS:

I. Obtaining Information and Submitting Comments.

A. Obtaining Information

Please refer to Docket ID NRC-2015-0224 when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- **Federal Rulemaking Web Site:** Go to https://www.regulations.gov and search for Docket ID NRC-2015-0224.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at https://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, at 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 the Availability of Documents section.

- **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.

B. Submitting Comments

Please include Docket ID NRC-2015-0224 in your comment submission.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at https://www.regulations.gov as well as enter the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment into ADAMS. Comments received after **[INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE FEDERAL REGISTER]** ~~this date~~ will be considered if it is practical to do so, but the NRC is able to ensure consideration only for comments received on or before this date. Comments received on this direct final rule will also be considered to be comments on a companion proposed rule published in the Proposed Rules section of this issue of the *Federal Register*.

II. Rulemaking Procedure.

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Because the NRC considers this action to be non-controversial, the NRC is using the "direct final rule procedure" for this rule. The rule will become effective on **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. However, if the NRC receives significant adverse comments by **[INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, then the NRC will publish a document that withdraws this direct final rule and would subsequently address the comments received in any final rule as a response to the companion proposed rule published in the Proposed Rules section of this issue of the *Federal Register*. Absent significant modifications to the proposed revisions requiring republication, the NRC does not intend to initiate a second comment period on this action.

A significant adverse comment is a comment ~~where~~ in which the commenter explains why the rule would be inappropriate~~, including challenges to the rule's underlying premise or approach, or would be ineffective or unacceptable without a change~~. A comment is adverse and significant if it meets the following criteria:

1) The comment opposes the rule and provides a reason sufficient to require a substantive response in a notice-and-comment process. For example, a substantive response is required when~~:~~

a) The comment causes the NRC to reevaluate (or reconsider) its position or conduct additional analysis;

b) The comment raises an issue serious enough to warrant a substantive response to clarify or complete the record; or

c) The comment raises a relevant issue that was not previously addressed or considered by the NRC.

2) The comment proposes a change or an addition to the rule, and it is apparent that the rule would be ineffective or unacceptable without incorporation of the change or addition.

3) The comment causes the NRC to make a change (other than editorial) to the rule.

For detailed instructions on filing comments, please see the ADDRESSES section in the companion proposed rule published in the Proposed Rules section of this issue of the *Federal Register*.

III. Background.

Part 52 of title 10 of the *Code of Federal Regulations* (10 CFR), "Licenses, Certifications, and Approvals for Nuclear Power Plants," subpart B, "Standard Design Certifications," presents the process for obtaining standard design certifications. On December 23, 2014, KEPCO/KHNP submitted its application for certification of the APR1400 standard ~~plant~~ design (ADAMS Accession No. ML15006A098) to the NRC ~~(ADAMS Accession No. ML15006A098)~~ under subpart B of 10 CFR part 52. The NRC published a notice of receipt of the application in the *Federal Register* (80 FR 5792; February 3, 2015). ~~KEPCO/KHNP submitted its application in accordance with Subpart B of 10 CFR part 52.~~ On March 12, 2015, the NRC formally accepted the application as a docketed application for design certification (80 FR 13035; March 12, 2015). The pre-application information submitted before the NRC formally accepted the application can be found in ADAMS under Docket No. PROJ0782.

IV. Discussion.

Final Safety Evaluation Report

The NRC issued ~~a~~the final safety evaluation report for the APR1400 design ~~i~~on September 28, 2018. The final safety evaluation report is available in ADAMS under Accession No. ML18087A364. The NRC will publish the final safety evaluation report as a NUREG titled, "Final Safety Evaluation Report Related to the Certification of the Advanced Power Reactor 1400 Standard Design." The final safety evaluation report is based on the NRC's review of revision 3 of the APR1400 design control document.

APR1400 DC Rule

The following discussion describes the purpose and key aspects of each section of the APR1400 DC rule. All section and paragraph references are to the provisions being added as appendix F to the regulations in 10 CFR part 52, unless otherwise noted. The NRC has modeled the APR1400 DC rule on existing DC rules, with certain modifications where necessary to account for differences in the APR1400 design documentation, design features, and environmental assessment (including severe accident mitigation design alternatives). As a result, DC rules are standardized to the extent practical.

A. Introduction (Section I)

The purpose of Section I of appendix F to 10 CFR part 52 is to identify the standard ~~plant~~ design ~~that would be~~ approved by this DC rule and the applicant for certification of the standard ~~plant~~ design. Identification of the design certification applicant is necessary to implement appendix F to 10 CFR part 52 for two reasons. First, ~~the implementation of~~ § 52.63(c) ~~depends on~~ identifies the design certification applicant as a potential source for~~whether~~ an applicant for a combined license (COL) ~~contracts with the design certification applicant~~ to obtain the generic design control document and supporting design information. If the COL applicant does not ~~use~~ obtain the design information from the design certification applicant, ~~to provide the design information and~~ but instead uses a~~n~~ different entity~~alternate nuclear plant vendor~~, then the COL applicant must meet the requirements in § 52.73. Second, paragraph X.A.1 of the rule ~~would~~ requires that the identified design certification applicant maintain the generic design control document throughout the time that appendix F to 10 CFR part 52 may be referenced.

B. Definitions (Section II)

The purpose of Section II of appendix F to 10 CFR part 52 is to define specific terminology with respect to this~~e~~ ~~design certification~~ DC rule. During development of the first two DC rules, the NRC decided that there would be both generic (master) design control documents maintained by the NRC and the design certification applicant, as well as individual plant-specific design control documents maintained by each applicant or licensee that references a ~~10 CFR part 52 appendix~~certified standard design. This distinction is necessary in order to specify the relevant plant-specific requirements to applicants and licensees referencing appendix F to 10 CFR part 52. In order to facilitate the maintenance of the master design control documents, the NRC requires that each

Commented [A1]: Clarified to use the term in the § 52.1 definition and to account for the existence of appendix N to part 52.

application for a standard design certification be updated to include an electronic copy of the final version of the design control document. The final version is required to incorporate all amendments to the design control document submitted since the original application, as well as any changes directed by the NRC as a result of its review of the original design control document or as a result of public comments. This final version is the master design control document incorporated by reference in the DC rule. The master design control document will be revised as needed to include generic changes to the version of the design control document that is approved in this design certification rulemaking. These changes would occur as the result of generic rulemaking by the NRC, under the change criteria in Section VIII.

The NRC also requires each applicant and licensee referencing appendix F to 10 CFR part 52 to submit and maintain a plant-specific design control document as part of the COL final safety analysis report. This plant-specific design control document must either include or incorporate by reference the information in the generic design control document. The plant-specific design control document would be updated as necessary to reflect the generic changes to the design control document that the NRC may adopt through rulemaking, plant-specific departures from the generic design control document that the NRC 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. Therefore, the plant-specific design control document functions similar to an updated final safety analysis report because it ~~would~~ provides the most complete and accurate information on a plant's design-basis for that part of the plant that would be within the scope of appendix F to 10 CFR part 52.

The NRC is treating the technical specifications in Chapter 16 of the generic design control document as a special category of information and designating them as generic technical specifications in order to facilitate the special treatment of this information under appendix F to 10 CFR part 52. A COL applicant must submit plant-specific technical specifications that consist of the generic technical specifications, which may be modified as specified in paragraph VIII.C, and the remaining site-specific information needed to complete the technical specifications. The final safety analysis report that is required by § 52.79 will consist of the plant-specific design control document, the site-specific final safety analysis report, and the plant-specific technical specifications.

The terms Tier 1, Tier 2, and COL items (license information) are defined in appendix F to 10 CFR part 52 because these concepts were not envisioned when 10 CFR part 52 was developed. The design certification applicants and the NRC use these terms in implementing the two-tiered rule structure (the DCD is divided into Tiers 1 and 2 to support the rule structure) that was proposed by representatives of the nuclear industry after publication of 10 CFR part 52. The Commission approved the use of a two-tiered rule structure in its staff requirements memorandum, dated February 15, 1991, on SECY-90-377, "Requirements for Design Certification under 10 CFR part 52," dated November 8, 1990 (ADAMS Accession No. ML003707892).

Tier 1 information means the portion of the design-related information contained in the generic DCD that is approved and certified by this appendix. Tier 2 information means the portion of the design-related information contained in the generic DCD that is approved but not certified by this appendix. The change process for Tier 2 information is similar to, but not identical to, the change process set forth in 10 CFR 50.59. The regulations in § 50.59 describe when a licensee may make changes to a plant as described in its final safety analysis report without a license amendment. Because the change process for Tier 2 information provided in Section VIII of this DC rule provides more specific criteria than § 50.59, as described in § 50.59(c)(4), the definitions and criteria of § 50.59 are not applicable to this process.~~Because of some differences in how~~

~~the change control requirements are structured in the DC rules, certain definitions contained in § 50.59 are not applicable to 10 CFR part 52 and are not being included in this direct final rule.~~ The NRC is including a definition for a "*Departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses*" (paragraph II.F~~G~~), which is appropriate to include in this direct final rule, so that the eight criteria in paragraph VIII.B.5.b will be implemented for new reactors~~,~~ as intended.

C. Scope and Contents (Section III)

The purpose of Section III of appendix F to 10 CFR part 52 is to describe and define the scope and content of this design certification, how to obtain a copy of the generic design control document, requirements for incorporation by reference of the DC rule, and ~~to set forth~~ how documentation discrepancies or inconsistencies are to be resolved.

Paragraph III.A is the required statement of the Office of the Federal Register for approval of the incorporation by reference of the APR1400 design control document, revision 3. In addition, this paragraph provides the information on how to obtain a copy of the design control document.

Paragraph III.B is the requirement for COL applicants and licensees referencing the APR1400 design control document to comply with the requirements of this appendix in order to benefit from the issue finality afforded the certified design. The legal effect of incorporation by reference is that the incorporated material has the same legal status as if it were published in the *Code of Federal Regulations*. This material, like any other properly-issued regulation, has the force and effect of law. Tier 1 and Tier 2 information (including the technical and topical reports referenced in Chapter 1)~~,~~ and generic technical specifications have been combined into a single document called the generic design control document~~,~~ in order to effectively control this information and facilitate its incorporation by reference into the rule. In addition, paragraph III.B clarifies that the conceptual design information and KEPCO/KHNP's evaluation of severe accident mitigation design alternatives are not considered to be part of appendix F to 10 CFR part 52. As provided by § 52.47(a)(24), these conceptual designs are not part of appendix F to 10 CFR part 52 and, therefore, are not applicable to an application that references appendix F to 10 CFR part 52. Therefore, ~~such~~ an applicant referencing appendix F to 10 CFR part 52 would not be required to conform to the conceptual design information that was provided by the design certification applicant. The conceptual design information, which consists of site-specific design features, was required to facilitate the design certification review. Similarly, the severe accident mitigation design alternatives were required to facilitate the environmental assessment.

Paragraphs III.C and III.D set forth the manner by which potential conflicts are to be resolved and identify the controlling document. Paragraph III.C establishes the Tier 1 description in the design control document as controlling in the event of an inconsistency between the Tier 1 and Tier 2 information in the design control document. Paragraph III.D establishes the generic design control document as the controlling document in the event of an inconsistency between the design control document and the final safety evaluation report for the certified standard design.

Paragraph III.E makes it clear that design activities entirely outside the scope of the design certification may be performed using actual site characteristics. This provision applies to site-specific portions of the plant, such as the administration building.

D. Additional Requirements and Restrictions (Section IV)

Section IV of appendix F to 10 CFR part 52 sets forth additional requirements and restrictions imposed upon an applicant who references appendix F to 10 CFR part 52.

Paragraph IV.A sets forth the information requirements for COL applicants and distinguishes between information and documents that must be *included* in the application or the design control document and those which may be *incorporated by reference*. Any incorporation by reference in the application should be clear and should specify the title, date, edition, or version of a document~~,~~ and the page number(s)~~,~~ and table(s) containing the relevant information to be incorporated. The legal effect of such an incorporation by reference into the application is that appendix F to 10 CFR part 52 would be legally binding on the applicant or licensee.

In paragraph IV.B the NRC reserves the right to determine how appendix F to 10 CFR part 52 may be referenced under 10 CFR part 50. This determination may occur in the context of a subsequent rulemaking modifying 10 CFR part 52 or this DC rule~~,~~ or on a case-by-case basis in the context of a specific application for a 10 CFR part 50 construction permit or operating license. This provision is necessary because the previous DC rules were not implemented in the manner that was originally envisioned at the time that 10 CFR part 52 was issued. The NRC's concern is with the manner by which the inspections, tests, analyses, and acceptance criteria (ITAAC) were developed and the lack of experience with design certifications in a licensing proceeding. Therefore, it is appropriate that the NRC retain some discretion regarding the manner by which appendix F to 10 CFR part 52 could be referenced in a 10 CFR part 50 licensing proceeding.

E. Applicable Regulations (Section V)

The purpose of Section V of appendix F to 10 CFR part 52 is to specify the regulations that were applicable and in effect at the time this design certification was approved. These regulations consist of the technically relevant regulations identified in paragraph V.A, except for the regulations in paragraph V.B that would not be applicable to this certified design.

F. Issue Resolution (Section VI)

The purpose of Section VI of appendix F to 10 CFR part 52 is to identify the scope of issues that ~~would be~~ are resolved by the NRC through this rulemaking and, therefore, are "matters resolved" within the meaning and intent of § 52.63(a)(5). The section is divided into five parts: paragraph VI.A identifies the NRC's safety findings in adopting appendix F to 10 CFR part 52, paragraph VI.B identifies the scope and nature of issues that ~~would be~~ are resolved by this rulemaking, paragraph VI.C identifies issues~~,~~ that ~~which~~ are not resolved by this rulemaking, ~~and~~ paragraph VI.D identifies the issue finality restrictions applicable to the NRC with respect to appendix F to 10 CFR part 52, and paragraph VI.E identifies the availability of secondary resources.

Paragraph VI.A describes the nature of the NRC's findings in general terms and makes the findings required by § 52.54 for the NRC's approval of this DC rule.

Paragraph VI.B sets forth the scope of issues that may not be challenged as a matter of right in subsequent proceedings. The introductory phrase of paragraph VI.B clarifies that issue resolution, as described in the remainder of the paragraph, extends to the delineated NRC proceedings for plants referencing appendix F to 10 CFR part 52. The remainder of paragraph VI.B describes the categories of information for which there is issue resolution.

Paragraph VI.C reserves the right of the NRC to impose operational requirements on applicants that reference appendix F to 10 CFR part 52. This provision reflects the fact that only some operational requirements, including portions of the generic technical specifications in Chapter 16 of the design control document, and no operational programs (e.g., operational quality assurance), were completely ~~or comprehensively~~ reviewed by the NRC in this design certification rulemaking proceeding. Therefore, the issue finality provisions of § 52.63 apply only to those operational requirements that either the NRC completely reviewed and approved~~,~~ or formed the basis of an NRC safety finding of the adequacy of the APR1400, as documented in the NRC's final safety evaluation report. 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 final safety analysis report.[1] The NRC's choice of the regulatory vehicle for imposing the operational requirements will depend upon the following, 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 § 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.

Also, paragraph VI.C allows the NRC to impose future operational requirements (distinct from design matters) on applicants who reference this design certification. 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 § 52.63. The

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

requirement to perform these testing programs is contained in the Tier 1 information. However, ITAAC 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 ITAAC are satisfied. In the absence of detailed design information to evaluate the need for and develop specific post-fuel load verifications for these matters, the NRC is reserving the right to impose, at the time of COL issuance, license conditions addressing post-fuel load verification activities for portions of the plant within the scope of this design certification.

Paragraph VI.D requires the NRC to follow~~reiterates~~ the restrictions ~~(~~contained in Section VIII~~)~~ ~~placed upon the NRC~~ when ~~ordering~~ requiring generic or plant-specific modifications, changes, or additions to structures, systems, and components~~,~~; design features~~,~~; design criteria~~,~~; and ITAAC within the scope of the certified design.

Paragraph VI.E ensures that the NRC will specify at an appropriate time the procedures on how to obtain access to sensitive unclassified and non-safeguards information (SUNSI) and safeguards information (SGI) for the APR1400 DC rule. Access to such information would be for the sole purpose of requesting or participating in certain specified hearings, such as hearings required by § 52.85 or an adjudicatory hearing. For proceedings where the notice of hearing was published before the effective date of the final rule, the Commission's order governing access to SUNSI and SGI shall be used to govern access to such information within the scope of the rulemaking. For proceedings in which the notice of hearing or opportunity for hearing is published after the effective date of the final rule, paragraph VI.E applies and governs access to SUNSI and SGI.

G. Duration of this Appendix (Section VII)

The purpose of Section VII of appendix F to 10 CFR part 52 is, in part, to specify the period during which this design certification may be referenced by an applicant for a COL, under § 52.55, and the period it will remain valid when the design certification is referenced. For example, if an application references this design certification during the 15-year period, then the design certification would be effective for that application until it ~~the application~~ is withdrawn or the license issued on that application expires, including periods of operation under a renewed license. The NRC intends for appendix F to 10 CFR part 52 to remain valid for the life of the plants that reference~~s~~ the design certification to achieve the benefits of standardization and licensing stability. This means that changes to, or plant-specific departures from, information in the plant-specific design control document must be made under the change processes in Section VIII for the life of ~~the~~ a plant that references this DC rule.

H. Processes for Changes and Departures (Section VIII)

The purpose of Section VIII of appendix F to 10 CFR part 52 is to set forth the processes for generic changes to, or plant-specific departures (including exemptions) from~~,~~ the design control document. The NRC adopted this restrictive change process in order to achieve a more stable licensing process for applicants and licensees that reference DC rules. Section VIII is divided into three paragraphs, which correspond to Tier 1, Tier 2, and operational requirements.

Generic *changes* (called "modifications" in § 52.63(a)(3)) must be accomplished by rulemaking because the intended subject of the change is this DC rule itself, as is contemplated by § 52.63(a)(1). Consistent with § 52.63(a)(3), any generic rulemaking changes are applicable to all plants referencing this DC rule, absent circumstances which render the change technically irrelevant. By contrast, plant-specific *departures* could be either an order to one or more applicants or licensees; or an applicant or licensee-initiated departure applicable only to that applicant's or licensee's plant(s), similar to a § 50.59 departure or an exemption. Because these plant-specific departures will result in a design control document that is unique for that plant, Section X would require an applicant or licensee to maintain a plant-specific design control document. For purposes of brevity, the following discussion refers to the processes for both generic changes and plant-specific departures as "change processes." Section VIII refers to an exemption from one or more requirements of this appendix and addresses the criteria for granting an exemption. The NRC cautions that when the exemption involves an underlying substantive requirement (i.e., a requirement outside this appendix), then the applicant or licensee requesting the exemption must demonstrate that an exemption from the underlying applicable requirement meets the criteria of § 52.7 ~~and~~ or § 50.12.

For the APR1400 DC review, the staff followed the approach described in SECY-17-0075, "Planned Improvements in Design Certification Tiered Information Designations," (ADAMS Accession No. ML16196A321), to evaluate the applicant's designation of information as Tier 1 or Tier 2 information. Unlike prior design certification applications, this application did not contain any Tier 2* information. As described in SECY-17-0075, in each of the prior design certification rules in appendices A through D to 10 CFR ~~P~~part 52, ~~Appendices A through D,~~ information contained in the DCD was divided into three designations: Tier 1, Tier 2, and Tier 2*. Tier 1 information is the portion of design-related information in the generic DCD that the Commission approves in the ~~P~~part 52 design certification rule appendices. To change Tier 1 information, NRC approval by rulemaking or approval of an exemption from the certified design rule is required. Tier 2 information is also approved by the Commission in the ~~P~~part 52 design certification rule appendices, but it is not certified and licensees who reference the design can change this information using the process outlined in Section

VIII of the appendices. This change process is similar to that in 10 CFR 50.59 and is generally referred to as the "50.59-like" process. If the criteria in Section VIII are met, a licensee can change Tier 2 information without prior NRC approval. The NRC created a third category, Tier 2*, to address industry requests to minimize the scope of Tier 1 information and provide greater flexibility for making changes. Tier 2* information is included in Tier 2 ~~and~~ although it has the same safety significance as Tier 1 information, but the NRC decided to provide more flexibility for licensees to change this type of information. In prior design certification rules, Tier 2* is significant information included only in Tier 2 that cannot be changed without prior NRC approval of a license amendment requesting the change.

The applicant included Tier 1 and Tier 2 information in the APR1400 DC application and did not designate or categorize any information as Tier 2* information. Generally, where an applicant includes only Tier 1 and Tier 2 information in an application, the staff will evaluate the Tier 2 information to determine whether any of that information requires NRC approval before it is changed. If the staff identifies any such information in Tier 2, then the staff will request that the applicant revise the application to categorize that information as Tier 1 or Tier 2*, depending on whether the change must be made by approval of a license amendment and an exemption requesting the change (Tier 1)~~,~~ or a license amendment alone (Tier 2*). Because the applicant did not designate any information as Tier 2* information, the staff also considered whether the applicant had included information in Tier 2 that prior DC applicants had identified as Tier 2* but that the NRC staff determined should be categorized as Tier 1. Using requests for additional information, the staff questioned KEPCO/KHNP's categorization of certain information as Tier 2 that past DC applicants had identified as Tier 2* and, in some instances, the staff requested that the applicant revise the application to add that information to Tier 1. This approach required staff and KEPCO/KHNP to identify for each request for additional information the verifiable, important to safety parameters ~~which~~ that must be included in Tier 1 to be certified in the rule and verified by ITAAC. After several public meetings, some information was added to or updated in Tier 1 (including modifications to some ITAAC) and the requests for additional information were resolved and closed without the designation of any Tier 2* information.

Of these updates in Tier 1, the most significant concerned the design parameters for the critical structural sections[2] for seismic Category I structures. Past DC applications identified dimensions of length to define critical structural sections as Tier 2* information. During recent construction activities for another design, actual dimensional lengths were found to be outside of their design tolerances. This variance did not necessarily reduce safety but did require additional license amendments to resolve the issue associated with the design tolerances, resulting in increased costs and possible construction schedule impacts. For the APR1400 design, the ~~resolution was to revise~~ Tier 1 information and the ITAAC for these critical structural sections ~~to~~ used the design load and design load capacity in lieu of dimensions of length, as specific dimensions are not necessarily as important to safety. By focusing on important to safety parameters and including them in ITAAC, ~~rather than in Tier 2* information (thus eliminating the need for Tier 2* information),~~ the staff expects that the need for license amendments to address changes during construction will be greatly reduced while still maintaining reasonable assurance of adequate protection.

[2] When evaluating the acceptability of the information for seismic Category I structures, the staff's review focuses on a subset of structural information that includes seismic analysis methods, key parameters of seismic Category I structures, and the design of "critical sections." The use of critical sections in the design of safety-related structures is a risk-informed graded approach to achieve the reasonable assurance of safety. In lieu of the safety review of a large number of structural component designs, the staff performs a detailed review of a limited number of critical sections described in the design control document, Section 3.8, that contribute to the overall risk significance of the structures. This approach provides the staff with reasonable assurance of the overall safety performance of the structures based on the successful performance of these limited, but critical, risk-significant locations. However, even minor changes to these critical sections could, when applied to the entire safety-related structure, result in significant changes to the overall performance of the structure and, therefore, invalidate the basis for the staff's approval.

Tier 1 information

Paragraph A describes the change process for changes to Tier 1 information that are accomplished by rulemakings that amend the generic design control document and are governed by the standards in § 52.63(a)(1). A generic change under § 52.63(a)(1) will not be made to a certified design while it is in effect unless the change: 1) is necessary for compliance with NRC regulations applicable and in effect at the time the certification was issued; 2) is necessary to provide adequate protection of the public health and safety or the common defense and security; 3) reduces unnecessary regulatory burden and maintains protection to public health and safety and common defense and security; 4) provides the detailed design information necessary to resolve select design acceptance criteria; 5) is necessary to correct~~s~~ material errors in the certification information; 6) substantially increases overall safety, reliability, or security of a facility and the costs of the change are justified; or 7) contributes to increased standardization of the certification information. The rulemakings must provide for notice and opportunity for public comment on the proposed change, as required by § 52.63(a)(2). The NRC will give consideration as to whether the benefits justify the costs for plants that are already licensed or for which an application for a permit or license is under consideration except for those changes that are necessary to provide adequate protection of the public health and safety or the common defense and security.

Departures from Tier 1 may occur in two ways: 1) the NRC may order a licensee to depart from Tier 1, as provided in paragraph A.3~~;~~ or 2) an applicant or licensee may request an exemption from Tier 1, as addressed in paragraph A.4. If the NRC seeks to order a licensee to depart from Tier 1, paragraph A.3 would require that the NRC find both that the departure is necessary ~~for~~ either to assure adequate protection of the public health and safety or the common defense and security or ~~for~~to secure compliance with the NRC's regulations applicable and in effect at the time of approval of the design certification and that special circumstances are present. Paragraph A.4 ~~would~~ provides that exemptions from Tier 1 requested by an applicant or licensee are governed by the requirements of §§ 52.63(b)(1) and 52.98(f), which provides an opportunity for a hearing. In addition, the NRC would not grant requests for exemptions that ~~may~~ will result in a significant decrease in the level of safety otherwise provided by the design.

Tier 2 information

Paragraph B describes the change processes for the Tier 2 information; which have the same elements as the Tier 1 change process, but some of the standards for plant-specific orders and exemptions would be different. Generic Tier 2 changes would be accomplished by rulemaking that would amend the generic design control document and would be governed by the standards in § 52.63(a)(1). A generic change under § 52.63(a)(1) would not be made to a certified design while it is in effect unless the change: 1) is necessary for compliance with NRC regulations that were applicable and in effect at the time the certification was issued; 2) is necessary to provide adequate protection of the public health and safety or the common defense and security; 3) reduces unnecessary regulatory burden and maintains protection to public health and safety and the common defense and security; 4) provides the detailed design information necessary to resolve select design acceptance criteria; 5) is necessary to correct~~s~~ material errors in the certification information; 6) substantially increases overall safety, reliability, or security of a facility and the costs of the change are justified; or 7) contributes to increased standardization of the certification information.

Departures from Tier 2 would occur in four ways: 1) the NRC may order a plant-specific departure, as set forth in paragraph B.3; 2) an applicant or licensee may request an exemption from a Tier 2 requirement as set forth in paragraph B.4; 3) a licensee may make a departure without prior NRC approval under paragraph B.5; or 4) the licensee may request NRC approval for proposed departures which do not meet the requirements in paragraph B.5 as provided in paragraph B.5.e.

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

An applicant or licensee referencing this DC rule is ~~would be~~ permitted to request an exemption from Tier 2 information as set forth in paragraph B.4. The applicant or licensee would have to demonstrate that the exemption complies with one of the special circumstances in regulations governing specific exemptions in § 50.12(a). In addition, the NRC would not grant requests for exemptions that ~~may~~ will result in a significant decrease in the level of safety otherwise provided by the design. However, unlike Tier 1 changes, the special circumstances for the exemption do not have to outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. If the exemption is requested by an applicant for a license, the exemption would be subject to litigation in the same manner as other issues in the licensing hearing, consistent with § 52.63(b)(1). If the exemption is requested by a licensee, then the exemption would be subject to ~~litigation~~ an opportunity for hearing in the same manner as ~~a~~ license amendments.

Paragraph B.5 would allow an applicant or licensee to depart from Tier 2 information, without prior NRC approval, if ~~it~~ the departure does not involve a change to~~,~~ or departure from~~,~~ Tier 1 information~~,~~, or the technical specifications, ~~or~~ and the departure does not require a license amendment under paragraphs B.5.b or c. The technical specifications referred to in B.5.a of this paragraph are the technical specifications in Chapter 16 of the generic design control document, including bases, for departures made prior to the issuance of the COL. After the issuance of the COL, the plant-specific technical specifications would be controlling under paragraph B.5. The requirement for a license amendment in paragraph B.5.b would be similar to the requirement in § 50.59 and would apply to all of the information in Tier 2 except for the information that resolves the severe accident issues.

Paragraph B.5.b addresses information described in the design control document to address aircraft impacts, in accordance with § 52.47(a)(28). Under § 52.47(a)(28), applicants are required to include the information required by § 50.150(b) in their design control document. 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 aircraft impact assessment required by § 50.150(a). The applicant or licensee is also required to describe in the plant-specific design control document how the modified design features and functional capabilities continue to meet the assessment requirements in § 50.150(a)(1). Submittal of this updated information is governed by the reporting requirements in Section X.B.

During an ongoing adjudicatory proceeding (e.g., for issuance of a COL) a party who believes that an applicant or licensee has not complied with paragraph B.5 when departing from Tier 2 information may petition to admit such a contention into the proceeding under paragraph B.5.g. As set forth in paragraph B.5.g, the petition would have to comply with the requirements of § 2.309 and show that the departure does not comply with paragraph B.5. If on the basis of the petition and any responses thereto, the presiding officer in the proceeding determines that the required showing has been made, the matter would be certified to the Commission for its final determination. In the absence of a proceeding, assertions of ~~nonconformance~~ noncompliance with paragraph B.5 requirements applicable to Tier 2 departures would be treated as petitions for enforcement action under § 2.206.

Operational Requirements

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

The process in paragraph C.1 for making generic changes to the generic technical specifications in Chapter 16 of the design control document or other operational requirements in the generic design control document ~~would be~~ is accomplished by rulemaking and governed by the backfit standards in § 50.109. The determination of whether the generic technical specifications and other operational requirements were completely reviewed and approved in the design certification rulemaking ~~would be~~ is based upon the extent to which the NRC reached a safety conclusion in the final safety evaluation report on this matter. If a technical specification or operational requirement was completely reviewed and finalized in the design certification rulemaking, then the requirement of § 50.109 would apply because a position was taken on that safety matter. Generic changes made under paragraph VIII.C.1 would be applicable to all applicants or licensees referencing this DC rule ~~(refer to~~ as described in paragraph C.2~~)~~, unless the change is made technically irrelevant ~~because of~~ by a plant-specific departure.

Some generic technical specifications 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 APR1400 DC rule must replace the values in brackets with final plant-specific values (refer to guidance provided in Regulatory Guide 1.206, Revision 1, "Applications for Nuclear Power Plants"). The values in brackets are neither part of the DC rule nor are they binding. Therefore, the replacement of bracketed values with final plant-specific values does not require an exemption from the generic technical specifications.

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

Paragraph C.6 states that the generic technical specifications would have no further effect on the plant-specific technical specifications after the issuance of a license that references this appendix ~~and the change process~~. After a license is issued, the bases for the plant-specific TS would be controlled by the bases change provision set forth in the administrative controls section of the plant-specific technical specifications.

I. [RESERVED] (Section IX)

This section is reserved for future use. The matters discussed in this section of earlier design certification rules – inspections, tests, analyses, and acceptance criteria – are now addressed in the substantive provisions of 10 CFR part 52. Accordingly, there is no need to repeat these regulatory provisions in the APR1400 design certification rule.

However, this section is being reserved to maintain consistent section numbering with other design certification rules.

J. Records and Reporting (Section X)

The purpose of Section X of appendix F to 10 CFR part 52 is to set forth the requirements that will apply to maintaining records of changes to and departures from the generic design control document, which are to be reflected in the plant-specific design control document. Section X also sets forth the requirements for submitting reports (including updates to the plant-specific design control document) to the NRC. This section of appendix F to 10 CFR part 52 is similar to the requirements for records and reports in 10 CFR part 50, except for minor differences in information collection and reporting requirements.

Paragraph X.A.1 requires that a generic design control document including SUNSI and SGI referenced in the generic design control document be maintained by the applicant for this rule. The generic design control document concept was developed, in part, to meet the requirements for incorporation by reference, including public availability of documents incorporated by reference. However, the SUNSI and SGI could not be included in the generic design control document because they are not publicly available. Nonetheless, the SUNSI and SGI were reviewed by the NRC and, as stated in paragraph VI.B.2, the NRC would consider the information to be resolved within the meaning of § 52.63(a)(5). Because this information is not in the generic design control document, this information, or its equivalent, is required to be provided by an applicant for a license referencing this DC rule. Only the generic design control document is identified and incorporated by reference into this rule. The generic design control document and the NRC-approved version of the SUNSI and SGI must be maintained by the applicant (KEPCO/KHNP) for the period of time that appendix F to 10 CFR part 52 may be referenced.

Paragraphs X.A.2 and X.A.3 place recordkeeping requirements on ~~the~~ an applicant or licensee that reference this design certification so that its plant-specific design control document accurately reflect both generic changes to the generic design control document and plant-specific departures made under Section VIII. The term "plant-specific" is used in paragraph X.A.2 and other sections of appendix F to 10 CFR part 52 to distinguish between the generic design control document that would be incorporated by reference into appendix F to 10 CFR part 52, and the plant-specific design control document that the COL applicant is required to submit under paragraph IV.A. The requirement to maintain changes to the generic design control document is explicitly stated in order to ensure that these changes are not only reflected in the generic design control document, which will be maintained by the applicant for the design certification, but also in the plant-specific design control document. Therefore, records of generic changes to the design control document will be required to be maintained by both entities to ensure that both entities have up-to-date design control documents.

Paragraph X.A.4.a requires the DC rule applicant to maintain a copy of the aircraft impact assessment analysis for the term of the certification and any renewal. This provision, which is consistent with § 50.150(c)(3), would 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 appendix F to 10 CFR part 52 to maintain a copy of the aircraft impact assessment performed to comply with the requirements of § 50.150(a) throughout the pendency of the application and for the term of the license and any renewal. This provision is consistent with § 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 § 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 final safety analysis report required by § 52.79 will contain the plant-specific design control document 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 final safety analysis report for a facility (required by § 52.79)~~,~~ but is not part of the plant-specific design control document (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 design control document is part of the final safety analysis report for the facility.

Paragraph X.B.1 requires applicants or licensees that reference this rule to submit reports that describe departures from the design control document and include a summary of the written evaluations. The requirement for the written evaluations is set forth in paragraph X.A.~~1~~3. The frequency of the report submittals is set forth in paragraph X.B.3. The requirement for submitting a summary of the evaluations ~~will be~~ is similar to the requirement in § 50.59(d)(2).

Paragraph X.B.2 requires applicants or licensees that reference this rule to submit updates to the design control document, which include both generic changes and plant-specific departures, as set forth in paragraph X.B.3. The requirements in paragraph X.B.3 for submitting reports will vary according to certain time periods during a facility's lifetime. If a potential applicant for a COL that references this rule decides to depart from the generic design control document prior to submission of the application, then paragraph X.B.3.a will require that the updated design control document 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 design control document 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 support planning for 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 ITAAC under § 52.103(g), as well as any finding on interim operation under Section 189.a(1)(B)(iii) of the Atomic Energy Act of 1954, as amended. Once a facility begins operation (for a COL under 10 CFR part 52, after the Commission has made a finding under § 52.103(g)), the frequency of reporting will be governed by the requirements in paragraph X.B.3.c.

V. APR1400 Standard Design Approval.

On March 8, 2018, as part of the submission of revision 2 of the design control document (ADAMS Accession No. ML18079A146), KEPCO/KHNP requested the NRC provide a final design approval for the APR1400 design. On August 13, 2018, as part of the submission of revision 3 of the design control document (ADAMS Accession No. ML18228A680), KEPCO/KHNP corrected their request for a final design approval to a request for a standard design approval. A standard design approval for the APR1400, revision 3, was issued on September 28, 2018 (ADAMS Accession No. ML18261A187) following the NRC's issuance of the APR1400 final safety evaluation report.

The finality of ~~the~~ standard design approvals is discussed in § 52.145. The standard design approval is valid for 15 years from the date of issuance, as described in § 52.147.

VI. Section-by-Section Analysis.

The following paragraphs describe the specific changes in this direct final rule:

Section 52.11, Information collection requirements: OMB approval.

In § 52.11, this direct final rule adds new appendix F to 10 CFR part 52 to the list of information collection requirements in paragraph (b) of this section.

Appendix F to Part 52—Design Certification Rule for the APR1400 Design

This direct final rule adds appendix F to 10 CFR part 52 to incorporate the APR1400 standard ~~plant~~ design into the NRC's regulations. Applicants or licensees intending to construct and operate a plant using an APR1400 design may do so by referencing the DC rule.

VII. Regulatory Flexibility Certification.

Under the Regulatory Flexibility Act (5 U.S.C. 605(b)), the NRC certifies that this direct final rule does not have a significant economic impact on a substantial number of small entities. This direct final rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810).

VIII. Regulatory Analysis.

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

IX. Backfitting and Issue Finality.

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

This initial DC rule does not constitute backfitting as defined in the backfit rule (10 CFR 50.109) because there are no existing operating licenses under 10 CFR part 50, COLs or manufacturing licenses under 10 CFR part 52 referencing this DC rule and because this DC rule does not modify the standard design approval for the APR 1400.

This initial DC rule 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 DC rules in appendices A through E to 10 CFR part 52 or the standard design approval for APR1400, and no COLs or manufacturing licenses issued by the NRC at this time reference a final APR1400 DC rule.

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.

X. Voluntary Consensus Standards.

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this direct final rule, the NRC certifies the APR1400 standard ~~plant~~ design for use in nuclear power plant licensing under 10 CFR parts 50 or 52. Design certifications are not generic rulemakings establishing a generally applicable standard with which all 10 CFR parts 50 and 52 nuclear power plant licensees must comply. Design certifications are Commission approvals of specific nuclear power plant designs by rulemaking.

Furthermore, design certifications are initiated by an applicant for rulemaking, rather than by the NRC. This action does not constitute the establishment of a standard that contains generally applicable requirements.

XI. Plain Writing.

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

XII. Environmental Assessment and Final Finding of No Significant Environmental Impact.

The NRC ~~conducted an environmental assessment (ADAMS Accession No. ML18306A607) and~~ has determined under the National Environmental Policy Act of 1969, as amended (NEPA), and the NRC's regulations in subpart A of 10 CFR part 51, that this direct final rule, if confirmed, would not be a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement 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 DC rule does not authorize the siting, construction, or operation of a facility referencing any particular design; it only codifies the APR1400 design in a rule. The NRC will evaluate the environmental impacts and issue an environmental impact statement as appropriate under NEPA as part of the application for the construction and operation of a facility referencing any particular DC rule.

In addition, consistent with 10 CFR 51.30(d) and 10 CFR 51.32(b), the NRC has prepared a final environmental assessment (ADAMS Accession No. ML18306A607) for the APR1400 design addressing various design alternatives to prevent and mitigate severe accidents. The environmental assessment is based, in part, upon the NRC's review of KEPCO/KHNP's evaluation of various design alternatives to prevent and mitigate severe accidents in APR1400-E-P-NR-14006, Revision 2, "Severe Accident Mitigation Design Alternatives (SAMDAs) for the APR1400" (ML18235A158). Based upon review of KEPCO/KHNP's evaluation, the Commission concludes that: (1) KEPCO/KHNP identified a reasonably complete set of potential design alternatives to prevent and mitigate severe accidents for the APR1400 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 APR1400 are very low on an absolute scale. These issues are considered resolved for the APR1400 design. Based on its own independent evaluation, the NRC reached the same conclusion as KEPCO/KHNP that none of the possible candidate design alternatives are potentially cost beneficial for the APR1400 design. This independent evaluation was based on reasonable treatment of costs, benefits, and sensitivities. The NRC concludes that KEPCO/KHNP has adequately identified areas where risk potentially could be reduced in a cost-beneficial manner and adequately assessed whether the implementation of the identified potential severe accident mitigation design alternatives or candidate design alternatives would be cost-beneficial for the given site parameters. Therefore, the NRC finds that the evaluation performed by KEPCO/KHNP is reasonable and sufficient.

The determination of this environmental assessment is that there will be no significant offsite impact to the public from this action. The environmental assessment is available as indicated under Section XVI, "Availability of Documents."

XIII. Paperwork Reduction Act Statement.

The burden to the public for the information collection(s) is estimated to average 37 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Further information about information collection requirements associated with this direct final rule can be found in the companion proposed rule published in the Proposed Rule section in this issue of the *Federal Register*.

This direct final rule is being issued prior to approval by the Office of Management and Budget (OMB) of these information collection requirements, which were submitted under OMB control number 3150-0151. When OMB notifies the NRC of its decision, the NRC will publish a document in the *Federal Register* providing notice of the effective date of the information collections or, if approval is denied, providing notice of what action we plan to take.

Send comments on any aspect of these information collections, including suggestions for reducing the burden, to the Information Services Branch, U.S. Nuclear Regulatory Commission, Washington, District of Columbia 20555-0001, or by email to INFOCOLLECTS.RESOURCE@NRC.GOV; and to OMB Office of Information and Regulatory Affairs (3150-0151), Attn: Desk Officer for the Nuclear Regulatory

Commission, 725 17th Street, NW Washington, District of Columbia 20503; e-mail: oira_submission@omb.eop.gov.

Public Protection Notification.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

XIV. Congressional Review Act.

This final rule is a rule as defined in the Congressional Review Act (5 U.S.C. 801-808). However, the Office of Management and Budget has not found it to be a major rule as defined in the Congressional Review Act.

XV. Agreement State Compatibility.

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

XVI. Availability of Documents.

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

Documents Related to APR1400 Design Certification Rule

DOCUMENT	ADAMS ACCESSION NO. / WEB LINK / *FEDERAL REGISTER* CITATION
SECY-19~~XX~~-0020~~XXXX~~, "Direct Final Rule – APR1400 Design Certification"	ML18302A069
KEPCO/KHNP Application for Design Certification of the APR1400 Design	ML15006A037
APR1400 Design Control Document, Revision 3	ML18228A667
APR1400 Final Safety Evaluation Report	ML18087A364
APR1400 Environmental Assessment	ML18306A607
APR1400 Standard Design Approval	ML18261A187
Regulatory History of Design Certification[3]	ML003761550
KHNP Topical and Technical Reports	
APR1400-F-A-TR-12004-NP-A, Realistic Evaluation Methodology for Large-Break LOCA of the APR1400, Rev. 1 (August 2018)	ML18233A431
APR1400-F-C-TR-12002-NP-A, KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design, Rev. 0 (April 2017)	ML17115A559
APR1400-F-M-TR-13001-NP-A, PLUS7 Fuel Design for the APR1400, Rev. 1 (August 2018)	ML18232A140
APR1400-K-Q-TR-11005-NP-A, KHNP Quality Assurance Program Description (QAPD) for the APR1400 Design Certification, Rev. 2 (October 2016)	ML18085B044
APR1400-Z-M-TR-12003-NP-A, Fluidic Device Design for the APR1400, Rev. 0 (April 2017)	ML17129A597
APR1400-E-I-NR-14001-NP, Human Factors Engineering Program Plan, Rev. 4 (July 2018)	ML18212A345

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

APR1400-E-I-NR-14002-NP, Operating Experience Review Implementation Plan, Rev. 2 (January 2018)	ML18081A101
APR1400-E-I-NR-14003-NP, Functional Requirements Analysis and Function Allocation Implementation Plan, Rev. 2 (January 2018)	ML18081A091
APR1400-E-I-NR-14004-NP, Task Analysis Implementation Plan, Rev. 3 (May 2018)	ML18178A223
APR1400-E-I-NR-14006-NP, Treatment of Important Human Actions Implementation Plan, Rev. 3 (May 2018)	ML18178A224
APR1400-E-I-NR-14007-NP, Human-System Interface Design Implementation Plan, Rev. 3 (May 2018)	ML18178A212
APR1400-E-I-NR-14008-NP, Human Factors Verification and Validation Implementation Plan, Rev. 3 (May 2018)	ML18178A213
APR1400-E-I-NR-14010-NP, Human Factors Verification and Validation Scenarios, Rev. 2 (January 2018)	ML18081A088
APR1400-E-I-NR-14011-NP, Basic Human-System Interface, Rev. 3 (May 2018)	ML18178A214
APR1400-E-I-NR-14012-NP, Style Guide, Rev. 2 (January 2018)	ML18081A096
APR1400-E-N-NR-14001-NP, Design Features To Address GSI-191, Rev. 3 (February 2018)	ML18057B532
APR1400-E-P-NR-14005-NP, Evaluations and Design Enhancements To Incorporate Lessons Learned from Fukushima Dai-Ichi Nuclear Accident, Rev. 2 (July 2017)	ML18044B042
APR1400-E-S-NR-14004-NP, Evaluation of Effects of HRHF Response Spectra on SSCs, Rev. 3 (December 2017)	ML18078A709
APR1400-E-S-NR-14005-NP, Evaluation of Structure-Soil-Structure Interaction (SSSI) Effects, Rev. 2 (December 2017)	ML18078A699
APR1400-E-S-NR-14006-NP, Stability Check for NI Common Basemat, Rev. 5 (May 2018)	ML18178A221
APR1400-F-A-NR-14001-NP, Small Break LOCA Evaluation Model, Rev. 1 (March 2017)	ML17114A524
APR1400-F-A-NR-14003-NP, Post-LOCA Long Term Cooling Evaluation Model, Rev. 1 (March 2017)	ML17114A526
APR1400-H-N-NR-14012-NP, Mechanical Analysis for New and Spent Fuel Storage Racks, Rev. 3 (August 2017)	ML17244A015
APR1400-K-I-NR-14005-NP, Staffing and Qualifications Implementation Plan, Rev. 1 (February 2018)	ML17094A152
APR1400-K-I-NR-14009-NP, Design Implementation Plan, Rev. 1 (February 2017)	ML17094A153
APR1400-Z-A-NR-14006-NP, Non-LOCA Safety Analysis Methodology, Rev. 1 (February 2017)	ML17094A139

APR1400-Z-A-NR-14007-NP, Mass and Energy Release Methodologies for LOCA and MSLB, Rev. 2 (May 2018)	ML18212A338
APR1400-Z-A-NR-14011-NP, Criticality Analysis of New and Spent Fuel Storage Racks, Rev. 3 (May 2018)	ML18214A561
APR1400-A-N-NR-17001-NP (WCAP-17889-P), Validation of SCALE 6.1.2 with 238-Group ENDF/B-VII.0 Cross Section Library for APR1400 Design Certification, Rev. 0 (June 2014)	ML18044B051
APR1400-Z-J-NR-14001-NP, Safety I&C System, Rev. 3 (May 2018)	ML18212A341
APR1400-Z-J-NR-14003-NP, Software Program Manual, Rev. 3 (May 2018)	ML18214A559
APR1400-E-J-NR-17001-NP, Secure Development and Operational Environment for APR1400 Computer-Based I&C Safety Systems, Rev. 0 (September 2017)	ML18108A470
APR1400-Z-J-NR-14004-NP, Uncertainty Methodology and Application for Instrumentation, Rev. 2 (January 2018)	ML18086B757
APR1400-Z-J-NR-14005-NP, Setpoint Methodology for Safety-Related Instrumentation, Rev. 2 (January 2018)	ML18087A106
APR1400-E-J-NR-14001-NP, Component Interface Module, Rev. 1 (March 2017)	ML17094A131
APR1400-F-C-NR-14003-NP, Functional Design Requirements for a Core Protection Calculator System for APR1400, Rev. 1 (March 2017)	ML17114A522
APR1400-Z-A-NR-14019-NP, CCF Coping Analysis, Rev. 3 (July 2018)	ML18225A340
APR1400-Z-J-NR-14002-NP, Diversity and Defense-in-Depth, Rev. 3 (May 2018)	ML18214A557
APR1400-Z-J-NR-14012-NP, Control System CCF Analysis, Rev. 3 (May 2018)	ML18212A343
APR1400-Z-J-NR-14013-NP, Response Time Analysis of Safety I&C System, Rev. 2 (January 2018)	ML18087A110
APR1400-Z-M-NR-14008-NP, Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown, Rev. 1 (January 2018)	ML18087A112
APR1400-F-C-NR-14001-NP, CPC Setpoint Analysis Methodology for APR1400, Rev. 3 (June 2018)	ML18199A563
APR1400-F-C-NR-14002-NP, Functional Design Requirements for a Core Operating Limit Supervisory System for APR1400, Rev. 1 (February 2017)	ML17094A132
APR1400-E-B-NR-16001-NP, Evaluation of Main Steam and Feedwater Piping Applied to the Graded Approach for the APR1400, Rev. 0 (July 2017)	ML18178A215
APR1400-E-B-NR-16002-NP, Evaluation of Safety Injection and Shutdown Cooling Piping Applied to	ML18178A217

the Graded Approach for the APR1400, Rev. 1 (May 2018)	
APR1400-H-N-NR-14005-NP, Summary Stress Report for Primary Piping, Rev. 2 (September 2016)	ML18178A218
APR1400-E-X-NR-14001-NP, Equipment Qualification Program, Rev. 4 (July 2018)	ML18214A563
Westinghouse Topical Report	
WCAP-10697-NP-A, Common Qualified Platform Topical Report, Rev. 3 (February 2013)	ML13112A108
Combustion Engineering, Inc. Technical Reports	
CEN-312-NP, Overview Description of the Core Operating Limit Supervisory System (COLSS), Rev. 1-NP (November 1986)	ML19066A067
CEN-310-NP-A, CPC and Methodology for the CPC Improvement Program (April 1986)	ML19066A085

The NRC may post materials related to this document, including public comments, on the Federal Rulemaking Web site at https://www.regulations.gov under Docket ID NRC-2015-0224. The Federal Rulemaking Web site allows you to receive alerts when changes or additions occur in a docket folder. To subscribe: 1) navigate to the docket folder (NRC-2015-0224); 2) click the "Sign up for E-mail Alerts" link; and 3) enter your e-mail address and select how frequently you would like to receive e-mails (daily, weekly, or monthly).

XVII. Procedures for Access to Proprietary and Safeguards Information for Preparation of Comments on the APR1400 Design Certification Rule

This section contains instructions regarding how the non-publicly available documents related to this rule, and specifically those listed in Table 1.6-1 and 1.6-2 beginning on page 1.6-2 of Tier 2 of the DCD, may be accessed by interested persons who wish to comment on the design certification. These documents contain proprietary information and safeguards information (SGI). Requirements for access to SGI are primarily set forth in 10 CFR parts 2 and 73. This section provides information specific to this rule; however, nothing in this section is intended to conflict with the SGI regulations.

Interested persons who desire access to proprietary information on the APR1400 design should first request access to that information from KEPCO/KHNP, the design certification applicant. A request for access should be submitted to the NRC if the applicant does not either grant or deny access by the 10-day deadline described in the following section.

One of the non-publicly available documents, APR1400-E-A-NR-14002-P-SGI, contains both proprietary information and SGI. If you need access to proprietary information in that document in order to develop comments within the scope of this rule, then your request for access should first be submitted to KEPCO/KHNP in accordance with the previous paragraph. By contrast, if you need access to the SGI in order to provide comments, then your request for access to the SGI must be submitted to the NRC as described further in this section. Therefore, if you need access to both proprietary information and SGI in that document then you should request access to the information in separate requests submitted to both KEPCO/KHNP and the NRC.

Submitting a Request to the NRC for Access

Within 10 days after publication of this rule, any individual or entity who believes access to proprietary information or SGI is necessary in order to submit comments on this APR1400 design certification rule may request access to such information. Requests for access to proprietary information or SGI submitted more than 10 days after publication of this document will not be considered absent a showing of good cause for the late filing explaining why the request could not have been filed earlier.

The requestor shall submit a letter requesting permission to access proprietary information and/or SGI to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Attention: Rulemakings and Adjudications Staff, Washington, DC 20555–

0001. The expedited delivery or courier mail address is: Office of the Secretary, U.S. Nuclear Regulatory Commission, Attention: Rulemakings and Adjudications Staff, 11555 Rockville Pike, Rockville, Maryland 20852. The email address for the Office of the Secretary is rulemaking.comments@nrc.gov. The requester must send a copy of the request to the design certification applicant at the same time as the original transmission to the NRC using the same method of transmission. Requests to the applicant must be sent to Yun-Ho Kim, President, KHNP Central Research Institute, 70, 1312-gil, Yuseong-daero, Yuseong-gu, Daejeon, 34101, Korea.

The request must include the following information:

1. The name of this design certification, APR1400 Design Certification; the rulemaking identification number, RIN 3150–AJ67; the rulemaking docket number, NRC–2015–0224; and the *Federal Register* citation for this rule.
2. The name, address, and email or FAX number of the requester.
3. If the requester is an entity, the name of the individual(s) to whom access is to be provided, including the identity of any expert, consultant, or assistant who will aid the requestor in evaluating the information.
4. If the request is for proprietary information, the requester's need for the information in order to prepare meaningful comments on the design certification must be demonstrated. Each of the following areas must be addressed with specificity:
   a. The specific issue or subject matter on which the requester wishes to comment;
   b. An explanation why information ~~which~~ that is publicly available is insufficient to provide the basis for developing meaningful comment on the APR1400 design certification rule with respect to the issue or subject matter described in paragraph 4.a. of this section; and

c. The technical competence (demonstrable knowledge, skill, training or education) of the requestor to effectively utilize the requested proprietary information to provide the basis for meaningful comment. Technical competence may be shown by reliance on a qualified expert, consultant, or assistant who satisfies these criteria.

d. A chronology and discussion of the requester's attempts to obtain the information from the design certification applicant, and the final communication from the requester to the applicant and the applicant's response, if any was provided, with respect to the request for access to proprietary information must be submitted.

5. If the request is for SGI, a statement that explains each individual's "need to know" the SGI, as required by 10 CFR 73.2 and 10 CFR 73.22(b)(1). Consistent with the definition of "need to know" as stated in 10 CFR 73.2, the statement must explain:

    a. The specific issue or subject matter on which the requester wishes to comment;

    b. An explanation of why publicly available information is insufficient to provide the basis for developing meaningful comment on the design certification with respect to the issue or subject matter described in paragraph 5.a. of this section and why the SGI requested is indispensable in order to develop meaningful comments;[4] and

    c. The technical competence (demonstrable knowledge, skill, training or education) of the requestor to effectively utilize the requested SGI to

[4] Broad SGI requests under these procedures are unlikely to meet the standard for need to know. Furthermore, NRC staff redaction of information from requested documents before their release may be appropriate to comport with this requirement. The procedures in this document do not authorize unrestricted disclosure or less scrutiny of a requester's need to know than ordinarily would be applied in connection with either adjudicatory or non-adjudicatory access to SGI.

provide the basis and specificity for meaningful comment. Technical competence may be shown by reliance on a qualified expert, consultant, or assistant who satisfies these criteria.

d. A completed Form SF-85, "Questionnaire for Non-Sensitive Positions," for each individual who would have access to SGI. The completed Form SF-85 will be used by the Office of Administration to conduct the background check required for access to SGI, as required by 10 CFR part 2, subpart C, and 10 CFR 73.22(b)(2), to determine the requestor's trustworthiness and reliability. For security reasons, Form SF-85 can only be submitted electronically through the electronic questionnaire for investigations processing (e-QIP) website, a secure website that is owned and operated by the Office of Personnel Management. To obtain online access to the form, the requestor should contact the NRC's Office of Administration at 301-415-3710.[5]

e. A completed Form FD-258 (fingerprint card), signed in original ink, and submitted in accordance with 10 CFR 73.57(d). Copies of Form FD-258 may be obtained by writing the Office of Administrative Services, Mail Services Center, Mail Stop P1-37, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by email to MAILSVC.Resource@nrc.gov. The fingerprint card will be used to satisfy the requirements of 10 CFR part 2, subpart C, 10 CFR 73.22(b)(1), and Section 149 of the Atomic Energy Act of 1954, as amended, which mandates that all persons with access to SGI must be fingerprinted for an FBI identification and criminal history records check.

[5] The requester will be asked to provide his or her full name, social security number, date and place of birth, telephone number, and email address.

f. A check or money order in the amount of $357.00[6] payable to the U.S. Nuclear Regulatory Commission for each individual for whom the request for access has been submitted; and

g. If the requester or any individual who will have access to SGI believes they belong to one or more of the categories of individuals relieved from the criminal history records check and background check requirements, as stated in 10 CFR 73.59, the requester should also provide a statement specifically stating which relief the requester is invoking, and explaining the requester's basis (including supporting documentation) for believing that the relief is applicable. While processing the request, the NRC's Office of Administration, Personnel Security Branch, will make a final determination whether the stated relief applies. Alternatively, the requester may contact the Office of Administration for an evaluation of their status prior to submitting the request. Persons who are not subject to the background check are not required to complete the SF–85 or Form FD–258; however, all other requirements for access to SGI, including the need to know, are still applicable.

Copies of documents and materials required by paragraphs 5.d.–g., as applicable, of this section must be sent to the following address: Office of Administration, U.S. Nuclear Regulatory Commission, Personnel Security Branch, Mail Stop TWF–07D04M, 11555 Rockville Pike, Rockville, MD 20852. These documents and materials should not be included with the request letter to the Office of the Secretary, but the request letter should state that the forms and fees have been submitted as required.

[6] This fee is subject to change pursuant to the Office of Personnel Management's adjustable billing rates.

To avoid delays in processing requests for access to SGI, all forms should be reviewed for completeness and accuracy (including legibility) before submitting them to the NRC. The NRC will return incomplete or illegible packages to the sender without processing.

Based on an evaluation of the information submitted under paragraphs 4.a.–4.d. or 5.a.–g. of this section, as applicable, the NRC staff will determine within 10 days of receipt of the written access request whether the requester has established a legitimate need for access to proprietary information or need to know the SGI requested.

Determination of Legitimate Need for Access

For proprietary information access requests, if the NRC staff determines that the requester has established a legitimate need for access to proprietary information, the NRC staff will notify the requester in writing that access to proprietary information has been granted. The NRC staff must first notify the design certification applicant of the staff's determination to grant access to the requester not less than 10 days before informing the requester of the staff's decision. If the applicant wishes to challenge the NRC staff's determination, it must follow the procedures in Predisclosure Procedures for Proprietary Information Constituting Trade Secrets or Confidential Commercial or Financial Information of this section. The NRC staff will not provide ~~the requestor~~ access to disputed proprietary information to the requester until the procedures are completed as described in Predisclosure Procedures for Proprietary Information Constituting Trade Secrets or Confidential Commercial or Financial Information of this section. The written notification will contain instructions on how the requestor may obtain copies of the requested documents, and any other conditions that may apply to access to those documents. These conditions may include, but are not limited to, the signing of a Non-Disclosure Agreement or Affidavit setting forth terms and conditions to prevent the unauthorized or inadvertent disclosure of proprietary information by each individual who will be granted access.

For requests for access to SGI, if the NRC staff determines that the requester has established a need to know the SGI, the NRC's Office of Administration will then determine, based upon completion of the background check, whether the proposed recipient is trustworthy and reliable, as required for access to SGI by 10 CFR 73.22(b). If the NRC's Office of Administration determines that the individual or individuals are trustworthy and reliable, the NRC will promptly notify the requester in writing. The notification will provide the names of approved individuals as well as the conditions under which the SGI will be provided. Those conditions may include, but are not limited to, the signing of a Non-Disclosure Agreement or Affidavit by each individual who will be granted access to SGI.

Release and Storage of SGI

Prior to providing SGI to the requester, the NRC staff will conduct (as necessary) an inspection to confirm that the recipient's information protection system is sufficient to satisfy the requirements of 10 CFR 73.22. Alternatively, recipients may opt to view SGI at an approved SGI storage location rather than establish their own SGI protection program to meet SGI protection requirements.

Filing of Comments on the APR1400 Design Certification Rule Based on Non-Public Information

Any comments in this rulemaking proceeding that are based upon the disclosed proprietary or SGI information must be filed by the requester no later than 25 days after receipt of (or access to) that information, or the close of the public comment period, whichever is later. The commenter must comply with all NRC requirements regarding the submission of proprietary and SGI information to the NRC when submitting comments to the NRC (including marking and transmission requirements).

Review of Denials of Access

If the request for access to proprietary information or SGI is denied by the NRC staff, the NRC staff shall promptly notify the requester in writing, briefly stating the reason or reasons for the denial.

Before the Office of Administration makes a final adverse determination regarding the trustworthiness and reliability of the proposed recipient(s) for access to SGI, the Office of Administration, in accordance with 10 CFR 2.336(f)(1)(iii), must provide the proposed recipient(s) any records that were considered in the trustworthiness and reliability determination, including those required to be provided under 10 CFR 73.57(e)(1), so that the proposed recipient(s) have an opportunity to correct or explain the record.

Appeals from a denial of access must be made to the NRC's Executive Director for Operations (EDO) under 10 CFR 9.29. The decision of the EDO constitutes final agency action under 10 CFR 9.29(d).

Predisclosure Procedures for Proprietary Information Constituting Trade Secrets or Confidential Commercial or Financial Information

The NRC will follow the procedures in 10 CFR 9.28 if the NRC staff determines, under the Determination of Legitimate Need for Access of this section, that access to proprietary information constituting trade secrets or confidential commercial or financial information will be provided to the requester. However, any objection filed by the applicant under 10 CFR 9.28(b) must be filed within 15 days of the NRC staff notice in the Determination of Legitimate Need for Access of this section rather than the 30-day period provided for under 10 CFR 9.28(b). In applying the provisions of 10 CFR 9.28, the applicant for the design certification rule will be treated as the "submitter."

XVIII. Incorporation by Reference—Reasonable Availability to Interested Parties

The NRC is incorporating by reference the APR1400 design control document, revision 3. As described in the "Discussion" section of this document, the generic design control document combined into a single document Tier 1 and Tier 2 information (including the technical and topical reports referenced in Chapter 1)~~,~~ and generic technical specifications in order to effectively control this information and facilitate its incorporation by reference into the rule.

The NRC is required by law to obtain approval for incorporation by reference from the Office of the Federal Register (OFR). The OFR's requirements for incorporation by reference are set forth in 1 CFR part 51. The OFR regulations require an agency to include in a direct final rule a discussion of the ways that the materials the agency incorporates by reference are reasonably available to interested parties or how it worked to make those materials reasonably available to interested parties. The discussion in this section complies with the requirement for direct final rules as set forth in 1 CFR 51.5(b)(2).

The NRC considers "interested parties" to include all potential NRC stakeholders, not only the individuals and entities regulated or otherwise subject to the NRC's regulatory oversight. These NRC stakeholders are not a homogenous group but vary with respect to the considerations for determining reasonable availability. Therefore, the NRC distinguishes between different classes of interested parties for the purposes of determining whether the material is "reasonably available." The NRC considers the following to be classes of interested parties in NRC rulemakings with regard to the material to be incorporated by reference:

• Individuals and small entities regulated or otherwise subject to the NRC's regulatory oversight (this class also includes applicants and potential applicants or licenses and other NRC regulatory approvals) and who are subject to the material to be incorporated by reference by rulemaking. In this context, "small entities" has the same meaning as a "small entity" under 10 CFR 2.810.

• Large entities otherwise subject to the NRC's regulatory oversight (this class also includes applicants and potential applicants for licenses and other NRC regulatory approvals) and who are subject to the material to be incorporated by reference by rulemaking. In this context, "large entities" are those ~~which~~ that do not qualify as a "small entity" under 10 CFR 2.810.

• Non-governmental organizations with institutional interests in the matters regulated by the NRC.

• Other Federal agencies, states, local governmental bodies (within the meaning of 10 CFR 2.315(c)).

• Federally-recognized and State-recognized[7] Indian tribes.

• Members of the general public (i.e., individual, unaffiliated members of the public who are not regulated or otherwise subject to the NRC's regulatory oversight) who may wish to gain access to the materials which the NRC incorporates by reference by rulemaking in order to participate in the rulemaking process.

The NRC makes the materials incorporated by reference available for inspection to all interested parties, by appointment, at the NRC Technical Library, which is located

[7] State-recognized Indian tribes are not within the scope of 10 CFR 2.315(c). However, for purposes of the NRC's compliance with 1 CFR 51.5, "interested parties" includes a broad set of stakeholders, including State-recognized Indian tribes.

at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852; telephone: 301-415-7000; email: Library.Resource@nrc.gov. In addition, as described in Section XVI of this notice, documents related to this rule are available online in the NRC's Agencywide Documents Access and Management System (ADAMS) Public Documents collection at https://www.nrc.gov/reading-rm/adams.html.

The NRC concludes that the materials the NRC is incorporating by reference in this rule are reasonably available to all interested parties because the materials are available to all interested parties in multiple ways and in a manner consistent with their interest in the materials.

List of Subjects in 10 CFR Part 52

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

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; the Nuclear Waste Policy Act of 1982, as amended; and 5 U.S.C. 552 and 553, the NRC is amending 10 CFR part 52:

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

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

Authority: Atomic Energy Act of 1954, secs. 103, 104, 147, 149, 161, 181, 182, 183, 185, 186, 189, 223, 234 (42 U.S.C. 2133, 2134, 2167, 2169, 2201, 2231, 2232, 2233, 2235, 2236, 2239, 2273, 2282); Energy Reorganization Act of 1974, secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); 44 U.S.C. 3504 note.

§ 52.11 [Amended]

2. In § 52.11(b), add "F," in alphabetical order to the list of appendices.

3. Add Appendix F to part 52 to read as follows:

Appendix F to Part 52—Design Certification Rule for the APR1400 Design

I. INTRODUCTION

Appendix F constitutes the standard design certification for the Advanced Power Reactor 1400 (APR1400) design, in accordance with 10 CFR part 52, subpart B. The applicant for certification of the APR1400 design is Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. (KEPCO/KHNP).

II. DEFINITIONS

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

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

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

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

1. Definitions and general provisions;

2. Design descriptions;

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

4. Significant site parameters; and

5. Significant interface requirements.

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

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

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

3. COL Items (COL license information), which identify certain matters that must be addressed in the site-specific portion of the final safety analysis report by an applicant who references this appendix. These items constitute information requirements but are not the only acceptable set of information in the final safety analysis report. An applicant

may depart from or omit these items, provided that the departure or omission is identified and justified in the final safety analysis report. After issuance of a construction permit or COL, these items are not requirements for the licensee unless such items are restated in the final safety analysis report.

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

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

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

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

III. Scope and Contents

A. Incorporation by reference approval. The APR1400 material is approved for incorporation by reference by the Director of the Office of the Federal Register under 5 U.S.C. 552(a) and 1 CFR part 51. You may obtain copies of the generic DCD from Yun-Ho Kim, President, KHNP Central Research Institute, 70, 1312-gil, Yuseong-daero, Yuseong-gu, Daejeon, 34101, Korea. You can view the generic DCD online in the NRC Library at https://www.nrc.gov/reading-rm/adams.html. In ADAMS, search under ADAMS Accession No. ML18228A667. If you do not have access to ADAMS or if you have problems accessing documents located in ADAMS, contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, at 301-415-3747, or by e-mail at PDR.Resource@nrc.gov. Copies of this document are available for examination and copying at the NRC's PDR located at Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852. Copies are also available for examination at the NRC Library located at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852, telephone: 301-415-5610, e-mail: Library.Resource@nrc.gov. All approved material is available for inspection at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202-741-6030 or go to https://www.archives.gov/federal-register/cfr/ibrlocations.html.

1. APR1400 Design Control Document Tier 1 (APR1400-K-X-IT-14001-NP), Revision 3 (August 2018).

2. APR1400 Design Control Document Tier 2 (APR1400-K-X-FS-14002-NP), Revision 3 (August 2018), including:

a. Chapter 1, Introduction and General Description of the Plant.

KHNP Topical and Technical Reports

i. APR1400-F-A-TR-12004-NP-A, Realistic Evaluation Methodology for Large-Break LOCA of the APR1400, Rev. 1 (August 2018).

ii. APR1400-F-C-TR-12002-NP-A, KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design, Rev. 0 (April 2017).

iii. APR1400-F-M-TR-13001-NP-A, PLUS7 Fuel Design for the APR1400, Rev. 1 (August 2018).

iv. APR1400-K-Q-TR-11005-NP-A, KHNP Quality Assurance Program Description (QAPD) for the APR1400 Design Certification, Rev. 2 (October 2016).

v. APR1400-Z-M-TR-12003-NP-A, Fluidic Device Design for the APR1400, Rev. 0 (April 2017).

vi. APR1400-E-I-NR-14001-NP, Human Factors Engineering Program Plan, Rev. 4 (July 2018).

vii. APR1400-E-I-NR-14002-NP, Operating Experience Review Implementation Plan, Rev. 2 (January 2018).

viii. APR1400-E-I-NR-14003-NP, Functional Requirements Analysis and Function Allocation Implementation Plan, Rev. 2 (January 2018).

ix. APR1400-E-I-NR-14004-NP, Task Analysis Implementation Plan, Rev. 3 (May 2018).

x. APR1400-E-I-NR-14006-NP, Treatment of Important Human Actions Implementation Plan, Rev. 3 (May 2018).

xi. APR1400-E-I-NR-14007-NP, Human-System Interface Design Implementation Plan, Rev. 3 (May 2018).

xii. APR1400-E-I-NR-14008-NP, Human Factors Verification and Validation Implementation Plan, Rev. 3 (May 2018).

xiii. APR1400-E-I-NR-14010-NP, Human Factors Verification and Validation Scenarios, Rev. 2 (January 2018).

xiv. APR1400-E-I-NR-14011-NP, Basic Human-System Interface, Rev. 3 (May 2018).

xv. APR1400-E-I-NR-14012-NP, Style Guide, Rev. 2 (January 2018).

xvi. APR1400-E-N-NR-14001-NP, Design Features ~~t~~To Address GSI-191, Rev. 3 (February 2018).

xvii. APR1400-E-P-NR-14005-NP, Evaluations and Design Enhancements ~~t~~To Incorporate Lessons Learned from Fukushima Dai-Ichi Nuclear Accident, Rev. 2 (July 2017).

xviii. APR1400-E-S-NR-14004-NP, Evaluation of Effects of HRHF Response Spectra on SSCs, Rev. 3 (December 2017).

xix. APR1400-E-S-NR-14005-NP, Evaluation of Structure-Soil-Structure Interaction (SSSI) Effects, Rev. 2 (December 2017).

xx. APR1400-E-S-NR-14006-NP, Stability Check for NI Common Basemat, Rev. 5 (May 2018).

xxi. APR1400-F-A-NR-14001-NP, Small Break LOCA Evaluation Model, Rev. 1 (March 2017).

xxii. APR1400-F-A-NR-14003-NP, Post-LOCA Long Term Cooling Evaluation Model, Rev. 1 (March 2017).

xxiii. APR1400-H-N-NR-14012-NP, Mechanical Analysis for New and Spent Fuel Storage Racks, Rev. 3 (August 2017).

xxiv. APR1400-K-I-NR-14005-NP, Staffing and Qualifications Implementation Plan, Rev. 1 (February 2018).

xxv. APR1400-K-I-NR-14009-NP, Design Implementation Plan, Rev. 1 (February 2017).

xxvi. APR1400-Z-A-NR-14006-NP, Non-LOCA Safety Analysis Methodology, Rev. 1 (February 2017).

xxvii. APR1400-Z-A-NR-14007-NP, Mass and Energy Release Methodologies for LOCA and MSLB, Rev. 2 (May 2018).

xxviii. APR1400-Z-A-NR-14011-NP, Criticality Analysis of New and Spent Fuel Storage Racks, Rev. 3 (May 2018).

xxix. APR1400-A-N-NR-17001-NP (WCAP-17889-P), Validation of SCALE 6.1.2 with 238-Group ENDF/B-VII.0 Cross Section Library for APR1400 Design Certification, Rev. 0 (June 2014).

xxx. APR1400-Z-J-NR-14001-NP, Safety I&C System, Rev. 3 (May 2018).

xxxi. APR1400-Z-J-NR-14003-NP, Software Program Manual, Rev. 3 (May 2018).

xxxii. APR1400-E-J-NR-17001-NP, Secure Development and Operational Environment for APR1400 Computer-Based I&C Safety Systems, Rev. 0 (September 2017).

xxxiii. APR1400-Z-J-NR-14004-NP, Uncertainty Methodology and Application for Instrumentation, Rev. 2 (January 2018).

xxxiv. APR1400-Z-J-NR-14005-NP, Setpoint Methodology for Safety-Related Instrumentation, Rev. 2 (January 2018).

xxxv. APR1400-E-J-NR-14001-NP, Component Interface Module, Rev. 1 (March 2017).

xxxvi. APR1400-F-C-NR-14003-NP, Functional Design Requirements for a Core Protection Calculator System for APR1400, Rev. 1 (March 2017).

xxxvii. APR1400-Z-A-NR-14019-NP, CCF Coping Analysis, Rev. 3 (July 2018).

xxxviii. APR1400-Z-J-NR-14002-NP, Diversity and Defense-in-Depth, Rev. 3 (May 2018).

xxxix. APR1400-Z-J-NR-14012-NP, Control System CCF Analysis, Rev. 3 (May 2018).

xl. APR1400-Z-J-NR-14013-NP, Response Time Analysis of Safety I&C System, Rev. 2 (January 2018).

xli. APR1400-Z-M-NR-14008-NP, Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown, Rev. 1 (January 2018).

xlii. APR1400-F-C-NR-14001-NP, CPC Setpoint Analysis Methodology for APR1400, Rev. 3 (June 2018).

xliii. APR1400-F-C-NR-14002-NP, Functional Design Requirements for a Core Operating Limit Supervisory System for APR1400, Rev. 1 (February 2017).

xliv. APR1400-E-B-NR-16001-NP, Evaluation of Main Steam and Feedwater Piping Applied to the Graded Approach for the APR1400, Rev. 0 (July 2017).

xlv. APR1400-E-B-NR-16002-NP, Evaluation of Safety Injection and Shutdown Cooling Piping Applied to the Graded Approach for the APR1400, Rev. 1 (May 2018).

xlvi. APR1400-H-N-NR-14005-NP, Summary Stress Report for Primary Piping, Rev. 2 (September 2016).

xlvii. APR1400-E-X-NR-14001-NP, Equipment Qualification Program, Rev. 4 (July 2018).

Westinghouse Topical Report

xlviii. WCAP-10697-NP-A, Common Qualified Platform Topical Report, Rev. 3 (February 2013).

Combustion Engineering, Inc. Technical Reports

xlix. CEN-312-NP, Overview Description of the Core Operating Limit Supervisory System (COLSS), Rev. 1-NP (November 1986).

l. CEN-310-NP-A, CPC and Methodology for the CPC Improvement Program (April 1986).

b. Chapter 2, Site Characteristics.

c. Chapter 3, Design of Structures, Systems, Components, and Equipment.

d. Chapter 4, Reactor.

e. Chapter 5, Reactor Coolant System and Connecting Systems.

f. Chapter 6, Engineered Safety Features.

g. Chapter 7, Instrumentation and Controls.

h. Chapter 8, Electric Power.

i. Chapter 9, Auxiliary Systems.

j. Chapter 10, Steam and Power Conversion System.

k. Chapter 11, Radioactive Waste Management.

l. Chapter 12, Radiation Protection.

m. Chapter 13, Conduct of Operations.

n. Chapter 14, Verification Programs.

o. Chapter 15, Transient and Accident Analyses.

p. Chapter 16, Technical Specifications.

q. Chapter 17, Quality Assurance and Reliability Assurance.

r. Chapter 18, Human Factors Engineering.

s. Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation.

B. An applicant or licensee referencing this appendix, in accordance with Section IV of this appendix, shall incorporate by reference and comply with the requirements of this appendix except as otherwise provided in this appendix.

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

D. If there is a conflict between the generic DCD and either the application for the design certification of the APR1400 design or the NUREG, "Final Safety Evaluation Report Related to Certification of the APR1400 Standard Design," then the generic DCD controls.

E. Design activities for structures, systems, and components that are entirely 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 that wishes to reference 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 APR1400 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 items; and

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

3. Include, in the plant-specific DCD, the sensitive, unclassified, non-safeguards information (including proprietary information and security-related information) and safeguards information referenced in the APR1400 generic DCD.

4. Include, as part of its application, a demonstration that an entity other than KEPCO/KHNP is qualified to supply the APR1400 design, unless KEPCO/KHNP supplies the design for the applicant's use.

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

V. APPLICABLE REGULATIONS

A. Except as indicated in paragraph B of this section, the regulations that apply to the APR1400 design are in 10 CFR parts 20, 50, ~~52,~~ 73, and 100, codified as of **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, that are applicable and technically relevant, as described in the final safety evaluation report.

B. The APR1400 design is exempt from portions of the following regulations:

Commented [A2]: Staff should return to the usage in DC rules in 10 CFR part 52, appendices A-D of treating this designation as an indication of the content of the individual paragraphs set off with em dashes rather than as the title of the section immediately preceding it as was done in 10 CFR part 52, appendix E. If use of the title is desired or appropriate, it should be set off with quotation marks and the appropriate punctuation.

1. Paragraph (f)(2)(iv) of 10 CFR 50.34 – ~~Contents of Applications: Technical Information-~~ Plant Safety Parameter Display Console – codified as of **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**.

VI. ISSUE RESOLUTION

A. The Commission has determined that the structures, systems, and components and design features of the APR1400 design comply with the provisions of the Atomic Energy Act of 1954, as amended, and the applicable regulations identified in Section V of this appendix; and therefore, provide adequate protection to the health and safety of the public. A conclusion that a matter is resolved includes the finding that additional or alternative structures, systems, and components, design features, design criteria, testing, analyses, acceptance criteria, or justifications are not necessary for APR1400 design.

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

1. All nuclear safety issues associated with the information in the final safety evaluation report, Tier 1, Tier 2, and the rulemaking record for certification of the APR1400 design, with the exception of generic TS and other operational requirements;

2. All nuclear safety and safeguards issues associated with the referenced information in the 53 non-public documents in Tables 1.6-1 and 1.6-2 of Tier 2 of the DCD, which contain sensitive unclassified non-safeguards information (including proprietary information and security-related information) and safeguards information and which, in context, are intended as requirements in the generic DCD for the APR1400 design;

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

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

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

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

7. All environmental issues concerning severe accident mitigation design alternatives associated with the information in the NRC's environmental assessment for the APR1400 design (ADAMS Accession No. ML18306A607) and APR1400-E-P-NR-14006, Revision 2, "Severe Accident Mitigation Design Alternatives (SAMDAs) for the APR1400" (ML18235A158) for plants referencing this appendix whose site characteristics fall within those site parameters specified in APR1400-E-P-NR-14006.

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, and components or design features as described in the generic DCD;

2. Provide additional or alternative structures, systems, and 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, and 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 to review portions of the design certification or references containing safeguards information or sensitive unclassified non-safeguards information (including proprietary information, such as trade secrets and commercial or financial information obtained from a person that are privileged or confidential (10 CFR 2.390 and 10 CFR part 9), and security-related information), for the purpose of participating in the hearing required by § 52.85, the hearing provided under § 52.103, or in any other proceeding relating to this appendix, in which interested persons have a right to request an adjudicatory hearing.

VII. DURATION OF THIS APPENDIX

This appendix may be referenced for a period of 15 years from **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, except as provided for in §§ 52.55(b) and 52.57(b). This appendix remains valid for an applicant or licensee who references this appendix until the application is withdrawn or the license expires, including any period of extended operation under a renewed license.

VIII. PROCESSES FOR CHANGES AND DEPARTURES

A. Tier 1 information.

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

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

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

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

B. Tier 2 information.

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

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

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

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

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

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

litigation in the same manner as other issues material to the license hearing. The granting of an exemption to a licensee must be subject to an opportunity for a hearing in the same manner as license amendments.

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

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

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

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

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

(4) Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the plant-specific DCD;

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

(6) Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any evaluated previously in the plant-specific DCD;

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

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

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

(1) There is a substantial increase in the probability of an ex-vessel severe accident such that a particular ex-vessel severe accident previously reviewed and determined to be not credible could become credible; or

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

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

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

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

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

C. Operational requirements.

1. Changes to APR1400 DC generic TS and other operational requirements that were completely reviewed and approved in the design certification rulemaking and do not require a change to a design feature in the generic DCD are governed by the requirements in 10 CFR 50.109. Changes that require a change to a design feature in the generic DCD are governed by the requirements in paragraphs A or B of this section.

2. Changes to APR1400 DC generic TS and other operational requirements are applicable to all applicants who reference this appendix, except those for which the change has been rendered technically irrelevant by action taken under paragraphs C.3 or C.4 of this section.

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

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

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

3. An applicant or licensee who references this appendix shall prepare and maintain written evaluations 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 periods of renewal).

4.a. The applicant for the APR1400 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 periods 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 departure. 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 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 50.71(e) and 52.3.

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

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

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

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

Dated at Rockville, Maryland, this xxth day of Xxxxx, 2019.

For the Nuclear Regulatory Commission.

Annette Vietti-Cook,
Secretary of the Commission.

[7590-01-P]

KLS Edits

NUCLEAR REGULATORY COMMISSION

10 CFR Part 52

[NRC-2015-0224]

RIN 3150-AJ67

Advanced Power Reactor 1400 (APR1400) Design Certification

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is proposing to amend its regulations to certify the Advanced Power Reactor 1400 (APR1400) standard ~~plant~~ design. ~~This action is necessary so that a~~Applicants or licensees intending to construct and operate an APR1400 standard ~~plant~~ design may do so by referencing this design certification (DC) rule. The applicant for the certification of the APR1400 standard ~~plant~~ design is Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. (KEPCO/KHNP).

DATES: Submit comments by **[INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. Comments received after this date will be considered if it is practical to do so, but the NRC is able to ensure consideration only for comments received on or before this date.

ADDRESSES: You may submit comments by any of the following methods:

- **Federal Rulemaking Web Site:** Go to https://www.regulations.gov and search for Docket ID NRC-2015-0224. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical questions contact the individuals listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **E-mail comments to:** Rulemaking.Comments@nrc.gov. If you do not receive an automatic e-mail reply confirming receipt, then contact us at 301-415-1677.

- **Fax comments to:** Secretary, U.S. Nuclear Regulatory Commission at 301-415-1101.

- **Mail comments to:** Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff.

- **Hand deliver comments to:** 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. (Eastern Time) Federal workdays; telephone: 301-415-1677.

For additional direction on obtaining information and submitting comments, see "Obtaining Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

FOR FURTHER INFORMATION CONTACT: Yanely Malave, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-1519;~~,~~ e-mail: Yanely.Malave@nrc.gov, or William Ward, Office of New Reactors, telephone: 301-415-7038;~~,~~ e-mail: William.Ward@nrc.gov. Both are staff of the U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

TABLE OF CONTENTS:

I. Obtaining Information and Submitting Comments.

A. Obtaining Information

Please refer to Docket ID NRC-2015-0224 when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- **Federal Rulemaking Web Site:** Go to https://www.regulations.gov and search for Docket ID NRC-2015-0224.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at https://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, at 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 the "Availability of Documents" section.

- **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.

B. Submitting Comments

Please include Docket ID NRC-2015-0224 in your comment submission.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at https://www.regulations.gov as well as enter the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment into ADAMS.

II. Rulemaking Procedure

Because the NRC considers this action to be non-controversial, the NRC is publishing this proposed rule concurrently with a direct final rule in the Rules and Regulations section of this issue of the *Federal Register*. The direct final rule will become effective on **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. However, if the NRC receives significant adverse comments on this proposed rule by **[INSERT DATE 30 DAYS AFTER DATE OF**

PUBLICATION IN THE *FEDERAL REGISTER*], then the NRC will publish a document that withdraws the direct final rule. If the direct final rule is withdrawn, the NRC would address the comments received in response to these proposed revisions in any subsequent final rule. Absent significant modifications to the proposed revisions requiring republication, the NRC does not intend to initiate a second comment period on this action in the event the direct final rule is withdrawn.

A significant adverse comment is a comment ~~where~~ in which the commenter explains why the rule would be inappropriate~~, including challenges to the rule's underlying premise or approach, or would be ineffective or unacceptable without a change~~. A comment is adverse and significant if:

1) The comment opposes the rule and provides a reason sufficient to require a substantive response in a notice-and-comment process. For example, a substantive response is required when:

a) The comment causes the NRC to reevaluate (or reconsider) its position or conduct additional analysis;

b) The comment raises an issue serious enough to warrant a substantive response to clarify or complete the record; or

c) The comment raises a relevant issue that was not previously addressed or considered by the NRC.

2) The comment proposes a change or an addition to the rule, and it is apparent that the rule would be ineffective or unacceptable without incorporation of the change or addition.

3) The comment causes the NRC to make a change (other than editorial) to the rule.

For procedural information and the regulatory analysis, see the direct final rule published in the Rules and Regulations section of this issue of the *Federal Register*.

III. Background

Part 52 of title 10 of the *Code of Federal Regulations* (10 CFR), "Licenses, Certifications, and Approvals for Nuclear Power Plants," subpart B, "Standard Design Certifications," presents the process for obtaining standard design certifications. On December 23, 2014, KEPCO/KHNP submitted its application for certification of the APR1400 standard ~~plant~~ design (ADAMS Accession No. ML15006A098) to the NRC under subpart B of 10 CFR part 52. The NRC published a notice of receipt of the application in the *Federal Register* (80 FR 5792; February 3, 2015). ~~The KEPCO/KHNP submitted its application in accordance with subpart B of 10 CFR part 52.~~ On March 12, 2015, the NRC formally accepted the application as a docketed application for design certification (80 FR 13035; March 12, 2015). The pre-application information submitted before the NRC formally accepted the application can be found in ADAMS under Docket No. PROJ0782.

The NRC issued the final safety evaluation report for the APR1400 design on September 28, 2018. The final safety evaluation report is available in ADAMS under Accession No. ML18087A364. The NRC will publish ~~the~~ a final safety evaluation report in a NUREG titled, "Final Safety Evaluation Report Related to the Certification of the Advanced Power Reactor 1400 Standard Design." The final safety evaluation report is based on the NRC's review of revision 3 of the APR1400 design certification document.

IV. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this proposed rule, the NRC ~~intends~~ proposes to certify the APR1400 standard ~~plant~~ design for use in nuclear power plant licensing under 10 CFR parts 50 or 52. Design certifications are not generic rulemakings establishing a generally applicable standard with which all 10 CFR parts 50 and 52 nuclear power plant licensees must comply. Design certifications are Commission approvals of specific nuclear power plant designs by rulemaking. Furthermore, design certifications are initiated by an applicant for rulemaking, rather than by the NRC. This action does not constitute the establishment of a standard that contains generally applicable requirements.

V. Plain Writing

The Plain Writing Act of 2010 (Pub. L. 111-274) requires Federal agencies to write documents in a clear, concise, and well-organized manner that also follows other best practices appropriate to the subject or field and the intended audience. The NRC has written this document to be consistent with the Plain Writing Act as well as the Presidential Memorandum, "Plain Language in Government Writing," published June 10, 1998 (63 FR 31883). The NRC requests comment on the proposed rule with respect to clarity and effectiveness of the language used.

VI. Paperwork Reduction Act

This proposed rule contains (a) new or amended collection(s) of information subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501, et seq.). This proposed rule has been submitted to the Office of Management and Budget for review and approval of the information collection(s).

Type of submission, new or revision: Revision.

The title of the information collection: Appendix F to 10 CFR part 52 Design Certification Rule for the APR1400 Design.

The form number if applicable: NA.

How often the collection is required or requested: On occasion.

Who will be required or asked to respond: Applicant for a combined license or a design certification amendment.

An estimate of the number of annual responses: 1 (0 annual responses and 1 recordkeeper).

The estimated number of annual respondents: 1.

An estimate of the total number of hours needed annually to comply with the information collection requirement or request: Approximately 37 hours of additional recordkeeping burden. The only burden associated with this rule will be for recordkeeping by the applicant for this design certification.

Abstract: The NRC is proposing to amend its regulations to certify the APR1400 standard ~~plant~~ design. This action is necessary so that applicants or licensees intending to construct and operate an APR1400 standard ~~plant~~ design may do so by referencing this DC rule. The applicant for certification of the APR1400 standard ~~plan~~ design is KEPCO/KHNP.

The NRC is seeking public comment on the potential impact of the information collection contained in this proposed rule and on the following issues:

1) Is the proposed information collection necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?

2) Is the estimate of the burden of the proposed information collection accurate?

3) Is there a way to enhance the quality, utility, and clarity of the information to be collected?

4) How can the burden of the proposed information collection on respondents be minimized, including the use of automated collection techniques or other forms of information technology?

A copy of the OMB clearance package is available in ADAMS under Accession No. ML18302A089 or may be viewed free of charges at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O1-F21, Rockville, MD 20852. You may obtain information and comment submissions related to the OMB clearance package by searching on https://www.regulations.gov under Docket ID NRC-2015-0224.

You may submit comments on any aspect of these proposed information collection(s), including suggestions for reducing the burden and on the above issues, by the following methods:

- **Federal rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2015-0224.
- **Mail comments to:** Information Services Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150–0151) Office of Management and Budget, Washington, DC 20503; e-mail: oira_submission@omb.eop.gov.

Submit comments by **[INSERT DATE 30 DAYS AFTER PUBLICATION IN THE *FEDERAL REGISTER*]**. Comments received after this date will be considered if it is practical to do so, but the NRC staff is able to ensure consideration only for comments received on or before this date.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

VII. Availability of Documents

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

Documents Related to APR1400 Design Certification Rule

DOCUMENT	ADAMS ACCESSION NO. / WEB LINK / *FEDERAL REGISTER* CITATION
SECY-19~~XX~~-0020~~XXXX~~, "Direct Final Rule – APR1400 Design Certification"	ML18302A069
KEPCO and KHNP Application for Design Certification of the APR1400 Design	ML15006A037
APR1400 Design Control Document, Revision 3	ML18228A667
APR1400 Final Safety Evaluation Report	ML18087A364
APR1400 Environmental Assessment	ML18306A607
APR1400 Standard Design Approval	ML18261A187
Regulatory History of Design Certification[1]	ML003761550
KHNP Topical and Technical Reports	
APR1400-F-A-TR-12004-NP-A, Realistic Evaluation Methodology for Large-Break LOCA of the APR1400, Rev. 1 (August 2018)	ML18233A431

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

APR1400-F-C-TR-12002-NP-A, KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design, Rev. 0 (April 2017)	ML17115A559
APR1400-F-M-TR-13001-NP-A, PLUS7 Fuel Design for the APR1400, Rev. 1 (August 2018)	ML18232A140
APR1400-K-Q-TR-11005-NP-A, KHNP Quality Assurance Program Description (QAPD) for the APR1400 Design Certification, Rev. 2 (October 2016)	ML18085B044
APR1400-Z-M-TR-12003-NP-A, Fluidic Device Design for the APR1400, Rev. 0 (April 2017)	ML17129A597
APR1400-E-I-NR-14001-NP, Human Factors Engineering Program Plan, Rev. 4 (July 2018)	ML18212A345
APR1400-E-I-NR-14002-NP, Operating Experience Review Implementation Plan, Rev. 2 (January 2018)	ML18081A101
APR1400-E-I-NR-14003-NP, Functional Requirements Analysis and Function Allocation Implementation Plan, Rev. 2 (January 2018)	ML18081A091
APR1400-E-I-NR-14004-NP, Task Analysis Implementation Plan, Rev. 3 (May 2018)	ML18178A223
APR1400-E-I-NR-14006-NP, Treatment of Important Human Actions Implementation Plan, Rev. 3 (May 2018)	ML18178A224
APR1400-E-I-NR-14007-NP, Human-System Interface Design Implementation Plan, Rev. 3 (May 2018)	ML18178A212
APR1400-E-I-NR-14008-NP, Human Factors Verification and Validation Implementation Plan, Rev. 3 (May 2018)	ML18178A213
APR1400-E-I-NR-14010-NP, Human Factors Verification and Validation Scenarios, Rev. 2 (January 2018)	ML18081A088
APR1400-E-I-NR-14011-NP, Basic Human-System Interface, Rev. 3 (May 2018)	ML18178A214
APR1400-E-I-NR-14012-NP, Style Guide, Rev. 2 (January 2018)	ML18081A096
APR1400-E-N-NR-14001-NP, Design Features To Address GSI-191, Rev. 3 (February 2018)	ML18057B532
APR1400-E-P-NR-14005-NP, Evaluations and Design Enhancements To Incorporate Lessons Learned from Fukushima Dai-Ichi Nuclear Accident, Rev. 2 (July 2017)	ML18044B042
APR1400-E-S-NR-14004-NP, Evaluation of Effects of HRHF Response Spectra on SSCs, Rev. 3 (December 2017)	ML18078A709
APR1400-E-S-NR-14005-NP, Evaluation of Structure-Soil-Structure Interaction (SSSI) Effects, Rev. 2 (December 2017)	ML18078A699
APR1400-E-S-NR-14006-NP, Stability Check for NI Common Basemat, Rev. 5 (May 2018)	ML18178A221
APR1400-F-A-NR-14001-NP, Small Break LOCA Evaluation Model, Rev. 1 (March 2017)	ML17114A524

APR1400-F-A-NR-14003-NP, Post-LOCA Long Term Cooling Evaluation Model, Rev. 1 (March 2017)	ML17114A526
APR1400-H-N-NR-14012-NP, Mechanical Analysis for New and Spent Fuel Storage Racks, Rev. 3 (August 2017)	ML17244A015
APR1400-K-I-NR-14005-NP, Staffing and Qualifications Implementation Plan, Rev. 1 (February 2018)	ML17094A152
APR1400-K-I-NR-14009-NP, Design Implementation Plan, Rev. 1 (February 2017)	ML17094A153
APR1400-Z-A-NR-14006-NP, Non-LOCA Safety Analysis Methodology, Rev. 1 (February 2017)	ML17094A139
APR1400-Z-A-NR-14007-NP, Mass and Energy Release Methodologies for LOCA and MSLB, Rev. 2 (May 2018)	ML18212A338
APR1400-Z-A-NR-14011-NP, Criticality Analysis of New and Spent Fuel Storage Racks, Rev. 3 (May 2018)	ML18214A561
APR1400-A-N-NR-17001-NP (WCAP-17889-P), Validation of SCALE 6.1.2 with 238-Group ENDF/B-VII.0 Cross Section Library for APR1400 Design Certification, Rev. 0 (June 2014)	ML18044B051
APR1400-Z-J-NR-14001-NP, Safety I&C System, Rev. 3 (May 2018)	ML18212A341
APR1400-Z-J-NR-14003-NP, Software Program Manual, Rev. 3 (May 2018)	ML18214A559
APR1400-E-J-NR-17001-NP, Secure Development and Operational Environment for APR1400 Computer-Based I&C Safety Systems, Rev. 0 (September 2017)	ML18108A470
APR1400-Z-J-NR-14004-NP, Uncertainty Methodology and Application for Instrumentation, Rev. 2 (January 2018)	ML18086B757
APR1400-Z-J-NR-14005-NP, Setpoint Methodology for Safety-Related Instrumentation, Rev. 2 (January 2018)	ML18087A106
APR1400-E-J-NR-14001-NP, Component Interface Module, Rev. 1 (March 2017)	ML17094A131
APR1400-F-C-NR-14003-NP, Functional Design Requirements for a Core Protection Calculator System for APR1400, Rev. 1 (March 2017)	ML17114A522
APR1400-Z-A-NR-14019-NP, CCF Coping Analysis, Rev. 3 (July 2018)	ML18225A340
APR1400-Z-J-NR-14002-NP, Diversity and Defense-in-Depth, Rev. 3 (May 2018)	ML18214A557
APR1400-Z-J-NR-14012-NP, Control System CCF Analysis, Rev. 3 (May 2018)	ML18212A343
APR1400-Z-J-NR-14013-NP, Response Time Analysis of Safety I&C System, Rev. 2 (January 2018)	ML18087A110
APR1400-Z-M-NR-14008-NP, Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown, Rev. 1 (January 2018)	ML18087A112

APR1400-F-C-NR-14001-NP, CPC Setpoint Analysis Methodology for APR1400, Rev. 3 (June 2018)	ML18199A563
APR1400-F-C-NR-14002-NP, Functional Design Requirements for a Core Operating Limit Supervisory System for APR1400, Rev. 1 (February 2017)	ML17094A132
APR1400-E-B-NR-16001-NP, Evaluation of Main Steam and Feedwater Piping Applied to the Graded Approach for the APR1400, Rev. 0 (July 2017)	ML18178A215
APR1400-E-B-NR-16002-NP, Evaluation of Safety Injection and Shutdown Cooling Piping Applied to the Graded Approach for the APR1400, Rev. 1 (May 2018)	ML18178A217
APR1400-H-N-NR-14005-NP, Summary Stress Report for Primary Piping, Rev. 2 (September 2016)	ML18178A218
APR1400-E-X-NR-14001-NP, Equipment Qualification Program, Rev. 4 (July 2018)	ML18214A563
Westinghouse Topical Report	
WCAP-10697-NP-A, Common Qualified Platform Topical Report, Rev. 3 (February 2013)	ML13112A108
Combustion Engineering, Inc. Technical Reports	
CEN-312-NP, Overview Description of the Core Operating Limit Supervisory System (COLSS), Rev. 1-NP (November 1986)	ML19066A067
CEN-310-NP-A, CPC and Methodology for the CPC Improvement Program (April 1986)	ML19066A085

The NRC may post materials related to this document, including public comments, on the Federal Rulemaking Web site at https://www.regulations.gov under Docket ID NRC-2015-0224. The Federal Rulemaking Web site allows you to receive alerts when changes or additions occur in a docket folder. To subscribe, ~~:~~ 1) navigate to the docket folder (NRC-2015-0224); 2) click the "Sign up for E-mail Alerts" link; and 3) enter your e-mail address and select how frequently you would like to receive e-mails (daily, weekly, or monthly).

List of Subjects in 10 CFR Part 52

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

Dated at Rockville, Maryland, this xxth day of Xxxxx, 2019.

For the Nuclear Regulatory Commission.

Annette Vietti-Cook,
Secretary of the Commission.

KLS Edits

ENVIRONMENTAL ASSESSMENT BY THE U.S. NUCLEAR REGULATORY COMMISSION RELATING TO THE CERTIFICATION OF THE APR1400 STANDARD ~~PLANT~~ DESIGN

DOCKET NO. 52-046

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UNITED STATES NUCLEAR REGULATORY COMMISSION

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

RELATING TO THE CERTIFICATION OF THE

APR1400 STANDARD ~~PLANT~~ DESIGN

DOCKET NO. 52-046

The U.S. Nuclear Regulatory Commission (NRC) is issuing a design certification (DC) for the Advanced Power Reactor 1400 (APR1400) standard ~~plant~~ design in response to an application submitted on December 23, 2014, by Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. , hereinafter referred to as KEPCO/KHNP or the applicant. The NRC has decided to adopt DC rules as appendices to Part 52 of Title 10 of the *Code of Federal Regulations* (10 CFR).

The NRC has performed the following environmental assessment of the environmental impacts of the new rule and has documented its finding of no significant impact ~~in accordance with the requirements of~~ under 10 CFR 51.21 and the National Environmental Policy Act of 1969, as amended. This environmental assessment addresses the severe accident mitigation design alternatives (SAMDAs) that the NRC has considered for the APR1400 standard ~~plant~~ design. This environmental assessment does not address the site-specific environmental impacts of constructing and operating any facility that references the APR1400 DC at a particular site; those impacts will be evaluated as part of any application(s) for the siting, construction, or operation of such a facility.

As discussed in Section 5.0 of this environmental assessment, the NRC has determined that issuing this DC does not constitute a major Federal action significantly affecting the quality of the human environment. This finding is based on the generic finding made in 10 CFR 51.32(b)(1) that there is no significant environmental impact associated with the certification of a standard ~~plant~~ design under 10 CFR Part 52, Subpart B. The action does not authorize the siting, construction, or operation of a facility using the APR1400 standard ~~plant~~ design. Rather, it merely codifies the APR1400 standard ~~plant~~ design in a rule that could be referenced in a future construction permit (CP), combined license (COL), or operating license (OL) application. Furthermore, because the certification is a rule rather than a physical action, it does not involve the commitment of any resources that have alternative uses. As explained in the statements of consideration for "Licenses, Certifications, and Approvals for Nuclear Power Plants; Final Rule," (72 FR 49352, 49427; August 28, 2007), the 10 CFR 51.32(b)(1) generic finding of no significant impact is legally equivalent to a categorical exclusion. Therefore, the NRC has not prepared an environmental impact statement for the action.

Under 10 CFR 51.30(d), an environmental assessment for a DC must identify the proposed action and is otherwise limited to consideration of the costs and benefits of SAMDAs and the bases for not incorporating SAMDAs in the DC. As discussed in Section 4.0 of this environmental assessment, the NRC also reviewed KEPCO/KHNP's assessment of SAMDAs that generically apply to the APR1400 standard ~~plant~~ design. The NRC finds that KEPCO/KHNP's assessment took into consideration a reasonable set of SAMDAs, and that no additional SAMDAs beyond those currently incorporated into the APR1400 standard ~~plant~~ design would be cost-beneficial. This finding is applicable whether SAMDAs are considered at the time of the certification of the APR1400 standard ~~plant~~ design or are considered with respect to licensing a potential future facility referencing the APR1400 DC rule. In Appendix F to 10 CFR Part 52, a plant referencing the APR1400 DC rule should be sited at a location with site characteristics that are encompassed by the postulated site parameters for the DC reference plant site in APR1400-K-X-ER-14001-NP, Revision 2, "Applicant's Environmental Report – Standard Design Certification," issued August 2018, and in the supporting documents.

ENVIRONMENTAL ASSESSMENT

1.0 Identification of the Proposed Action

The proposed action is to certify the APR1400 standard ~~plant~~ design in Appendix F to 10 CFR Part 52. The new rule allows applicants to reference the certified APR1400 standard ~~plant~~ design as part of a COL application under 10 CFR Part 52, or may allow for a CP application under 10 CFR Part 50.

2.0 Need for the Proposed Action

The proposed action is to issue a rule amending 10 CFR Part 52 to certify the APR1400 standard ~~plant~~ design. The amendment allows an applicant to reference the certified APR1400 standard ~~plant~~ design as part of a COL application under 10 CFR Part 52, or may allow for a CP application under 10 CFR Part 50. Those portions of the APR1400 standard ~~plant~~ design included in the scope of the design certification rulemaking are not subject to further safety review or approval in a COL proceeding. In addition, the DC rule could resolve SAMDAs for any future COL applications for facilities that reference the certified APR1400 standard ~~plant~~ design.

3.0 Environmental Impact of the Proposed Action

The proposed action constitutes issuance of the DC as an amendment to 10 CFR Part 52 to certify the APR1400 standard ~~plant~~ design. As stated in 10 CFR 51.32(b)(1), the NRC has determined that there is no significant environmental impact associated with the issuance of a DC. The DC merely codifies the NRC's approval of the APR1400 standard ~~plant~~ design through its final safety evaluation report on the design issued during rulemaking (Agencywide Documents Access and Management System (ADAMS) Accession No.

ML18087A364). Furthermore, because the certification of the design constitutes only a rule rather than a physical action, it would not involve the commitment of any resources that have alternative uses.

As described in Section 4.0 of this environmental assessment, the NRC reviewed various alternative design features for preventing and mitigating severe accidents. The National Environmental Policy Act of 1969, as amended, requires consideration of alternatives to show that the DC rule is the appropriate course of action. The NRC's regulations at 10 CFR 51.55(a) ensure that the design ~~referenced in rulemaking~~to be certified does not exclude any cost-~~-~~beneficial design changes related to the prevention and mitigation of severe accidents.

Through its own independent analysis, the NRC concludes that KEPCO/KHNP adequately considered an appropriate set of SAMDAs and that none met the cost-~~-~~beneficial criteria. Although KEPCO/KHNP made no design changes as a result of considering SAMDAs, KEPCO/KHNP had already incorporated certain features in the APR1400 standard ~~plant~~ design on the basis of probabilistic risk assessment (PRA) results. Section 4.2 of this environmental assessment gives examples of these features. These design features relate to severe accident prevention and mitigation, but they were not considered in the SAMDA evaluation because they were already part of the APR1400 standard ~~plant~~ design (refer to Sections 19.2.2 and 19.2.3 of the design control document, "Severe Accident Prevention" and "Severe Accident Mitigation," respectively).

Finally, the DC rule, itself, does not authorize the siting, construction, or operation of a nuclear power plant. An applicant for a CP, early site permit, COL, or OL that references the APR1400 standard ~~plant~~ design will be required to address the environmental impacts of construction and operation for its specific site. The NRC will then evaluate the environmental impacts for that particular site and issue an environmental impact statement in accordance with 10 CFR Part 51 and the National Environmental Policy Act of 1969, as amended. However, the

SAMDA analysis that has been completed as part of this environmental assessment can be incorporated by reference into an environmental impact statement related to an application for siting, construction, or operation of a nuclear plant that references the APR1400 standard ~~plant~~ design.

4.0 Severe Accident Mitigation Design Alternatives

The proposed action provides finality in licensing proceedings on an application under 10 CFR Part 52 referencing the APR1400 DC rule and proposing a plant located on a site whose site characteristics fall within the postulated site parameters of the DC referenced plant site (i.e., the Surry Power Station site), as described in APR1400-K-X-ER-14001-NP and the supporting documents.

This section provides a summary of the NRC's review of KEPCO/KHNP's Standard Design Certification Environmental Report and the related APR1400 SAMDAs, as provided in APR1400-K-X-ER-14001-NP and the supporting documents. The specific details of the NRC's evaluation, summarized in this environmental assessment, are provided in a technical analysis report under ADAMS Accession No. ML18096A697.

4.1. Severe Accident Mitigation Design Alternatives

Consistent with the Commission's objectives of standardization and early resolution of design issues, the SAMDAs are being evaluated as part of the DC for the APR1400 standard ~~plant~~ design. In a 1985 policy statement (50 FR 32138; August 8, 1985), the Commission defined the term severe accident as an event that is beyond the substantial coverage of design-basis events, including events where there is substantial damage to the reactor core (whether or not there are serious offsite consequences). Design-basis events are events analyzed in accordance with the NRC's Standard Review Plan (NUREG-0800) and documented in Chapter 15, "Transient and Accident Analyses~~Safety Analysis~~," of the design control document.

As part of its DC application, KEPCO/KHNP performed a PRA for the APR1400 standard ~~plant~~ design to achieve the following objectives:

- to identify the dominant severe accident sequences that account for most of the core damage frequency and associated source terms for the design;
- to modify the design, on the basis of PRA insights, to prevent severe accidents or mitigate their consequences and thereby reduce the risk of such accidents; and
- to provide a qualitative basis for concluding that all reasonable steps have been taken to reduce the chances of severe accidents ~~to~~ occurring and to mitigate the consequences.

KEPCO/KHNP's PRA analysis is described in Chapter 19 of the APR1400 design control document, Revision 3.

The APR1400 Level 1 and Level 2 PRA models quantified six risk categories~~;~~ three for operations at-power and three for low-power and shutdown operations~~,~~ namely:

- at-power internal events
- at-power internal flooding events
- at-power internal fire events
- low-power and shutdown internal events
- low-power and shutdown internal flooding events
- low-power and shutdown internal fire events

The risks from other external events, such as high winds, seismic events, external flooding, and external fires~~, etc.~~, were determined by the PRA models to be negligible and were not further analyzed under the SAMDA assessment.

In addition to these safety considerations, applicants for reactor DCs or COLs must also consider alternative design features for severe accidents ~~as part~~in support of the NRC's environmental review. These requirements can be summarized as follows:

- Section 52.79(a)(46) requires a COL applicant to describe the plant-specific PRA and its results, with the aim of identifying potential improvements in the reliability of the core and containment heat removal systems that are significant and practical and~~, which~~ that do not impact excessively on the plant.
- Section 51.30(d) requires consideration of SAMDAs in an environmental assessment for a DC, while 10 CFR 51.50(c) sets forth the general requirements for an environmental report accompanying a COL application, including the requirement to evaluate SAMDAs.

Although these requirements are not directly related, they share common purposes, which are to consider alternatives to the proposed design, to evaluate whether potential alternative improvements in the plant design might significantly enhance safety performance during severe accidents, and to prevent reasonable alternatives from being foreclosed.

The NRC has determined that the generic evaluation of SAMDAs for the APR1400 standard design is both practical and warranted for two reasons. First, the design and construction of all plants licensed under 10 CFR Part 52 referencing the certified APR1400 standard ~~plant~~ design will be governed by the rule certifying a single design. Second, the site parameters in APR1400-K-X-ER-14001-NP and the supporting documents establish the consequences for a reasonable set of SAMDAs for the APR1400 standard ~~plant~~ design. The low residual risk of the APR1400 standard ~~plant~~ design and the limited potential for further risk reduction provides high confidence that additional cost-beneficial SAMDAs would not be found for sites with characteristics that fit within the site parameter envelope. If an actual characteristic for a particular site does not fall within the postulated site parameters, then SAMDAs that could be affected by the value of the site characteristic must be re-evaluated in the site-specific environmental report and the environmental impact statement prepared in connection with the application for the particular site. If the actual characteristics of a proposed site fall within the postulated site parameters, then the SAMDA analysis can be incorporated by reference in the site-specific environmental impact statement, and SAMDAs need not be re-evaluated in the environmental impact statement.

4.2. Potential Design Improvements Identified by KEPCO/KHNP

In APR1400-K-X-ER-14001-NP and the supporting documents, the applicant identified 153 candidate design alternatives, or design improvements, based on a review of the standard list of design alternatives provided in Table 14 of Nuclear Energy Institute 05-01A, "Severe Accident Mitigation Alternatives (SAMA) Analysis," and several license renewal environmental reports. KEPCO/KHNP eliminated certain candidate design alternatives from further consideration on the following bases:

- they were already implemented in the APR1400 standard ~~plant~~ design;
- they were not applicable to the APR1400 standard ~~plant~~ design or to the APR1400 DC;
- they had excessive implementation costs; or
- they were of very low benefit.

There were 30 candidate design alternatives that the APR1400 standard ~~plant~~ design already incorporated. The following are examples of candidate design alternatives already incorporated in the APR1400 standard design ~~such as the following~~:

- installing a gas turbine generator;
- installing an independent active or passive high pressure injection system;
- adding a diverse low pressure injection system;
- improving emergency core cooling system suction strainers;
- adding the ability to manually align the emergency core cooling system recirculation;
- adding the ability to automatically align the emergency core cooling system to recirculation mode upon refueling water storage tank depletion;
- providing an in-containment reactor water storage tank;

- creating a reactor coolant depressurization system; and
- installing an independent reactor coolant pump seal injection system, without a dedicated diesel.

The applicant initially screened the design alternatives based on their analysis in APR1400-K-X-ER-14001-NP, Section 4, "Unmitigated Risk Monetary Value." As described in Section 4.6.1 ~~below~~of this environmental assessment, if the implementation costs for a SAMDA candidate exceeded the calculated maximum benefit resulting in a negative Net Present Value, the SAMDA was not considered further. This screening process eliminated 30 potential design alternatives that were identified as being unfeasible due to excessive implementation costs or that provided negligible benefit. Another 54 SAMDA candidates were identified as not applicable to the DC stage of plant development (such as procedural processes, training, or design features not applicable at the DC stage). One potential design alternative was determined to be of very low benefit. The applicant retained the remaining 38 SAMDAs for further assessment in the cost-benefit analysis.

KEPCO/KHNP also applied insights from the APR1400 PRA by applying relevant guidance from Section 5.1, Probabilistic Safety Assessment Importance, ~~in~~ of Nuclear Energy Institute 05-01A. First, KEPCO/KHNP identified APR1400-specific dominant risk contributors, derived from the PRA, for further consideration for events. This subset of risk contributors was derived from an importance analysis of core damage cutsets using a Fussell-Vessely importance criterion of greater than 0.5 percent contribution to the total risk (i.e., the total core damage frequency). By applying this criterion, KEPCO/KHNP identified a number of basic events derived from the information in design control document Section 19.1. This process identified basic events in Section 7 of the environmental report that are associated with the six risk categories (see Tables 6a through 6f). Secondly, KEPCO/KHNP applied insights from the APR1400 PRA's top 100 cutsets by identifying any that were not included as part of the Fussell-

Vessely importance analysis review. KEPCO/KHNP identified these additional at-power and low-power and shutdown basic events, as provided in Tables 7a through 7f of the environmental report, for further consideration based on the information in design control document Section 19.1.

4.3. NRC Evaluation of Potential Design Improvements

The NRC found that the set of SAMDAs and basic events evaluated by KEPCO/KHNP addressed the major contributor to core damage. KEPCO/KHNP used a systematic and comprehensive process for identifying potential plant improvements for the APR1400 standard ~~plant~~ design, and the set of potential plant improvements identified by KEPCO/KHNP is reasonably comprehensive and, therefore, is acceptable for further evaluation. This process included reviewing insights from the plant-specific PRA study as well as assessing severe accident mitigation alternatives (SAMAs) based on accepted industry guidance.

The NRC has concluded that the applicant's assessment of the potential SAMDAs and their impacts on the APR1400 standard ~~plant~~ design is acceptable. The NRC's review did not reveal any additional design alternatives that the applicant should have considered.

4.4. Risk Reduction Potential of SAMDAs

4.4.1. KEPCO/KHNP Evaluation

KEPCO/KHNP evaluated the potential SAMDAs not screened out to assess their potential benefits by using bounding techniques to estimate the possible risk reduction. This is accomplished by associating the basic events identified with a Fussell-Vessely importance of greater than 0.5 percent, and from the top 100 cutsets to a particular SAMDA. This linkage to a SAMDA is provided for each basic event in APR1400-K-X-ER-14001-NP, Section~~s~~ 7.1 through Section 7.19. The basic event that a potential SAMDA is associated with is also provided in the "Qualitative Screening" column of Table 5 in APR1400-K-X-ER-14001-NP.

Because there are likely several basic events that are considered under a specific SAMDA, KEPCO/KHNP applied a factor of risk reduction based on the sum of Fussell-Vessely importance values for each basic event. KEPCO/KHNP determined the sum of Fussell-Vessely values for each basic event under the six risk categories for a total risk reduction percentage associated with a particular risk category (i.e., at-power internal events, internal flooding, ~~and~~ or internal fire; or low-power and shutdown internal events, internal flooding, ~~and~~ or internal fire). In several basic event cases, KEPCO/KHNP found that there were no Fussell-Vessely importance values; therefore the sum for a risk category would be zero. Section 4.4.2 discusses this assessment further.

4.4.2. NRC Evaluation

The NRC reviewed KEPCO/KHNP's bases for calculating the risk reduction for the various plant improvements and concludes that the rationale and assumptions for estimating risk reduction are reasonable. Specifically, the sum of Fussell-Vessely importance values for risk reductions is acceptable due to its conservatism (i.e., the estimated risk reduction is higher than what would actually be realized). Accordingly, the NRC based its estimates of averted risk for the potential SAMDAs on the resulting APR1400 risk reduction estimates.

4.5. Cost Impacts of Candidate SAMDAs

4.5.1. KEPCO/KHNP Evaluation

In performing the cost-~~-~~benefit analysis of the SAMDAs considered, the cost of enhancement (COE) implementation associated with potential events ~~are~~ is estimated from available information related to similar events and components of other nuclear power plant designs. The COE values of the APR1400 SAMDAs are derived from two sources. The first source is the compilation of information from the SAMA[1] analyses performed for the license renewal applications of the presently operating nuclear power plants as documented in the

[1] SAMAs are a subset of SAMDAs, which are attributes for the mitigation of severe accidents of design alternatives, procedural modifications, and training activities.

licensees' renewal environmental reports and in the final supplemental environmental impact statements under NUREG-1437. The second source is an assessment by the applicant, as presented in APR1400-K-X-ER-14001-NP. The publicly available license renewal SAMA costs are full-cost values, while the associated SAMDA costs applied by KEPCO/KHNP were conservatively set to half of the license renewal values based on an assumption that half of the cost would be from engineering and procedure updates. However, it is important to note that for license renewal SAMA evaluations, the full SAMA costs were applied in their cost-benefit analyses.

4.5.2. NRC Evaluation

On the basis of the analyses performed by KEPCO/KHNP, the NRC has concluded that the applicant's estimates of potential costs for the APR1400 SAMDAs are acceptable because the sources for the information and the cost estimates are both reasonable. First, the NRC applied this information in the cost–benefit analysis by using half of the SAMDA COE implementation value, as did KEPCO/KHNP for the APR1400 evaluation presented in APR1400-K-X-ER-14001-NP. Second, if SAMDAs were not further screened out based on the conservative assumptions, then the NRC applied the full COE implementation value. This approach facilitates the cost–benefit comparisons founded on a graded approach when assessing the averted costs using 7 percent and 3 percent discount rates. This approach is consistent with the guidance in Section 7.2 of Nuclear Energy Institute 05-01A.

4.6. Cost-Benefit Comparison

4.6.1. KEPCO/KHNP Evaluation

The methodology used by KEPCO/KHNP was based primarily on the NRC's guidance for performing cost-benefit analysis outlined in NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook." The guidance involves determining the net present value (NPV) for each SAMDA according to the following formula:

NPV = (APE + AOC + AOE + AOSC) - COE

Where:

NPV = Net present value of current risk ($);

APE = Present value of averted public exposure ($);

AOC = Present value of averted offsite property damage costs ($);

AOE = Present value of averted occupational exposure ($);

AOSC = Present value of averted onsite costs ($); and

COE = Cost of any enhancement implemented to reduce risk ($).

If the net present value of a SAMDA is negative, the cost of implementing the SAMDA is larger than the benefit associated with the SAMDA and it is not cost-beneficial. As noted above, 30 candidate SAMDAs were screened out of further analyses for this reason. If the SAMDA benefit exceeds the estimated cost resulting in a positive NPV, the SAMDA is potentially cost-beneficial.

For the representation of the maximum benefit that could be provided, the maximum benefit is calculated to be the sum of the four averted cost categories. It is represented as:

Maximum Benefit = APE + AOC + AOE + AOSC

Table 4.6.2-1 summarizes the applicant's and the NRC's estimates for each of the associated cost elements.

Table 4.6.2-1 Calculated Total Maximum Benefit

Risk Category	KEPCO/KHNP		NRC Staff	
	7%	3%	7%	3%
APE	$49,877	$98,622	$49,872	$98,612
AOC	$63,933	$126,417	$63,941	$126,429
AOE	$3,817	$8,787	$3,818	$8,786
$AOSC_{CD}$	$116,457	$276,642	$191,035	$453,773
$AOSC_{RP}$	$675,084	$1,134,638	$706,726	$1,879,727
Total Maximum Benefit	$909,168	$1,645,106	$1,015,393	$2,567,327

It is important to note that the monetary present value estimate for each risk attribute does not represent the expected reduction in risk resulting from a single accident. Rather, it is the present value of potential losses extending over the projected lifetime (in this case, 60 years) of the facility. Therefore, it reflects the expected annual loss resulting from a single accident, the possibility that such an accident could occur at any time over the licensed life, and the effect of discounting these potential future losses to present value.

The NRC issued Revision 4 of NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," in August 2004 to reflect the agency's policy on discount rates. NUREG/BR-0058, Revision 4, states that two sets of estimates should be developed — one at 3 percent and one at 7 percent. The applicant provided estimates using both discount rates.

Using the baseline 7 percent and the sensitivity 3 percent discount rates, KEPCO/KHNP calculated the maximum benefit for at-power internal events, internal flooding events, and internal fire events; along with low-power and shutdown internal events, internal flooding events, and internal fire events ~~for the baseline 7 percent and the sensitivity 3 percent discount rates~~. The results of the KEPCO/KHNP evaluation are provided in Table 4.6.2-1.

As previously discussed, 38 SAMDAs were carried to the next screening phase. In addition to these remaining SAMDAs, each basic event with a Fussell-Vessely importance of greater than 0.5 percent or part of the top 100 cutsets, if not already included as a basic event, ~~were~~ was reviewed to identify any potential SAMDAs. KEPCO/KHNP then related each of the 38 SAMDAs back to one or more of the basic events and assessed the NPV for each basic event with the following steps:

1. Assessed the maximum benefit for each basic event applying conservative assumptions for risk reductions to the AOE and AOSC categories;

2. Conservatively assessed the COE based on half of the SAMDA values obtained from source documents; and

3. Determined the NPV.

For each of the basic events/SAMDAs applying the 7 percent and 3 percent discount rates, KEPCO/KHNP evaluated the NPV and reached a conclusion of whether the enhancements were cost beneficial. KEPCO/KHNP determined, through its SAMDA~~S~~ analyses, that there were no potentially cost-beneficial enhancements for the 7 percent discount rate analysis. KEPCO/KHNP stated that its sensitivity analysis for the 3 percent discount rate showed a higher maximum benefit over the 7 percent discount rate. However, KEPCO/KHNP concluded that no design changes would provide a positive cost-benefit for either discount rate, if included in the APR1400 standard ~~plant~~ design.

4.6.2. NRC Evaluation

As shown in Table 4.6.2-1, the NRC's confirmatory analys~~i~~es for the 7 percent and 3 percent discount rates were in general agreement with the applicant for the offsite public exposure (i.e., APE), offsite property damage cost (i.e., AOC), and onsite occupational dose (i.e., AOE) averted costs. The NRC evaluation resulted in higher values than the applicant's evaluation for the onsite cleanup and decontamination (i.e., $AOSC_{CD}$) averted costs, with a similar higher result for the replacement power (i.e., $AOSC_{RP}$) averted costs.

In the $AOSC_{CD}$ evaluation, the NRC adjusted the base averted cost per event provided by NUREG/BR-0184, which was applied by KEPCO/KHNP, to current dollars, resulting in a higher value for the NRC's evaluation. The small difference between the NRC's and the applicant's $AOSC_{RP}$ averted costs for the 7 percent discount rate evaluation is principally due to applying different inflation factors to adjust the base replacement cost to current dollars. For the 3 percent discount rate analysis of the replacement power, KEPCO/KHNP applied a linear interpolation to the NPV for discount rates below 5 percent, as described near the end of

Section 5.7.6.2 of NUREG/BR-0184 (see page 5.45 of NUREG/BR-0184). Based on NRC experience in prior regulatory rulemaking analyses, the NRC applied the same replacement cost formula for both the 7 percent and 3 percent discount rates (see the formula in Section 5.7.6.2 of NUREG/BR-0184 on page 5.44). This is viewed by the NRC as being conservative as demonstrated by the larger replacement power averted cost in the NRC evaluation in comparison to the applicant's evaluation.

In its review, the NRC noted that the applicant used two assumed conservatisms in its cost-benefit analysis. The first case of conservatism involved the total averted costs in each analysis, where the applicant did not apply the percent risk reductions for the contribution to total core damage frequency to the population dose (i.e., APE) and offsite property damage (i.e., AOC) costs. The APE and AOC were based on MELCOR Accident Consequence Code System calculations and, thus, are directly tied to the size of a release. As shown by the NRC's 3 percent discount rate analysis compared to the KEPCO/KHNP 3 percent discount rate analysis, applying this reduction to only the onsite exposure (i.e., AOE) and onsite economic costs (i.e., AOSC), results in a conservative result. Namely, it will result in a total maximum benefit that is larger than if the percentage risk reduction is applied to all cost categories. The second conservative assumption involved the use of the determined COE values, as discussed in Section 4.5.1. As assessed by the NRC staff, when the applicant applies only half of the estimated COE value, the final determination of the cost-benefit analysis could more likely provide a positive NPV.

Even with the above discussed differences in the averted cost values, the NRC's confirmatory analysis also reached the same conclusion as KEPCO/KHNP that there were no cost-beneficial design alternatives when applying a 7 percent discount rate. This result is the same whether the applicant's conservative assumptions were, or were not, applied in the 7 percent discount rate analysis. Based on the NRC's review of the methodology and associated analysis, KEPCO/KHNP's assessment adequately addressed the cost-benefit analysis for the 7 percent discount rate.

For the 3 percent discount rate analysis, the NRC performed a confirmatory calculation to assess the costs and ~~-~~benefits applying the NRC results provided in Table 4.6.2-1, without applying KEPCO/KHNP's conservative assumptions. Specifically, the NRC also applied the risk reduction percentages to the APE and AOC, since they are also dependent on the released plume, and the NRC applied the full COE values. As a result, the NRC determined that there were no cost beneficial design alternatives when applying a 3 percent discount rate.

4.7. Conclusions on SAMDAs

The NRC reviewed KEPCO/KHNP's SAMDA analysis and concludes that the methods used and the implementation of the methods are appropriate. On the basis of the applicant's treatment of SAMDA benefits and costs, the NRC finds that the evaluation performed by KEPCO/KHNP is reasonable and sufficient. Based on its own independent evaluation, the NRC reached the same conclusion as KEPCO/KHNP that none of the possible candidate design alternatives are potentially cost beneficial for the APR1400 standard ~~plant~~ design. This independent evaluation was based on a reasonable treatment of costs, benefits, and sensitivities. Based on the NRC review of KEPCO/KHNP's evaluation, including KEPCO/KHNP's response to requests for additional information, the NRC concludes that KEPCO/KHNP has adequately identified areas where risk potentially could be reduced in a cost-beneficial manner and adequately assessed whether the implementation of the identified potential SAMDAs or candidate design alternatives would be cost-beneficial for the given site parameters.

Because of the magnitude of the negative NPV values, a SAMA based on operational procedures or training for an APR1400 reactor would have to cause a significant effect on the total core damage frequency ~~and/~~or have a low implementation cost to become cost-beneficial.

Based on its evaluation, the NRC concludes that it is unlikely that any of the SAMAs based on procedures or training would reduce the risk to be cost-beneficial for the given site parameters.

5.0 Finding of No Significant Impact

On the basis of 10 CFR 51.32(b)(1) and the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC is not required to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the DC rule and the documents referenced in the statement of considerations for the final rule. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, 20852. Publicly available records will be accessible electronically from the ADAMS Public Electronic Reading Room on the NRC Web site at https://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents in ADAMS should contact the NRC PDR reference staff at 1-800-397-4209 or 301-415-4737 or send an e-mail to pdr@nrc.gov.

AFFIRMATION ITEM

RESPONSE SHEET

TO: Annette Vietti-Cook, Secretary

FROM: Commissioner Baran

SUBJECT: SECY-19-0020: Direct Final Rule - Advanced Power Reactor 1400 Design Certification

Approved __X__ **Disapproved** ____ **Abstain** ____ **Not Participating** ____

COMMENTS: **Below** __X__ **Attached** __X__ **None** ____

Based on the NRC's staff's extensive safety evaluation of the APR1400 design, I approve publication of the direct final rule and companion proposed rule in the *Federal Register*, subject to the attached edits. I agree with the NRC staff that this design certification rule is unlikely to be the subject of significant adverse comments and therefore support proceeding with a direct final rule. The draft letters to Congressional Committees should be updated to reflect the current Chairmen and Ranking Members.

Entered in "STARS"
Yes __X__
No ____

SIGNATURE

3/11/19
DATE

[7590-01-P]

JMB Edits

NUCLEAR REGULATORY COMMISSION

10 CFR Part 52

[NRC-2015-0224]

RIN 3150-AJ67

Advanced Power Reactor 1400 (APR1400) Design Certification

AGENCY: Nuclear Regulatory Commission.

ACTION: Direct final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to certify the Advanced Power Reactor 1400 (APR1400) standard plant design. ~~This action is necessary so that~~ Aapplicants or licensees intending to construct and operate an APR1400 standard plant design may do so by referencing this design certification (DC) rule. The applicant for the certification of the APR1400 standard plant design is Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. (KEPCO/KHNP).

DATES: The final rule is effective **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, unless significant adverse comments are received by **[INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. If the direct final rule is withdrawn as a result of such comments, timely notice of the withdrawal will be published in the *Federal Register*. The incorporation by reference of certain publications listed in this regulation is approved by

of issues that would be resolved by this rulemaking, paragraph VI.C identifies issues that~~, which~~ are not resolved by this rulemaking, and paragraph VI.D identifies the issue finality restrictions applicable to the NRC with respect to appendix F to 10 CFR part 52.

Paragraph VI.A describes the nature of the NRC's findings in general terms and makes the findings required by § 52.54 for the NRC's approval of this DC rule.

Paragraph VI.B sets forth the scope of issues that may not be challenged as a matter of right in subsequent proceedings. The introductory phrase of paragraph VI.B clarifies that issue resolution, as described in the remainder of the paragraph, extends to the delineated NRC proceedings referencing appendix F to 10 CFR part 52. The remainder of paragraph VI.B describes the categories of information for which there is issue resolution.

Paragraph VI.C reserves the right of the NRC to impose operational requirements on applicants that reference appendix F to 10 CFR part 52. This provision reflects the fact that only some operational requirements, including portions of the generic technical specifications in Chapter 16 of the design control document, and no operational programs (e.g., operational quality assurance), were completely or comprehensively reviewed by the NRC in this design certification rulemaking proceeding. Therefore, the issue finality provisions of § 52.63 apply only to those operational requirements that either the NRC completely reviewed and approved, or formed the basis of an NRC safety finding of the adequacy of the APR1400, as documented in the NRC's final safety evaluation report. 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

applications identified dimensions of length to define critical structural sections as Tier 2* information. During recent construction activities for another design, actual dimensional lengths were found to be outside of their design tolerances. This variance did not necessarily reduce safety but did require additional license amendments to resolve the issue associated with the design tolerances, resulting in increased costs and possible construction schedule impacts. For the APR1400 design, the resolution was to revise Tier 1 and the ITAAC for these critical structural sections to use the design load and design load capacity in lieu of dimensions of length, as specific dimensions are not necessarily as important to safety. By focusing on important to safety parameters and including them in ITAAC, rather than in Tier 2* information (thus eliminating the need for Tier 2* information), the staff expects that the need for license amendments to address changes during construction will be greatly reduced while still maintaining reasonable assurance of adequate protection.

Tier 1 information

Paragraph A describes the change process for changes to Tier 1 information that are accomplished by rulemakings that amend the generic design control document and are governed by the standards in § 52.63(a)(1). A generic change under § 52.63(a)(1) will not be made to a certified design while it is in effect unless the change: 1) is necessary for compliance with NRC regulations applicable and in effect at the time the certification was issued; 2) is necessary to provide adequate protection of the public health and safety or common defense and security; 3) reduces unnecessary regulatory costs~~burden~~ and maintains protection to public health and safety and common defense and security; 4) provides the detailed design information necessary to resolve select design acceptance criteria; 5) corrects material errors in the certification information; 6) substantially increases overall safety, reliability, or security of a facility and the costs of the change are appropriate~~justified~~; or 7) contributes to increased standardization of the certification information. The rulemakings must provide for notice and opportunity for public comment on the proposed change, as required by § 52.63(a)(2). The NRC will give consideration as to whether the benefits justify the costs for plants that are already licensed or for which an application for a permit or license is under consideration.

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

Tier 2 information

Paragraph B describes the change processes for the Tier 2 information; which have the same elements as the Tier 1 change process, but some of the standards for plant-specific orders and exemptions would be different. Generic Tier 2 changes would be accomplished by rulemaking that would amend the generic design control document and would be governed by the standards in § 52.63(a)(1). A generic change under § 52.63(a)(1) would not be made to a certified design while it is in effect unless the change: 1) is necessary for compliance with NRC regulations that were applicable and in effect at the time the certification was issued; 2) is necessary to provide adequate protection of the public health and safety or common defense and security; 3) reduces unnecessary regulatory costs~~burden~~ and maintains protection to public health and safety and common defense and security; 4) provides the detailed design information necessary to resolve select design acceptance criteria; 5) corrects material errors in the certification information; 6) substantially increases overall safety, reliability, or security of a facility and the costs of the change are appropriate~~justified~~; or 7) contributes to increased standardization of the certification information.

Departures from Tier 2 would occur in four ways: 1) the NRC may order a plant-specific departure, as set forth in paragraph B.3; 2) an applicant or licensee may request an exemption from a Tier 2 requirement as set forth in paragraph B.4; 3) a licensee may make a departure without prior NRC approval under paragraph B.5; or 4) the licensee may request NRC approval for proposed departures which do not meet the requirements in paragraph B.5 as provided in paragraph B.5.e.

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

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

Paragraph B.5 would allow an applicant or licensee to depart from Tier 2 information, without prior NRC approval, if it does not involve a change to, or departure from, Tier 1 information, technical specifications, or does not require a license amendment under paragraphs B.5.b or c. The technical specifications referred to in B.5.a of this paragraph are the technical specifications in Chapter 16 of the generic design control document, including bases, for departures made prior to the issuance of the COL. After the issuance of the COL, the plant-specific technical specifications would be controlling under paragraph B.5. The requirement for a license amendment in paragraph B.5.b would be similar to the requirement in § 50.59 and would apply to all of the information in Tier 2 except for the information that resolves the severe accident issues.

Paragraph B.5.b addresses information described in the design control document to address aircraft impacts, in accordance with § 52.47(a)(28). Under § 52.47(a)(28), applicants are required to include the information required by § 50.150(b) in their design control document. 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 aircraft impact assessment required by § 50.150(a). The applicant or licensee is also required to describe in the plant-specific design control document how the modified design features and functional capabilities continue to meet the assessment requirements in § 50.150(a)(1). Submittal of this updated information is governed by the reporting requirements in Section X.B.

During an ongoing adjudicatory proceeding (e.g., for issuance of a COL) a party who believes that an applicant or licensee has not complied with paragraph B.5 when departing from Tier 2 information may petition to admit such a contention into the proceeding under paragraph B.5.g. As set forth in paragraph B.5.g, the petition would have to comply with the requirements of § 2.309 and show that the departure does not comply with paragraph B.5. If on the basis of the petition and any responses thereto, the presiding officer in the proceeding determines that the required showing has been made, the matter would be certified to the Commission for its final determination. In the absence of a proceeding, assertions of nonconformance with paragraph B.5 requirements applicable to Tier 2 departures would be treated as petitions for enforcement action under § 2.206.

Operational Requirements

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

The process in paragraph C.1 for making generic changes to the generic technical specifications in Chapter 16 of the design control document or other operational requirements in the generic design control document would be accomplished by rulemaking and governed by the backfit standards in § 50.109. The determination of whether the generic technical specifications and other operational requirements were completely reviewed and approved in the design certification rulemaking would be based upon the extent to which the NRC reached a safety conclusion in the final safety evaluation report on this matter. If a technical specification or operational requirement was completely reviewed and finalized in the design certification rulemaking, then the requirement of § 50.109 would apply because a position was taken on that safety matter. Generic changes made under paragraph VIII.C.1 would be applicable to all applicants or licensees (refer to paragraph C.2), unless the change is irrelevant because of a plant-specific departure.

Some generic technical specifications 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 APR1400 DC rule must replace the values in brackets with final plant-specific values (refer to guidance provided in Regulatory Guide 1.206, Revision 1, “Applications for Nuclear Power Plants”). The values in brackets are neither part of the DC rule nor are they binding. Therefore, the replacement of bracketed values with final plant-specific values does not require an exemption from the generic technical specifications.

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

Paragraph C.6 states that the generic technical specifications would have no further effect on the plant-specific technical specifications after the issuance of a license that references this appendix and the change process. After a license is issued, the bases for the plant-specific TS would be controlled by the bases change provision set forth in the administrative controls section of the plant-specific technical specifications.

plant-specific departures, as set forth in paragraph X.B.3. The requirements in paragraph X.B.3 for submitting reports will vary according to certain time periods during a facility's lifetime. If a potential applicant for a COL that references this rule decides to depart from the generic design control document prior to submission of the application, then paragraph X.B.3.a will require that the updated design control document 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 design control document 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 frequency is appropriate~~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 support planning for 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 ITAAC under § 52.103(g), as well as any finding on interim operation under Section 189.a(1)(B)(iii) of the Atomic Energy Act of 1954, as amended. Once a facility begins operation (for a COL under 10 CFR part 52, after the Commission has made a finding under § 52.103(g)), the frequency of reporting will be governed by the requirements in paragraph X.B.3.c.

V. APR1400 Standard Design Approval.

authorize the siting, construction, or operation of a facility referencing any particular design; it only codifies the APR1400 design in a rule. The NRC will evaluate the environmental impacts and issue an environmental impact statement as appropriate under NEPA as part of the application for the construction and operation of a facility referencing any particular DC rule.

In addition, consistent with 10 CFR 51.30(d) and 10 CFR 51.32(b), the NRC has prepared a final environmental assessment (ADAMS Accession No. ML18306A607) for the APR1400 design addressing various design alternatives to prevent and mitigate severe accidents. The environmental assessment is based, in part, upon the NRC's review of KEPCO/KHNP's evaluation of various design alternatives to prevent and mitigate severe accidents in APR1400-E-P-NR-14006, Revision 2, ''Severe Accident Mitigation Design Alternatives (SAMDAs) for the APR1400'' (ML18235A158). Based upon review of KEPCO/KHNP's evaluation, the Commission concludes that: (1) KEPCO/KHNP identified a reasonably complete set of potential design alternatives to prevent and mitigate severe accidents for the APR1400 design; (2) none of the potential design alternatives are warranted~~justified~~ on the basis of cost-benefit considerations; and (3) it is unlikely that other design changes would be identified and warranted~~justified~~ during the term of the design certification on the basis of cost-benefit considerations because the estimated core damage frequencies for the APR1400 are very low on an absolute scale. These issues are considered resolved for the APR1400 design. Based on its own independent evaluation, the NRC reached the same conclusion as KEPCO/KHNP that none of the possible candidate design alternatives are potentially cost beneficial for the APR1400 design. This independent evaluation was based on reasonable treatment of costs, benefits, and sensitivities. The NRC concludes that KEPCO/KHNP has adequately identified areas where risk potentially could be reduced in a cost-beneficial manner and adequately assessed whether the implementation of the

1. Paragraph (f)(2)(iv) of 10 CFR 50.34 – Contents of Applications: Technical Information – codified as of **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**.

VI. Issue Resolution

A. The Commission has determined that the structures, systems, and components and design features of the APR1400 design comply with the provisions of the Atomic Energy Act of 1954, as amended, and the applicable regulations identified in Section V of this appendix; and therefore, provide adequate protection to the health and safety of the public. A conclusion that a matter is resolved includes the finding that additional or alternative structures, systems, and components, design features, design criteria, testing, analyses, acceptance criteria, or justifications are not necessary for the APR1400 design.

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

1. All nuclear safety issues associated with the information in the final safety evaluation report, Tier 1, Tier 2, and the rulemaking record for certification of the APR1400 design, with the exception of generic TS and other operational requirements;

2. All nuclear safety and safeguards issues associated with the referenced information in the 53 non-public documents in Tables 1.6-1 and 1.6-2 of Tier 2 of the DCD, which contain sensitive unclassified non-safeguards information (including proprietary information and security-related information) and safeguards information and which, in context, are ~~intended as~~ requirements in the generic DCD for the APR1400 design;

[7590-01-P]

JMB edits

NUCLEAR REGULATORY COMMISSION

10 CFR Part 52

[NRC-2015-0224]

RIN 3150-AJ67

Advanced Power Reactor 1400 (APR1400) Design Certification

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is proposing to amend its regulations to certify the Advanced Power Reactor 1400 (APR1400) standard plant design. ~~This action is necessary so that~~ Aapplicants or licensees intending to construct and operate an APR1400 standard plant design may do so by referencing this design certification (DC) rule. The applicant for the certification of the APR1400 standard plant design is Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. (KEPCO/KHNP).

DATES: Submit comments by **[INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. Comments received after this date will be considered if it is practical to do so, but the NRC is able to ensure consideration only for comments received on or before this date.

AFFIRMATION ITEM

RESPONSE SHEET

TO: Annette Vietti-Cook, Secretary

FROM: Commissioner Burns

SUBJECT: SECY-19-0020: Direct Final Rule - Advanced Power Reactor 1400 Design Certification

Approved _X_ **Disapproved** ___ **Abstain** ___ **Not Participating** ___

COMMENTS: **Below** _X_ **Attached** _X_ **None** ___

I approve the direct final rule and companion proposed rule for the APR1400 design certification for publication in the *Federal Register*, subject to the attached edits. I also approve the related environmental assessment, subject to the attached edits.

Stephen Burns

Signature

19 March 2019

Date

Entered in STARS

Yes _X_

No ___

NUCLEAR REGULATORY COMMISSION

10 CFR Part 52

[NRC-2015-0224]

RIN 3150-AJ67

Advanced Power Reactor 1400 (APR1400) Design Certification

AGENCY: Nuclear Regulatory Commission.

ACTION: Direct final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to certify the Advanced Power Reactor 1400 (APR1400) standard plant design. This action is necessary so that applicants or licensees intending to construct and operate an APR1400 standard plant design may do so by referencing this design certification (DC) rule. The applicant for the certification of the APR1400 standard plant design is Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. (KEPCO/KHNP).

DATES: The final rule is effective **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, unless significant adverse comments are received by **[INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. If the direct final rule is withdrawn as a result of such comments, timely notice of the withdrawal will be published in the *Federal Register*. The incorporation by reference of certain publications listed in this regulation is approved by

the Director of the Office of the Federal Register as of **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE FEDERAL REGISTER]**.

ADDRESSES: You may submit comments by any of the following methods (unless this document describes a different method for submitting comments on a specific subject):

- **Federal Rulemaking Web Site:** Go to https://www.regulations.gov/ and search for Docket ID NRC-2015-0224. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical questions contact the individuals listed in the FOR FURTHER INFORMATION CONTACT section of this document.
- **E-mail comments to:** Rulemaking.Comments@nrc.gov. If you do not receive an automatic e-mail reply confirming receipt, then contact us at 301-415-1677.
- **Fax comments to:** Secretary, U.S. Nuclear Regulatory Commission at 301-415-1101.
- **Mail comments to:** Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff.
- **Hand deliver comments to:** 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. (Eastern Time) Federal workdays; telephone: 301-415-1677.

For additional direction on obtaining information and submitting comments, see "Obtaining Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

FOR FURTHER INFORMATION CONTACT: Yanely Malave-Velez, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-1519, e-mail:

Yanely.Malave@nrc.gov, or William Ward, Office of New Reactors, telephone: 301-415-7038, e-mail: William.Ward@nrc.gov. Both are staff of the U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

TABLE OF CONTENTS:

I. Obtaining Information and Submitting Comments.

A. Obtaining Information

Please refer to Docket ID NRC-2015-0224 when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- **Federal Rulemaking Web Site:** Go to https://www.regulations.gov and search for Docket ID NRC-2015-0224.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at https://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 the Availability of Documents section.

- **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.

B. Submitting Comments

Please include Docket ID NRC-2015-0224 in your comment submission.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at https://www.regulations.gov as well as enter the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment into ADAMS. Comments received after **[INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE FEDERAL REGISTER]** this date will be considered if it is practical to do so, but the NRC is able to ensure consideration only for comments received on or before this date. Comments received on this direct final rule will also be considered to be comments on a companion proposed rule published in the Proposed Rules section of this issue of the *Federal Register*.

II. Rulemaking Procedure.

Because the NRC considers this action to be non-controversial, the NRC is using the "direct final rule procedure" for this rule. The rule will become effective on [INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE FEDERAL REGISTER]. However, if the NRC receives significant adverse comments by [INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE FEDERAL REGISTER], then the NRC will publish a document that withdraws this direct final rule and would subsequently address the comments received in any final rule as a response to the companion proposed rule published in the Proposed Rules section of this issue of the *Federal Register*. Absent significant modifications to the proposed revisions requiring republication, the NRC does not intend to initiate a second comment period on this action.

A significant adverse comment is a comment where the commenter explains why the rule would be inappropriate, including challenges to the rule's underlying premise or approach, or would be ineffective or unacceptable without a change. A comment is adverse and significant if:

1) The comment opposes the rule and provides a reason sufficient to require a substantive response in a notice-and-comment process. For example, a substantive response is required when:

a) The comment causes the NRC to reevaluate (or reconsider) its position or conduct additional analysis;

b) The comment raises an issue serious enough to warrant a substantive response to clarify or complete the record; or

c) The comment raises a relevant issue that was not previously addressed or considered by the NRC.

2) The comment proposes a change or an addition to the rule, and it is apparent that the rule would be ineffective or unacceptable without incorporation of the change or addition.

3) The comment causes the NRC to make a change (other than editorial) to the rule.

For detailed instructions on filing comments, please see the ADDRESSES section in the companion proposed rule published in the Proposed Rules section of this issue of the *Federal Register*.

III. Background.

Part 52 of title 10 of the *Code of Federal Regulations* (10 CFR), “Licenses, Certifications, and Approvals for Nuclear Power Plants,” subpart B, “Standard Design Certifications,” presents the process for obtaining standard design certifications. On December 23, 2014, KEPCO/KHNP submitted its application for certification of the APR1400 standard plant design to the NRC (ADAMS Accession No. ML15006A098). The NRC published a notice of receipt of the application in the *Federal Register* (80 FR

5792; February 3, 2015). KEPCO/KHNP submitted its application in accordance with Subpart B of 10 CFR part 52. On March 12, 2015, the NRC formally accepted the application as a docketed application for design certification (80 FR 13035; March 12, 2015). The pre-application information submitted before the NRC formally accepted the application can be found in ADAMS under Docket No. PROJ0782.

IV. Discussion.

Final Safety Evaluation Report

The NRC issued a final safety evaluation report for the APR1400 design in September 2018. The final safety evaluation report is available in ADAMS under Accession No. ML18087A364. The NRC will publish the final safety evaluation report as a NUREG titled, “Final Safety Evaluation Report Related to the Certification of the Advanced Power Reactor 1400 Standard Design.” The final safety evaluation report is based on the NRC’s review of revision 3 of the APR1400 design control document.

APR1400 DC Rule

The following discussion describes the purpose and key aspects of each section of the APR1400 DC rule. All section and paragraph references are to the provisions being added as appendix F to the regulations in 10 CFR part 52, unless otherwise noted. The NRC has modeled the APR1400 DC rule on existing DC rules, with certain modifications where necessary to account for differences in the APR1400 design documentation, design features, and environmental assessment (including severe accident mitigation design alternatives). As a result, DC rules are standardized to the extent practical.

A. Introduction (Section I)

The purpose of Section I of appendix F to 10 CFR part 52 is to identify the standard plant design that would be approved by this DC rule and the applicant for certification of the standard plant design. Identification of the design certification applicant is necessary to implement appendix F to 10 CFR part 52 for two reasons. First, the implementation of § 52.63(c) depends on whether an applicant for a combined license (COL) contracts with the design certification applicant to obtain the generic design control document and supporting design information. If the COL applicant does not use the design certification applicant to provide the design information and instead uses an alternate nuclear plant vendor, then the COL applicant must meet the requirements in § 52.73. Second, paragraph X.A.1 of the rule would require that the identified design certification applicant maintain the generic design control document throughout the time that appendix F to 10 CFR part 52 may be referenced.

B. Definitions (Section II)

The purpose of Section II of appendix F to 10 CFR part 52 is to define specific terminology with respect to the design certification rule. During development of the first two DC rules, the NRC decided that there would be both generic (master) design control documents maintained by the NRC and the design certification applicant, as well as individual plant-specific design control documents maintained by each applicant or licensee that references a 10 CFR part 52 appendix. This distinction is necessary in order to specify the relevant plant-specific requirements to applicants and licensees referencing appendix F to 10 CFR part 52. In order to facilitate the maintenance of the master design control documents, the NRC requires that each application for a standard design certification be updated to include an electronic copy of the final version of the design control document. The final version is required to incorporate all amendments to the design control document submitted since the original application, as well as any changes directed by the NRC as a result of its review of the original design control document or as a result of public comments. This final version is the master design control document incorporated by reference in the DC rule. The master design control document will be revised as needed to include generic changes to the version of the design control document approved in this design certification rulemaking. These changes would occur as the result of generic rulemaking by the NRC, under the change criteria in Section VIII.

The NRC also requires each applicant and licensee referencing appendix F to 10 CFR part 52 to submit and maintain a plant-specific design control document as part of the COL final safety analysis report. This plant-specific design control document must either include or incorporate by reference the information in the generic design control document. The plant-specific design control document would be updated as necessary to reflect the generic changes to the design control document that the NRC may adopt through rulemaking, plant-specific departures from the generic design control document that the NRC 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. Therefore, the plant-specific design control document functions similar to an updated final safety analysis report because it would provide the most complete and accurate information on a plant's design-basis for that part of the plant that would be within the scope of appendix F to 10 CFR part 52.

The NRC is treating the technical specifications in Chapter 16 of the generic design control document as a special category of information and designating them as generic technical specifications in order to facilitate the special treatment of this information under appendix F to 10 CFR part 52. A COL applicant must submit plant-specific technical specifications that consist of the generic technical specifications, which may be modified as specified in paragraph VIII.C, and the remaining site-specific information needed to complete the technical specifications. The final safety analysis report that is required by § 52.79 will consist of the plant-specific design control document, the site-specific final safety analysis report, and the plant-specific technical specifications.

The terms Tier 1, Tier 2, and COL items (license information) are defined in appendix F to 10 CFR part 52 because these concepts were not envisioned when 10 CFR part 52 was developed. The design certification applicants and the NRC use these terms in implementing the two-tiered rule structure (the DCD is divided into Tiers 1 and 2 to support the rule structure) that was proposed by representatives of the nuclear industry after publication of 10 CFR part 52. The Commission approved the use of a two-tiered rule structure in its staff requirements memorandum, dated February 15, 1991, on SECY-90-377, "Requirements for Design Certification under 10 CFR part 52," dated November 8, 1990 (ADAMS Accession No. ML003707892).

The change process for Tier 2 information is similar to, but not identical to, the change process set forth in 10 CFR 50.59. The regulations in § 50.59 describe when a licensee may make changes to a plant as described in its final safety analysis report without a license amendment. Because of some differences in how the change control requirements are structured in the DC rules, certain definitions contained in § 50.59 are not applicable to 10 CFR part 52 and are not being included in this direct final rule. The NRC is including a definition for a "*Departure from a method of evaluation*" (paragraph II.G), which is appropriate to include in this direct final rule, so that the eight criteria in paragraph VIII.B.5.b will be implemented for new reactors, as intended.

C. Scope and Contents (Section III)

The purpose of Section III of appendix F to 10 CFR part 52 is to describe and define the scope and content of this design certification, how to obtain a copy of the generic design control document, requirements for incorporation by reference of the DC rule, and to set forth how documentation discrepancies or inconsistencies are to be resolved.

Paragraph III.A is the required statement of the Office of the Federal Register for approval of the incorporation by reference of the APR1400 design control document, revision 3. In addition, this paragraph provides the information on how to obtain a copy of the design control document.

Paragraph III.B is the requirement for COL applicants and licensees referencing the APR1400 design control document. The legal effect of incorporation by reference is that the incorporated material has the same legal status as if it were published in the *Code of Federal Regulations*. This material, like any other properly-issued regulation, has the force and effect of law. Tier 1 and Tier 2 information (including the technical and topical reports referenced in Chapter 1), and generic technical specifications have been combined into a single document called the generic design control document, in order to effectively control this information and facilitate its incorporation by reference into the rule. In addition, paragraph III.B clarifies that the conceptual design information and KEPCO/KHNP's evaluation of severe accident mitigation design alternatives are not considered to be part of appendix F to 10 CFR part 52. As provided by § 52.47(a)(24), these conceptual designs are not part of appendix F to 10 CFR part 52 and, therefore, are not applicable to an application that references appendix F to 10 CFR part 52. Therefore, such an applicant would not be required to conform to the conceptual design information that was provided by the design certification applicant. The conceptual design information, which consists of site-specific design features, was required to facilitate the design certification review. Similarly, the severe accident mitigation design alternatives were required to facilitate the environmental assessment.

Paragraphs III.C and III.D set forth the manner by which potential conflicts are to be resolved and identify the controlling document. Paragraph III.C establishes the Tier 1 description in the design control document as controlling in the event of an inconsistency between the Tier 1 and Tier 2 information in the design control document. Paragraph III.D establishes the generic design control document as the controlling document in the event of an inconsistency between the design control document and the final safety evaluation report for the certified standard design.

Paragraph III.E makes it clear that design activities outside the scope of the design certification may be performed using actual site characteristics. This provision applies to site-specific portions of the plant, such as the administration building.

D. Additional Requirements and Restrictions (Section IV)

Section IV of appendix F to 10 CFR part 52 sets forth additional requirements and restrictions imposed upon an applicant who references appendix F to 10 CFR part 52.

Paragraph IV.A sets forth the information requirements for COL applicants and distinguishes between information and documents that must be *included* in the application or the design control document and those which may be *incorporated by reference*. Any incorporation by reference in the application should be clear and should specify the title, date, edition, or version of a document, the page number(s), and table(s) containing the relevant information to be incorporated. The legal effect of such an incorporation by reference into the application is that appendix F to 10 CFR part 52 would be legally binding on the applicant or licensee.

In paragraph IV.B the NRC reserves the right to determine how appendix F to 10 CFR part 52 may be referenced under 10 CFR part 50. This determination may occur in the context of a subsequent rulemaking modifying 10 CFR part 52 or this DC rule, or on a case-by-case basis in the context of a specific application for a 10 CFR part 50 construction permit or operating license. This provision is necessary because the previous DC rules were not implemented in the manner that was originally envisioned at the time that 10 CFR part 52 was issued. The NRC's concern is with the manner by which the inspections, tests, analyses, and acceptance criteria (ITAAC) were developed and the lack of experience with design certifications in a licensing proceeding. Therefore, it is appropriate that the NRC retain some discretion regarding the manner by which appendix F to 10 CFR part 52 could be referenced in a 10 CFR part 50 licensing proceeding.

E. Applicable Regulations (Section V)

The purpose of Section V of appendix F to 10 CFR part 52 is to specify the regulations that were applicable and in effect at the time this design certification was approved. These regulations consist of the technically relevant regulations identified in paragraph V.A, except for the regulations in paragraph V.B that would not be applicable to this certified design.

F. Issue Resolution (Section VI)

The purpose of Section VI of appendix F to 10 CFR part 52 is to identify the scope of issues that would be resolved by the NRC through this rulemaking and, therefore, are "matters resolved" within the meaning and intent of § 52.63(a)(5). The section is divided into five parts: paragraph VI.A identifies the NRC's safety findings in adopting appendix F to 10 CFR part 52, paragraph VI.B identifies the scope and nature of issues that would be resolved by this rulemaking, paragraph VI.C identifies issues~~,~~ ~~which~~that are not resolved by this rulemaking, and paragraph VI.D identifies the issue finality restrictions applicable to the NRC with respect to appendix F to 10 CFR part 52.

Paragraph VI.A describes the nature of the NRC's findings in general terms and makes the findings required by § 52.54 for the NRC's approval of this DC rule.

Paragraph VI.B sets forth the scope of issues that may not be challenged as a matter of right in subsequent proceedings. The introductory phrase of paragraph VI.B clarifies that issue resolution, as described in the remainder of the paragraph, extends to the delineated NRC proceedings referencing appendix F to 10 CFR part 52. The remainder of paragraph VI.B describes the categories of information for which there is issue resolution.

Paragraph VI.C reserves the right of the NRC to impose operational requirements on applicants that reference appendix F to 10 CFR part 52. This provision reflects the fact that only some operational requirements, including portions of the generic technical specifications in Chapter 16 of the design control document, and no operational programs (e.g., operational quality assurance), were completely or comprehensively reviewed by the NRC in this design certification rulemaking proceeding. Therefore, the issue finality provisions of § 52.63 apply only to those operational requirements that either the NRC completely reviewed and approved, or formed the basis of an NRC safety finding of the adequacy of the APR1400, as documented in the NRC's final safety evaluation report. 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 final safety analysis report.[1] 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 § 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.

Also, paragraph VI.C allows the NRC to impose future operational requirements (distinct from design matters) on applicants who reference this design certification. 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 § 52.63. The requirement to perform these testing programs is contained in the Tier 1 information. However, ITAAC 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 ITAAC are satisfied. In the absence of detailed design information to evaluate the need for and develop specific post-fuel load verifications for these matters, the NRC is reserving the right to impose, at the time of COL issuance, license conditions addressing post-fuel load verification activities for portions of the plant within the scope of this design certification.

Paragraph VI.D reiterates the restrictions (contained in Section VIII) placed upon the NRC when ordering generic or plant-specific modifications, changes, or additions to structures, systems, and components, design features, design criteria, and ITAAC within the scope of the certified design.

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

Paragraph VI.E ensures that the NRC will specify at an appropriate time the procedures on how to obtain access to sensitive unclassified and non-safeguards information (SUNSI) and safeguards information (SGI) for the APR1400 DC rule. Access to such information would be for the sole purpose of requesting or participating in certain specified hearings, such as hearings required by § 52.85 or an adjudicatory hearing. For proceedings where the notice of hearing was published before the effective date of the final rule, the Commission's order governing access to SUNSI and SGI shall be used to govern access to such information within the scope of the rulemaking. For proceedings in which the notice of hearing or opportunity for hearing is published after the effective date of the final rule, paragraph VI.E applies and governs access to SUNSI and SGI.

G. Duration of this Appendix (Section VII)

The purpose of Section VII of appendix F to 10 CFR part 52 is, in part, to specify the period during which this design certification may be referenced by an applicant for a COL, under § 52.55, and the period it will remain valid when the design certification is referenced. For example, if an application references this design certification during the 15-year period, then the design certification would be effective until the application is withdrawn or the license issued on that application expires. The NRC intends for appendix F to 10 CFR part 52 to remain valid for the life of the plant that references the design certification to achieve the benefits of standardization and licensing stability. This means that changes to, or plant-specific departures from, information in the plant-specific design control document must be made under the change processes in Section VIII for the life of the plant.

H. Processes for Changes and Departures (Section VIII)

The purpose of Section VIII of appendix F to 10 CFR part 52 is to set forth the processes for generic changes to, or plant-specific departures (including exemptions) from, the design control document. The NRC adopted this restrictive change process in order to achieve a more stable licensing process for applicants and licensees that reference DC rules. Section VIII is divided into three paragraphs, which correspond to Tier 1, Tier 2, and operational requirements.

Generic *changes* (called "modifications" in § 52.63(a)(3)) must be accomplished by rulemaking because the intended subject of the change is this DC rule itself, as is contemplated by § 52.63(a)(1). Consistent with § 52.63(a)(3), any generic rulemaking changes are applicable to all plants, absent circumstances which render the change technically irrelevant. By contrast, plant-specific *departures* could be either an order to one or more applicants or licensees; or an applicant or licensee-initiated departure applicable only to that applicant's or licensee's plant(s), similar to a § 50.59 departure or an exemption. Because these plant-specific departures will result in a design control document that is unique for that plant, Section X would require an applicant or licensee to maintain a plant-specific design control document. For purposes of brevity, the following discussion refers to the processes for both generic changes and plant-specific departures as "change processes." Section VIII refers to an exemption from one or more requirements of this appendix and addresses the criteria for granting an exemption. The NRC cautions that when the exemption involves an underlying substantive requirement (i.e., a requirement outside this appendix), then the applicant or licensee requesting the exemption must demonstrate that an exemption from the underlying applicable requirement meets the criteria of § 52.7 and § 50.12.

For the APR1400 DC review, the staff followed the approach described in SECY-17-0075, "Planned Improvements in Design Certification Tiered Information Designations," (ADAMS Accession No. ML16196A321), to evaluate the applicant's designation of information as Tier 1 or Tier 2 information. Unlike prior design certification applications, this application did not contain any Tier 2* information. As described in SECY-17-0075, in each of the prior design certification rules in 10 CFR Part 52, Appendices A through D, information contained in the DCD was divided into three designations: Tier 1, Tier 2, and Tier 2*. Tier 1 information is the portion of design-related information in the generic DCD that the Commission approves in the Part 52 design certification rule appendices. To change Tier 1 information, NRC approval by rulemaking or approval of an exemption from the certified design rule is required. Tier 2 information is also approved by the Commission in the Part 52 design certification rule appendices, but it is not certified and licensees who reference the design can change this information using the process outlined in Section VIII of the appendices. This change process is similar to that in 10 CFR 50.59 and is generally referred to as the "50.59-like" process. If the criteria in Section VIII are met, a licensee can change Tier 2 information without prior NRC approval. The NRC created a third category, Tier 2*, to address industry requests to minimize the scope of Tier 1 information and provide greater flexibility for making changes. Tier 2* information is included in Tier 2 and has the same safety significance as Tier 1 information, but the NRC decided to provide more flexibility for licensees to change this type of information. In prior design certification rules, Tier 2* is significant information included only in Tier 2 that cannot be changed without prior NRC approval of a license amendment requesting the change.

The applicant included Tier 1 and Tier 2 information in the APR1400 DC application and did not designate or categorize any information as Tier 2* information. Generally, where an applicant includes only Tier 1 and Tier 2 information in an application, the staff will evaluate the Tier 2 information to determine whether any of that information requires NRC approval before it is changed. If the staff identifies any such information in Tier 2, then the staff will request that the applicant revise the application to categorize that information as Tier 1 or Tier 2*, depending on whether the change must be made by approval of a license amendment and an exemption requesting the change (Tier 1), or a license amendment alone (Tier 2*). Because the applicant did not designate any information as Tier 2* information, the staff also considered whether the applicant had included information in Tier 2 that prior DC applicants had identified as Tier 2* but that the NRC staff determined should be categorized as Tier 1. Using requests for additional information, the staff questioned KEPCO/KHNP's categorization of certain information as Tier 2 that past DC applicants had identified as Tier 2* and, in some instances, the staff requested that the applicant revise the application to add that information to Tier 1. This approach required staff and KEPCO/KHNP to identify for each request for additional information the verifiable, important to safety parameters which must be included in Tier 1 to be certified in the rule and verified by ITAAC. After several public meetings, some information was added to or updated in Tier 1 (including modifications to some ITAAC) and the requests for additional information were resolved and closed without the designation of any Tier 2* information.

Of these updates in Tier 1, the most significant concerned the design parameters for the critical structural sections[2] for seismic Category I structures. Past DC

[2] When evaluating the acceptability of the information for seismic Category I structures, the staff's review focuses on a subset of structural information that includes seismic analysis methods, key parameters of seismic Category I structures, and the design of "critical sections." The use of critical sections in the design of safety-related structures is a risk-informed graded approach to achieve the reasonable assurance of safety. In lieu of the safety review of a large number of structural component designs, the staff performs a detailed review of a limited number of critical sections described in the design control document Section 3.8 that contribute to the overall risk significance of the structures. This approach provides the staff with reasonable assurance of the overall safety performance of the structures based on the successful performance of these limited, but critical, risk-significant locations. However, even minor changes to these critical sections could, when applied to the entire safety-related structure, result in significant changes to the overall performance of the structure and, therefore, invalidate the basis for the staff's approval.

applications identified dimensions of length to define critical structural sections as Tier 2* information. During recent construction activities for another design, actual dimensional lengths were found to be outside of their design tolerances. This variance ~~did not necessarily reduce safety but did~~ required additional license amendments to resolve the issue associated with the design tolerances, resulting in increased ~~costs and possible construction schedule impacts~~burden to the licensee without a commensurate safety benefit. For the APR1400 design, the resolution was to revise Tier 1 and the ITAAC for these critical structural sections to use the design load and design load capacity in lieu of dimensions of length, as specific dimensions are not necessarily as important to safety. By focusing on important to safety parameters and including them in ITAAC, rather than in Tier 2* information (thus eliminating the need for Tier 2* information), the staff expects that the need for license amendments to address changes during construction will be greatly reduced while still maintaining reasonable assurance of adequate protection of public health and safety.

Tier 1 information

Paragraph A describes the change process for changes to Tier 1 information that are accomplished by rulemakings that amend the generic design control document and are governed by the standards in § 52.63(a)(1). A generic change under § 52.63(a)(1) will not be made to a certified design while it is in effect unless the change: 1) is necessary for compliance with NRC regulations applicable and in effect at the time the certification was issued; 2) is necessary to provide adequate protection of the public health and safety or common defense and security; 3) reduces unnecessary regulatory burden and maintains protection to public health and safety and common defense and security; 4) provides the detailed design information necessary to resolve select design acceptance criteria; 5) corrects material errors in the certification information;

6) substantially increases overall safety, reliability, or security of a facility and the costs of the change are justified; or 7) contributes to increased standardization of the certification information. The rulemakings must provide for notice and opportunity for public comment on the proposed change, as required by § 52.63(a)(2). The NRC will give consideration as to whether the benefits justify the costs for plants that are already licensed or for which an application for a permit or license is under consideration.

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

Tier 2 information

Paragraph B describes the change processes for the Tier 2 information; which have the same elements as the Tier 1 change process, but some of the standards for plant-specific orders and exemptions would be different. Generic Tier 2 changes would be accomplished by rulemaking that would amend the generic design control document and would be governed by the standards in § 52.63(a)(1). A generic change under § 52.63(a)(1) would not be made to a certified design while it is in effect unless the change: 1) is necessary for compliance with NRC regulations that were applicable and in effect at the time the certification was issued; 2) is necessary to provide adequate protection of the public health and safety or common defense and security; 3) reduces unnecessary regulatory burden and maintains protection to public health and safety and common defense and security; 4) provides the detailed design information necessary to resolve select design acceptance criteria; 5) corrects material errors in the certification information; 6) substantially increases overall safety, reliability, or security of a facility and the costs of the change are justified; or 7) contributes to increased standardization of the certification information.

Departures from Tier 2 would occur in four ways: 1) the NRC may order a plant-specific departure, as set forth in paragraph B.3; 2) an applicant or licensee may request an exemption from a Tier 2 requirement as set forth in paragraph B.4; 3) a licensee may make a departure without prior NRC approval under paragraph B.5; or 4) the licensee may request NRC approval for proposed departures which do not meet the requirements in paragraph B.5 as provided in paragraph B.5.e.

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

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

Paragraph B.5 would allow an applicant or licensee to depart from Tier 2 information, without prior NRC approval, if it does not involve a change to, or departure from, Tier 1 information, technical specifications, or does not require a license amendment under paragraphs B.5.b or c. The technical specifications referred to in B.5.a of this paragraph are the technical specifications in Chapter 16 of the generic design control document, including bases, for departures made prior to the issuance of the COL. After the issuance of the COL, the plant-specific technical specifications would be controlling under paragraph B.5. The requirement for a license amendment in paragraph B.5.b would be similar to the requirement in § 50.59 and would apply to all of the information in Tier 2 except for the information that resolves the severe accident issues.

Paragraph B.5.b addresses information described in the design control document to address aircraft impacts, in accordance with § 52.47(a)(28). Under § 52.47(a)(28), applicants are required to include the information required by § 50.150(b) in their design control document. 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 aircraft impact assessment required by § 50.150(a). The applicant or licensee is also required to describe in the plant-specific design control document how the modified design features and functional capabilities continue to meet the assessment requirements in § 50.150(a)(1). Submittal of this updated information is governed by the reporting requirements in Section X.B.

During an ongoing adjudicatory proceeding (e.g., for issuance of a COL) a party who believes that an applicant or licensee has not complied with paragraph B.5 when departing from Tier 2 information may petition to admit such a contention into the proceeding under paragraph B.5.g. As set forth in paragraph B.5.g, the petition would have to comply with the requirements of § 2.309 and show that the departure does not comply with paragraph B.5. If on the basis of the petition and any responses thereto, the presiding officer in the proceeding determines that the required showing has been made, the matter would be certified to the Commission for its final determination. In the absence of a proceeding, assertions of nonconformance with paragraph B.5 requirements applicable to Tier 2 departures would be treated as petitions for enforcement action under § 2.206.

Operational Requirements

The change process for technical specifications and other operational requirements in the design control document ~~would be~~is set forth in Section VIII, paragraph C. The key to using the change processes described in Section VIII is to determine if the proposed change or departure would require a change to a design feature described in the generic design control document. If a design change is required, then the appropriate change process in paragraph A or B would apply.

However, if a proposed change to the technical specifications or other operational requirements does not require a change to a design feature in the generic design control document, then paragraph C would apply. This change process has elements similar to the Tier 1 and Tier 2 change processes in paragraphs A and B, but with significantly different change standards. Because of the different finality status for technical specifications and other operational requirements, the NRC designated a special category of information, consisting of the technical specifications and other operational requirements, with its own change process in paragraph C. The language in paragraph C also distinguishes between generic (Chapter 16 of the design control document) and plant-specific technical specifications to account for the different treatment and finality consistent with technical specifications before and after a license is issued.

The process in paragraph C.1 for making generic changes to the generic technical specifications in Chapter 16 of the design control document or other operational requirements in the generic design control document would be accomplished by rulemaking and governed by the backfit standards in § 50.109. The determination of whether the generic technical specifications and other operational requirements were completely reviewed and approved in the design certification rulemaking would be based upon the extent to which the NRC reached a safety conclusion in the final safety evaluation report on this matter. If a technical specification or operational requirement was completely reviewed and finalized in the design certification rulemaking, then the requirement of § 50.109 would apply because a position was taken on that safety matter. Generic changes made under paragraph VIII.C.1 would be applicable to all applicants or licensees (refer to paragraph C.2), unless the change is irrelevant because of a plant-specific departure.

Some generic technical specifications 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 APR1400 DC rule must replace the values in brackets with final plant-specific values (refer to guidance provided in Regulatory Guide 1.206, Revision 1, "Applications for Nuclear Power Plants"). The values in brackets are neither part of the DC rule nor are they binding. Therefore, the replacement of bracketed values with final plant-specific values does not require an exemption from the generic technical specifications.

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

Paragraph C.6 states that the generic technical specifications would have no further effect on the plant-specific technical specifications after the issuance of a license that references this appendix~~ and the change process~~. After a license is issued, the bases for the plant-specific TS would be controlled by the bases change provision set forth in the administrative controls section of the plant-specific technical specifications.

I. [RESERVED] (Section IX)

This section is reserved for future use. The matters discussed in this section of earlier design certification rules – inspections, tests, analyses, and acceptance criteria – are now addressed in the substantive provisions of 10 CFR part 52. Accordingly, there is no need to repeat these regulatory provisions in the APR1400 design certification rule. However, this section is being reserved to maintain consistent section numbering with other design certification rules.

J. Records and Reporting (Section X)

The purpose of Section X of appendix F to 10 CFR part 52 is to set forth the requirements that will apply to maintaining records of changes to and departures from the generic design control document, which are to be reflected in the plant-specific design control document. Section X also sets forth the requirements for submitting reports (including updates to the plant-specific design control document) to the NRC. This section of appendix F to 10 CFR part 52 is similar to the requirements for records and reports in 10 CFR part 50, except for minor differences in information collection and reporting requirements.

Paragraph X.A.1 requires that a generic design control document including SUNSI and SGI referenced in the generic design control document be maintained by the applicant for this rule. The generic design control document concept was developed, in part, to meet the requirements for incorporation by reference, including public availability of documents incorporated by reference. However, the SUNSI and SGI could not be included in the generic design control document because they are not publicly available.

Nonetheless, the SUNSI and SGI were reviewed by the NRC and, as stated in paragraph VI.B.2, the NRC would consider the information to be resolved within the meaning of § 52.63(a)(5). Because this information is not in the generic design control document, this information, or its equivalent, is required to be provided by an applicant for a license referencing this DC rule. Only the generic design control document is identified and incorporated by reference into this rule. The generic design control document and the NRC-approved version of the SUNSI and SGI must be maintained by the applicant (KEPCO/KHNP) for the period of time that appendix F to 10 CFR part 52 may be referenced.

Paragraphs X.A.2 and X.A.3 place recordkeeping requirements on the applicant or licensee that reference this design certification so that its plant-specific design control document accurately reflect both generic changes to the generic design control document and plant-specific departures made under Section VIII. The term "plant-specific" is used in paragraph X.A.2 and other sections of appendix F to 10 CFR part 52 to distinguish between the generic design control document that ~~would be~~is being incorporated by reference into appendix F to 10 CFR part 52, and the plant-specific design control document that the COL applicant is required to submit under paragraph IV.A. The requirement to maintain changes to the generic design control document is explicitly stated to ensure that these changes are not only reflected in the generic design control document, which will be maintained by the applicant for the design certification, but also in the plant-specific design control document. Therefore, records of generic changes to the design control document will be required to be maintained by both entities to ensure that both entities have up-to-date design control documents.

Paragraph X.A.4.a requires the DC rule applicant to maintain a copy of the aircraft impact assessment analysis for the term of the certification and renewal. This provision, which is consistent with § 50.150(c)(3), would 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 appendix F to 10 CFR part 52 to maintain a copy of the aircraft impact assessment performed to comply with the requirements of § 50.150(a) throughout the pendency of the application and for the term of the license. This provision is consistent with § 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 § 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 final safety analysis report required by § 52.79 will contain the plant-specific design control document 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 final safety analysis report for a facility (required by § 52.79), but is not part of the plant-specific design control document (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 design control document is part of the final safety analysis report for the facility.

Paragraph X.B.1 requires applicants or licensees that reference this rule to submit reports that describe departures from the design control document 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 will be similar to the requirement in § 50.59(d)(2).

Paragraph X.B.2 requires applicants or licensees that reference this rule to submit updates to the design control document, which include both generic changes and plant-specific departures, as set forth in paragraph X.B.3. The requirements in paragraph X.B.3 for submitting reports will vary according to certain time periods during a facility's lifetime. If a potential applicant for a COL that references this rule decides to depart from the generic design control document prior to submission of the application, then paragraph X.B.3.a will require that the updated design control document 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 design control document 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 support planning for 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 ITAAC under § 52.103(g), as well as any finding on interim operation under Section 189.a(1)(B)(iii) of the Atomic Energy Act of 1954, as amended. Once a facility begins operation (for a COL under 10 CFR part 52, after the Commission has made a finding under § 52.103(g)), the frequency of reporting will be governed by the requirements in paragraph X.B.3.c.

V. APR1400 Standard Design Approval.

On March 8, 2018, as part of the submission of revision 2 of the design control document (ADAMS Accession No. ML18079A146), KEPCO/KHNP requested the NRC provide a final design approval for the APR1400 design. On August 13, 2018, as part of the submission of revision 3 of the design control document (ADAMS Accession No. ML18228A680), KEPCO/KHNP corrected their request for a final design approval to a request for a standard design approval. A standard design approval for the APR1400, revision 3, was issued on September 28, 2018 (ADAMS Accession No. ML18261A187) following the NRC's issuance of the APR1400 final safety evaluation report.

The finality of the standard design approval is discussed in § 52.145. The standard design approval is valid for 15 years from the date of issuance, as described in § 52.147.

VI. Section-by-Section Analysis.

The following paragraphs describe the specific changes in this direct final rule:

Section 52.11, Information collection requirements: OMB approval.

In § 52.11, this direct final rule adds new appendix F to 10 CFR part 52 to the list of information collection requirements in paragraph (b) of this section.

Appendix F to Part 52—Design Certification Rule for the APR1400 Design

This direct final rule adds appendix F to 10 CFR part 52 to incorporate the APR1400 standard plant design into the NRC's regulations. Applicants or licensees intending to construct and operate a plant using an APR1400 design may do so by referencing the DC rule.

VII. Regulatory Flexibility Certification.

Under the Regulatory Flexibility Act (5 U.S.C. 605(b)), the NRC certifies that this direct final rule does not have a significant economic impact on a substantial number of small entities. This direct final rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810).

VIII. Regulatory Analysis.

The NRC has not prepared a regulatory analysis for this direct final rule. The NRC prepares regulatory analyses for rulemakings that establish generic regulatory requirements applicable to all licensees. Design certifications are not generic rulemakings in the sense that design certifications do not establish standards or requirements with which all licensees must comply. Rather, design certifications are NRC approvals of specific nuclear power plant designs by rulemaking, which then may be voluntarily referenced by applicants for COLs. Furthermore, an applicant for a design certification, rather than the NRC initiates design certification rulemakings. Preparation of a regulatory analysis in this circumstance would not be useful because the design to be certified is proposed by the applicant, rather than the NRC. For these reasons, the

NRC concludes that preparation of a regulatory analysis is neither required nor appropriate.

IX. Backfitting and Issue Finality.

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

This initial DC rule 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 DC rule.

This initial DC rule 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 DC rules in appendices A through E to 10 CFR part 52, and no COLs or manufacturing licenses issued by the NRC at this time reference a final APR1400 DC rule.

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.

X. Voluntary Consensus Standards.

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this direct final rule, the NRC certifies the APR1400 standard plant design for use in nuclear power plant licensing under 10 CFR parts 50 or 52. Design certifications are not generic rulemakings establishing a generally applicable standard with which all 10 CFR parts 50 and 52 nuclear power plant licensees must comply. Design certifications are Commission approvals of specific nuclear power plant designs by rulemaking. Furthermore, design certifications are initiated by an applicant for rulemaking, rather than by the NRC. This action does not constitute the establishment of a standard that contains generally applicable requirements.

XI. Plain Writing.

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

XII. Environmental Assessment and Final Finding of No Significant Environmental Impact.

The NRC conducted an environmental assessment (ADAMS Accession No. ML18306A607) and has determined under the National Environmental Policy Act of 1969, as amended (NEPA), and the NRC’s regulations in subpart A of 10 CFR part 51, that this direct final rule, if confirmed, would not be a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement 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 DC rule does not authorize the siting, construction, or operation of a facility referencing any particular design; it only codifies the APR1400 design in a rule. The NRC will evaluate the environmental impacts and issue an environmental impact statement as appropriate under NEPA as part of the application for the construction and operation of a facility referencing any particular DC rule.

In addition, consistent with 10 CFR 51.30(d) and 10 CFR 51.32(b), the NRC has prepared a final environmental assessment (ADAMS Accession No. ML18306A607) for the APR1400 design addressing various design alternatives to prevent and mitigate severe accidents. The environmental assessment is based, in part, upon the NRC's review of KEPCO/KHNP's evaluation of various design alternatives to prevent and mitigate severe accidents in APR1400-E-P-NR-14006, Revision 2, ''Severe Accident Mitigation Design Alternatives (SAMDAs) for the APR1400'' (ML18235A158). Based upon review of KEPCO/KHNP's evaluation, the Commission concludes that: (1) KEPCO/KHNP identified a reasonably complete set of potential design alternatives to prevent and mitigate severe accidents for the APR1400 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 APR1400 are very low on an absolute scale. These issues are considered resolved for the APR1400 design. Based on its own independent evaluation, the NRC reached the same conclusion as KEPCO/KHNP that none of the possible candidate design alternatives are potentially cost beneficial for the APR1400 design. This independent evaluation was based on reasonable treatment of costs, benefits, and sensitivities. The NRC concludes that KEPCO/KHNP has adequately identified areas where risk potentially could be reduced in a cost-beneficial manner and adequately assessed whether the implementation of the identified potential severe accident mitigation design alternatives or candidate design alternatives would be cost-beneficial for the given site parameters. Therefore, the NRC finds that the evaluation performed by KEPCO/KHNP is reasonable and sufficient.

The determination of this environmental assessment is that there will be no significant offsite impact to the public from this action. The environmental assessment is available as indicated under Section XVI, “Availability of Documents.”

XIII. Paperwork Reduction Act Statement.

The burden to the public for the information collection(s) is estimated to average 37 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Further information about information collection requirements associated with this direct final rule can be found in the companion proposed rule published in the Proposed Rule section in this issue of the *Federal Register*.

This direct final rule is being issued prior to approval by the Office of Management and Budget (OMB) of these information collection requirements, which were submitted under OMB control number 3150-0151. When OMB notifies the NRC of its decision, the NRC will publish a document in the *Federal Register* providing notice of the effective date of the information collections or, if approval is denied, providing notice of what action we plan to take.

Send comments on any aspect of these information collections, including suggestions for reducing the burden, to the Information Services Branch, U.S. Nuclear Regulatory Commission, Washington, District of Columbia 20555-0001, or by email to

INFOCOLLECTS.RESOURCE@NRC.GOV; and to OMB Office of Information and Regulatory Affairs (3150-0151), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, District of Columbia 20503; e-mail: oira_submission@omb.eop.gov.

Public Protection Notification.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

XIV. Congressional Review Act.

This final rule is a rule as defined in the Congressional Review Act (5 U.S.C. 801-808). However, the Office of Management and Budget has not found it to be a major rule as defined in the Congressional Review Act.

XV. Agreement State Compatibility.

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

XVI. Availability of Documents.

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

Documents Related to APR1400 Design Certification Rule

DOCUMENT	ADAMS ACCESSION NO. / WEB LINK / *FEDERAL REGISTER* CITATION
SECY-~~XX-XXXX~~19-0020, "Direct Final Rule – ~~APR~~Advanced Power Reactor 1400 Design Certification"	ML18302A069
KEPCO/KHNP Application for Design Certification of the APR1400 Design	ML15006A037
APR1400 Design Control Document, Revision 3	ML18228A667
APR1400 Final Safety Evaluation Report	ML18087A364
APR1400 Environmental Assessment	ML18306A607
APR1400 Standard Design Approval	ML18261A187
Regulatory History of Design Certification[3]	ML003761550
KHNP Topical and Technical Reports	
APR1400-F-A-TR-12004-NP-A, Realistic Evaluation Methodology for Large-Break LOCA of the APR1400, Rev. 1 (August 2018)	ML18233A431
APR1400-F-C-TR-12002-NP-A, KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design, Rev. 0 (April 2017)	ML17115A559
APR1400-F-M-TR-13001-NP-A, PLUS7 Fuel Design for the APR1400, Rev. 1 (August 2018)	ML18232A140
APR1400-K-Q-TR-11005-NP-A, KHNP Quality Assurance Program Description (QAPD) for the APR1400 Design Certification, Rev. 2 (October 2016)	ML18085B044
APR1400-Z-M-TR-12003-NP-A, Fluidic Device Design for the APR1400, Rev. 0 (April 2017)	ML17129A597

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

APR1400-E-I-NR-14001-NP, Human Factors Engineering Program Plan, Rev. 4 (July 2018)	ML18212A345
APR1400-E-I-NR-14002-NP, Operating Experience Review Implementation Plan, Rev. 2 (January 2018)	ML18081A101
APR1400-E-I-NR-14003-NP, Functional Requirements Analysis and Function Allocation Implementation Plan, Rev. 2 (January 2018)	ML18081A091
APR1400-E-I-NR-14004-NP, Task Analysis Implementation Plan, Rev. 3 (May 2018)	ML18178A223
APR1400-E-I-NR-14006-NP, Treatment of Important Human Actions Implementation Plan, Rev. 3 (May 2018)	ML18178A224
APR1400-E-I-NR-14007-NP, Human-System Interface Design Implementation Plan, Rev. 3 (May 2018)	ML18178A212
APR1400-E-I-NR-14008-NP, Human Factors Verification and Validation Implementation Plan, Rev. 3 (May 2018)	ML18178A213
APR1400-E-I-NR-14010-NP, Human Factors Verification and Validation Scenarios, Rev. 2 (January 2018)	ML18081A088
APR1400-E-I-NR-14011-NP, Basic Human-System Interface, Rev. 3 (May 2018)	ML18178A214
APR1400-E-I-NR-14012-NP, Style Guide, Rev. 2 (January 2018)	ML18081A096
APR1400-E-N-NR-14001-NP, Design Features to Address GSI-191, Rev. 3 (February 2018)	ML18057B532
APR1400-E-P-NR-14005-NP, Evaluations and Design Enhancements to Incorporate Lessons Learned from Fukushima Dai-Ichi Nuclear Accident, Rev. 2 (July 2017)	ML18044B042
APR1400-E-S-NR-14004-NP, Evaluation of Effects of HRHF Response Spectra on SSCs, Rev. 3 (December 2017)	ML18078A709
APR1400-E-S-NR-14005-NP, Evaluation of Structure-Soil-Structure Interaction (SSSI) Effects, Rev. 2 (December 2017)	ML18078A699
APR1400-E-S-NR-14006-NP, Stability Check for NI Common Basemat, Rev. 5 (May 2018)	ML18178A221
APR1400-F-A-NR-14001-NP, Small Break LOCA Evaluation Model, Rev. 1 (March 2017)	ML17114A524
APR1400-F-A-NR-14003-NP, Post-LOCA Long Term Cooling Evaluation Model, Rev. 1 (March 2017)	ML17114A526
APR1400-H-N-NR-14012-NP, Mechanical Analysis for New and Spent Fuel Storage Racks, Rev. 3 (August 2017)	ML17244A015
APR1400-K-I-NR-14005-NP, Staffing and Qualifications Implementation Plan, Rev. 1 (February 2018)	ML17094A152
APR1400-K-I-NR-14009-NP, Design Implementation Plan, Rev. 1 (February 2017)	ML17094A153

APR1400-Z-A-NR-14006-NP, Non-LOCA Safety Analysis Methodology, Rev. 1 (February 2017)	ML17094A139
APR1400-Z-A-NR-14007-NP, Mass and Energy Release Methodologies for LOCA and MSLB, Rev. 2 (May 2018)	ML18212A338
APR1400-Z-A-NR-14011-NP, Criticality Analysis of New and Spent Fuel Storage Racks, Rev. 3 (May 2018)	ML18214A561
APR1400-A-N-NR-17001-NP (WCAP-17889-P), Validation of SCALE 6.1.2 with 238-Group ENDF/B-VII.0 Cross Section Library for APR1400 Design Certification, Rev. 0 (June 2014)	ML18044B051
APR1400-Z-J-NR-14001-NP, Safety I&C System, Rev. 3 (May 2018)	ML18212A341
APR1400-Z-J-NR-14003-NP, Software Program Manual, Rev. 3 (May 2018)	ML18214A559
APR1400-E-J-NR-17001-NP, Secure Development and Operational Environment for APR1400 Computer-Based I&C Safety Systems, Rev. 0 (September 2017)	ML18108A470
APR1400-Z-J-NR-14004-NP, Uncertainty Methodology and Application for Instrumentation, Rev. 2 (January 2018)	ML18086B757
APR1400-Z-J-NR-14005-NP, Setpoint Methodology for Safety-Related Instrumentation, Rev. 2 (January 2018)	ML18087A106
APR1400-E-J-NR-14001-NP, Component Interface Module, Rev. 1 (March 2017)	ML17094A131
APR1400-F-C-NR-14003-NP, Functional Design Requirements for a Core Protection Calculator System for APR1400, Rev. 1 (March 2017)	ML17114A522
APR1400-Z-A-NR-14019-NP, CCF Coping Analysis, Rev. 3 (July 2018)	ML18225A340
APR1400-Z-J-NR-14002-NP, Diversity and Defense-in-Depth, Rev. 3 (May 2018)	ML18214A557
APR1400-Z-J-NR-14012-NP, Control System CCF Analysis, Rev. 3 (May 2018)	ML18212A343
APR1400-Z-J-NR-14013-NP, Response Time Analysis of Safety I&C System, Rev. 2 (January 2018)	ML18087A110
APR1400-Z-M-NR-14008-NP, Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown, Rev. 1 (January 2018)	ML18087A112
APR1400-F-C-NR-14001-NP, CPC Setpoint Analysis Methodology for APR1400, Rev. 3 (June 2018)	ML18199A563
APR1400-F-C-NR-14002-NP, Functional Design Requirements for a Core Operating Limit Supervisory System for APR1400, Rev. 1 (February 2017)	ML17094A132
APR1400-E-B-NR-16001-NP, Evaluation of Main Steam and Feedwater Piping Applied to the Graded Approach for the APR1400, Rev. 0 (July 2017)	ML18178A215

APR1400-E-B-NR-16002-NP, Evaluation of Safety Injection and Shutdown Cooling Piping Applied to the Graded Approach for the APR1400, Rev. 1 (May 2018)	ML18178A217
APR1400-H-N-NR-14005-NP, Summary Stress Report for Primary Piping, Rev. 2 (September 2016)	ML18178A218
APR1400-E-X-NR-14001-NP, Equipment Qualification Program, Rev. 4 (July 2018)	ML18214A563
Westinghouse Topical Report	
WCAP-10697-NP-A, Common Qualified Platform Topical Report, Rev. 3 (February 2013)	ML13112A108
Combustion Engineering, Inc. Technical Reports	
CEN-312-NP, Overview Description of the Core Operating Limit Supervisory System (COLSS), Rev. 1-NP (November 1986)	ML19066A067
CEN-310-NP-A, CPC and Methodology for the CPC Improvement Program (April 1986)	ML19066A085

The NRC may post materials related to this document, including public comments, on the Federal Rulemaking Web site at https://www.regulations.gov under Docket ID NRC-2015-0224. The Federal Rulemaking Web site allows you to receive alerts when changes or additions occur in a docket folder. To subscribe: 1) navigate to the docket folder (NRC-2015-0224); 2) click the “Sign up for E-mail Alerts” link; and 3) enter your e-mail address and select how frequently you would like to receive e-mails (daily, weekly, or monthly).

XVII. Procedures for Access to Proprietary and Safeguards Information for Preparation of Comments on the APR1400 Design Certification Rule

This section contains instructions regarding how the non-publicly available documents related to this rule, and specifically those listed in Table 1.6-1 and 1.6-2 beginning on page 1.6-2 of Tier 2 of the DCD, may be accessed by interested persons who wish to comment on the design certification. These documents contain proprietary information and safeguards information (SGI). Requirements for access to SGI are primarily set forth in 10 CFR parts 2 and 73. This section provides information specific to this rule; however, nothing in this section is intended to conflict with the SGI regulations.

Interested persons who desire access to proprietary information on the APR1400 design should first request access to that information from KEPCO/KHNP, the design certification applicant. A request for access should be submitted to the NRC if the applicant does not either grant or deny access by the 10-day deadline described in the following section.

One of the non-publicly available documents, APR1400-E-A-NR-14002-P-SGI, contains both proprietary information and SGI. If you need access to proprietary information in that document in order to develop comments within the scope of this rule, then your request for access should first be submitted to KEPCO/KHNP in accordance with the previous paragraph. By contrast, if you need access to the SGI in order to provide comments, then your request for access to the SGI must be submitted to the NRC as described further in this section. Therefore, if you need access to both proprietary information and SGI in that document then you should request access to the information in separate requests submitted to both KEPCO/KHNP and the NRC.

Submitting a Request to the NRC for Access

Within 10 days after publication of this rule, any individual or entity who believes access to proprietary information or SGI is necessary in order to submit comments on this APR1400 design certification rule may request access to such information. Requests for access to proprietary information or SGI submitted more than 10 days after publication of this document will not be considered absent a showing of good cause for the late filing explaining why the request could not have been filed earlier.

The requestor shall submit a letter requesting permission to access proprietary information and/or SGI to the Office of the Secretary, U.S. Nuclear Regulatory

Commission, Attention: Rulemakings and Adjudications Staff, Washington, DC 20555–0001. The expedited delivery or courier mail address is: Office of the Secretary, U.S. Nuclear Regulatory Commission, Attention: Rulemakings and Adjudications Staff, 11555 Rockville Pike, Rockville, Maryland 20852. The email address for the Office of the Secretary is rulemaking.comments@nrc.gov. The requester must send a copy of the request to the design certification applicant at the same time as the original transmission to the NRC using the same method of transmission. Requests to the applicant must be sent to Yun-Ho Kim, President, KHNP Central Research Institute, 70, 1312-gil, Yuseong-daero, Yuseong-gu, Daejeon, 34101, Korea.

The request must include the following information:

1. The name of this design certification, APR1400 Design Certification; the rulemaking identification number, RIN 3150–AJ67; the rulemaking docket number, NRC–2015–0224; and the *Federal Register* citation for this rule.
2. The name, address, and email or FAX number of the requester.
3. If the requester is an entity, the name of the individual(s) to whom access is to be provided, including the identity of any expert, consultant, or assistant who will aid the requestor in evaluating the information.
4. If the request is for proprietary information, the requester's need for the information in order to prepare meaningful comments on the design certification must be demonstrated. Each of the following areas must be addressed with specificity:
    a. The specific issue or subject matter on which the requester wishes to comment;
    b. An explanation why information which is publicly available is insufficient to provide the basis for developing meaningful comment on the APR1400

design certification rule with respect to the issue or subject matter described in paragraph 4.a. of this section; and

c. The technical competence (demonstrable knowledge, skill, training or education) of the requestor to effectively utilize the requested proprietary information to provide the basis for meaningful comment. Technical competence may be shown by reliance on a qualified expert, consultant, or assistant who satisfies these criteria.

d. A chronology and discussion of the requester's attempts to obtain the information from the design certification applicant, and the final communication from the requester to the applicant and the applicant's response, if any was provided, with respect to the request for access to proprietary information must be submitted.

5. If the request is for SGI, a statement that explains each individual's "need to know" the SGI, as required by 10 CFR 73.2 and 10 CFR 73.22(b)(1). Consistent with the definition of "need to know" as stated in 10 CFR 73.2, the statement must explain:

    a. The specific issue or subject matter on which the requester wishes to comment;

    b. An explanation of why publicly available information is insufficient to provide the basis for developing meaningful comment on the design certification with respect to the issue or subject matter described in paragraph 5.a. of this section and why the SGI requested is indispensable in order to develop meaningful comments;[4] and

[4] Broad SGI requests under these procedures are unlikely to meet the standard for need to know. Furthermore, NRC staff redaction of information from requested documents before their release may be appropriate to comport with this requirement. The procedures in this document do not authorize unrestricted disclosure or less scrutiny of a requester's need to know than ordinarily would be applied in connection with either adjudicatory or non-adjudicatory access to SGI.

c. The technical competence (demonstrable knowledge, skill, training or education) of the requestor to effectively utilize the requested SGI to provide the basis and specificity for meaningful comment. Technical competence may be shown by reliance on a qualified expert, consultant, or assistant who satisfies these criteria.

d. A completed Form SF-85, “Questionnaire for Non-Sensitive Positions,” for each individual who would have access to SGI. The completed Form SF-85 will be used by the Office of Administration to conduct the background check required for access to SGI, as required by 10 CFR part 2, subpart C, and 10 CFR 73.22(b)(2), to determine the requestor's trustworthiness and reliability. For security reasons, Form SF-85 can only be submitted electronically through the electronic questionnaire for investigations processing (e-QIP) website, a secure website that is owned and operated by the Office of Personnel Management. To obtain online access to the form, the requestor should contact the NRC's Office of Administration at 301-415-3710.[5]

e. A completed Form FD-258 (fingerprint card), signed in original ink, and submitted in accordance with 10 CFR 73.57(d). Copies of Form FD-258 may be obtained by writing the Office of Administrative Services, Mail Services Center, Mail Stop P1-37, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by email to MAILSVC.Resource@nrc.gov. The fingerprint card will be used to satisfy the requirements of 10 CFR part 2, subpart C, 10 CFR 73.22(b)(1), and Section 149 of the Atomic Energy Act of 1954, as amended, which

[5] The requester will be asked to provide his or her full name, social security number, date and place of birth, telephone number, and email address.

mandates that all persons with access to SGI must be fingerprinted for an FBI identification and criminal history records check.

f. A check or money order in the amount of $357.00[6] payable to the U.S. Nuclear Regulatory Commission for each individual for whom the request for access has been submitted; and

g. If the requester or any individual who will have access to SGI believes they belong to one or more of the categories of individuals relieved from the criminal history records check and background check requirements, as stated in 10 CFR 73.59, the requester should also provide a statement specifically stating which relief the requester is invoking, and explaining the requester's basis (including supporting documentation) for believing that the relief is applicable. While processing the request, the NRC's Office of Administration, Personnel Security Branch, will make a final determination whether the stated relief applies. Alternatively, the requester may contact the Office of Administration for an evaluation of their status prior to submitting the request. Persons who are not subject to the background check are not required to complete the SF–85 or Form FD–258; however, all other requirements for access to SGI, including the need to know, are still applicable.

Copies of documents and materials required by paragraphs 5.d.–g., as applicable, of this section must be sent to the following address: Office of Administration, U.S. Nuclear Regulatory Commission, Personnel Security Branch, Mail Stop TWF–07D04M, 11555 Rockville Pike, Rockville, MD 20852. These documents and materials

[6] This fee is subject to change pursuant to the Office of Personnel Management's adjustable billing rates.

should not be included with the request letter to the Office of the Secretary, but the request letter should state that the forms and fees have been submitted as required.

To avoid delays in processing requests for access to SGI, all forms should be reviewed for completeness and accuracy (including legibility) before submitting them to the NRC. The NRC will return incomplete or illegible packages to the sender without processing.

Based on an evaluation of the information submitted under paragraphs 4.a.–4.d. or 5.a.–g. of this section, as applicable, the NRC staff will determine within 10 days of receipt of the written access request whether the requester has established a legitimate need for access to proprietary information or need to know the SGI requested.

Determination of Legitimate Need for Access

For proprietary information access requests, if the NRC staff determines that the requester has established a legitimate need for access to proprietary information, the NRC staff will notify the requester in writing that access to proprietary information has been granted. The NRC staff must first notify the design certification applicant of the staff's determination to grant access to the requester not less than 10 days before informing the requester of the staff's decision. If the applicant wishes to challenge the NRC staff's determination, it must follow the procedures in Predisclosure Procedures for Proprietary Information Constituting Trade Secrets or Confidential Commercial or Financial Information of this section. The NRC staff will not provide the requester access to disputed proprietary information to the requester until the procedures are completed as described in Predisclosure Procedures for Proprietary Information Constituting Trade Secrets or Confidential Commercial or Financial Information of this section. The written notification will contain instructions on how the requestor may obtain copies of the requested documents, and any other conditions that may apply to access to those documents. These conditions may include, but are not limited to, the signing of a Non-Disclosure Agreement or Affidavit setting forth terms and conditions to prevent the unauthorized or inadvertent disclosure of proprietary information by each individual who will be granted access.

For requests for access to SGI, if the NRC staff determines that the requester has established a need to know the SGI, the NRC's Office of Administration will then determine, based upon completion of the background check, whether the proposed recipient is trustworthy and reliable, as required for access to SGI by 10 CFR 73.22(b). If the NRC's Office of Administration determines that the individual or individuals are trustworthy and reliable, the NRC will promptly notify the requester in writing. The notification will provide the names of approved individuals as well as the conditions under which the SGI will be provided. Those conditions may include, but are not limited to, the signing of a Non-Disclosure Agreement or Affidavit by each individual who will be granted access to SGI.

Release and Storage of SGI

Prior to providing SGI to the requester, the NRC staff will conduct (as necessary) an inspection to confirm that the recipient's information protection system is sufficient to satisfy the requirements of 10 CFR 73.22. Alternatively, recipients may opt to view SGI at an approved SGI storage location rather than establish their own SGI protection program to meet SGI protection requirements.

Filing of Comments on the APR1400 Design Certification Rule Based on Non-Public Information

Any comments in this rulemaking proceeding that are based upon the disclosed proprietary or SGI information must be filed by the requester no later than 25 days after receipt of (or access to) that information, or the close of the public comment period, whichever is later. The commenter must comply with all NRC requirements regarding the submission of proprietary and SGI to the NRC when submitting comments to the NRC (including marking and transmission requirements).

Review of Denials of Access

If the request for access to proprietary information or SGI is denied by the NRC staff, the NRC staff shall promptly notify the requester in writing, briefly stating the reason or reasons for the denial.

Before the Office of Administration makes a final adverse determination regarding the trustworthiness and reliability of the proposed recipient(s) for access to SGI, the Office of Administration, in accordance with 10 CFR 2.336(f)(1)(iii), must provide the proposed recipient(s) any records that were considered in the trustworthiness and reliability determination, including those required to be provided under 10 CFR 73.57(e)(1), so that the proposed recipient(s) have an opportunity to correct or explain the record.

Appeals from a denial of access must be made to the NRC's Executive Director for Operations (EDO) under 10 CFR 9.29. The decision of the EDO constitutes final agency action under 10 CFR 9.29(d).

Predisclosure Procedures for Proprietary Information Constituting Trade Secrets or Confidential Commercial or Financial Information

The NRC will follow the procedures in 10 CFR 9.28 if the NRC staff determines, under the Determination of Legitimate Need for Access of this section, that access to proprietary information constituting trade secrets or confidential commercial or financial information will be provided to the requester. However, any objection filed by the applicant under 10 CFR 9.28(b) must be filed within 15 days of the NRC staff notice in the Determination of Legitimate Need for Access of this section rather than the 30-day period provided for under 10 CFR 9.28(b). In applying the provisions of 10 CFR 9.28, the applicant for the design certification rule will be treated as the "submitter."

XVIII. Incorporation by Reference—Reasonable Availability to Interested Parties

The NRC is incorporating by reference the APR1400 design control document, revision 3. As described in the "Discussion" section of this document, the generic design control document combined into a single document Tier 1 and Tier 2 information (including the technical and topical reports referenced in Chapter 1), and generic technical specifications in order to effectively control this information and facilitate its incorporation by reference into the rule.

The NRC is required by law to obtain approval for incorporation by reference from the Office of the Federal Register (OFR). The OFR's requirements for incorporation by reference are set forth in 1 CFR part 51. The OFR regulations require an agency to include in a direct final rule a discussion of the ways that the materials the agency incorporates by reference are reasonably available to interested parties or how it worked to make those materials reasonably available to interested parties. The discussion in this section complies with the requirement for direct final rules as set forth in 1 CFR 51.5(b)(2).

The NRC considers "interested parties" to include all potential NRC stakeholders, not only the individuals and entities regulated or otherwise subject to the NRC's regulatory oversight. These NRC stakeholders are not a homogenous group but vary with respect to the considerations for determining reasonable availability. Therefore, the NRC distinguishes between different classes of interested parties for the purposes of determining whether the material is “reasonably available.” The NRC considers the following to be classes of interested parties in NRC rulemakings with regard to the material to be incorporated by reference:

• Individuals and small entities regulated or otherwise subject to the NRC’s regulatory oversight (this class also includes applicants and potential applicants or licenses and other NRC regulatory approvals) and who are subject to the material to be incorporated by reference by rulemaking. In this context, “small entities” has the same meaning as a “small entity” under 10 CFR 2.810.

• Large entities otherwise subject to the NRC’s regulatory oversight (this class also includes applicants and potential applicants for licenses and other NRC regulatory approvals) and who are subject to the material to be incorporated by reference by rulemaking. In this context, “large entities” are those which do not qualify as a “small entity” under 10 CFR 2.810.

• Non-governmental organizations with institutional interests in the matters regulated by the NRC.

• Other Federal agencies, states, local governmental bodies (within the meaning of 10 CFR 2.315(c)).

• Federally-recognized and State-recognized[7] Indian tribes.

• Members of the general public (i.e., individual, unaffiliated members of the public who are not regulated or otherwise subject to the NRC’s regulatory oversight) who may wish to gain access to the materials which the NRC incorporates by reference by rulemaking in order to participate in the rulemaking process.

[7] State-recognized Indian tribes are not within the scope of 10 CFR 2.315(c). However, for purposes of the NRC’s compliance with 1 CFR 51.5, “interested parties” includes a broad set of stakeholders, including State-recognized Indian tribes.

The NRC makes the materials incorporated by reference available for inspection to all interested parties, by appointment, at the NRC Technical Library, which is located at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852; telephone: 301-415-7000; email: Library.Resource@nrc.gov. In addition, as described in Section XVI of this notice, documents related to this rule are available online in the NRC's Agencywide Documents Access and Management System (ADAMS) Public Documents collection at https://www.nrc.gov/reading-rm/adams.html.

The NRC concludes that the materials the NRC is incorporating by reference in this rule are reasonably available to all interested parties because the materials are available to all interested parties in multiple ways and in a manner consistent with their interest in the materials.

List of Subjects in 10 CFR Part 52

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

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; the Nuclear Waste Policy Act of 1982, as amended; and 5 U.S.C. 552 and 553, the NRC is amending10 CFR part 52:

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

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

Authority: Atomic Energy Act of 1954, secs. 103, 104, 147, 149, 161, 181, 182, 183, 185, 186, 189, 223, 234 (42 U.S.C. 2133, 2134, 2167, 2169, 2201, 2231, 2232, 2233, 2235, 2236, 2239, 2273, 2282); Energy Reorganization Act of 1974, secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); 44 U.S.C. 3504 note.

§ 52.11 [Amended]

2. In § 52.11(b), add "F," in alphabetical order to the list of appendices.

3. Add Appendix F to part 52 to read as follows:

Appendix F to Part 52—Design Certification Rule for the APR1400 Design

I. INTRODUCTION

Appendix F constitutes the standard design certification for the Advanced Power Reactor 1400 (APR1400) design, in accordance with 10 CFR part 52, subpart B. The applicant for certification of the APR1400 design is Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. (KEPCO/KHNP).

II. DEFINITIONS

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

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

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

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

1. Definitions and general provisions;

2. Design descriptions;

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

4. Significant site parameters; and

5. Significant interface requirements.

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

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

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

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

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

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

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

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

III. SCOPE AND CONTENTS

A. Incorporation by reference approval. The APR1400 material is approved for incorporation by reference by the Director of the Office of the Federal Register under 5 U.S.C. 552(a) and 1 CFR part 51. You may obtain copies of the generic DCD from Yun-Ho Kim, President, KHNP Central Research Institute, 70, 1312-gil, Yuseong-daero, Yuseong-gu, Daejeon, 34101, Korea. You can view the generic DCD online in the NRC Library at https://www.nrc.gov/reading-rm/adams.html. In ADAMS, search under ADAMS Accession No. ML18228A667. If you do not have access to ADAMS or if you have problems accessing documents located in ADAMS, contact the NRC's Public

Document Room (PDR) reference staff at 1-800-397-4209, 301-415-3747, or by e-mail at PDR.Resource@nrc.gov. Copies of this document are available for examination and copying at the NRC's PDR located at Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852. Copies are also available for examination at the NRC Library located at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852, telephone: 301-415-5610, e-mail: Library.Resource@nrc.gov. All approved material is available for inspection at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202-741-6030 or go to https://www.archives.gov/federal-register/cfr/ibrlocations.html.

1. APR1400 Design Control Document Tier 1 (APR1400-K-X-IT-14001-NP), Revision 3 (August 2018).

2. APR1400 Design Control Document Tier 2 (APR1400-K-X-FS-14002-NP), Revision 3 (August 2018), including:

a. Chapter 1, Introduction and General Description of the Plant.

KHNP Topical and Technical Reports

i. APR1400-F-A-TR-12004-NP-A, Realistic Evaluation Methodology for Large-Break LOCA of the APR1400, Rev. 1 (August 2018).

ii. APR1400-F-C-TR-12002-NP-A, KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design, Rev. 0 (April 2017).

iii. APR1400-F-M-TR-13001-NP-A, PLUS7 Fuel Design for the APR1400, Rev. 1 (August 2018).

iv. APR1400-K-Q-TR-11005-NP-A, KHNP Quality Assurance Program Description (QAPD) for the APR1400 Design Certification, Rev. 2 (October 2016).

v. APR1400-Z-M-TR-12003-NP-A, Fluidic Device Design for the APR1400, Rev. 0 (April 2017).

vi. APR1400-E-I-NR-14001-NP, Human Factors Engineering Program Plan, Rev. 4 (July 2018).

vii. APR1400-E-I-NR-14002-NP, Operating Experience Review Implementation Plan, Rev. 2 (January 2018).

viii. APR1400-E-I-NR-14003-NP, Functional Requirements Analysis and Function Allocation Implementation Plan, Rev. 2 (January 2018).

ix. APR1400-E-I-NR-14004-NP, Task Analysis Implementation Plan, Rev. 3 (May 2018).

x. APR1400-E-I-NR-14006-NP, Treatment of Important Human Actions Implementation Plan, Rev. 3 (May 2018).

xi. APR1400-E-I-NR-14007-NP, Human-System Interface Design Implementation Plan, Rev. 3 (May 2018).

xii. APR1400-E-I-NR-14008-NP, Human Factors Verification and Validation Implementation Plan, Rev. 3 (May 2018).

xiii. APR1400-E-I-NR-14010-NP, Human Factors Verification and Validation Scenarios, Rev. 2 (January 2018).

xiv. APR1400-E-I-NR-14011-NP, Basic Human-System Interface, Rev. 3 (May 2018).

xv. APR1400-E-I-NR-14012-NP, Style Guide, Rev. 2 (January 2018).

xvi. APR1400-E-N-NR-14001-NP, Design Features to Address GSI-191, Rev. 3 (February 2018).

xvii. APR1400-E-P-NR-14005-NP, Evaluations and Design Enhancements to Incorporate Lessons Learned from Fukushima Dai-Ichi Nuclear Accident, Rev. 2 (July 2017).

xviii. APR1400-E-S-NR-14004-NP, Evaluation of Effects of HRHF Response Spectra on SSCs, Rev. 3 (December 2017).

xix. APR1400-E-S-NR-14005-NP, Evaluation of Structure-Soil-Structure Interaction (SSSI) Effects, Rev. 2 (December 2017).

xx. APR1400-E-S-NR-14006-NP, Stability Check for NI Common Basemat, Rev. 5 (May 2018).

xxi. APR1400-F-A-NR-14001-NP, Small Break LOCA Evaluation Model, Rev. 1 (March 2017).

xxii. APR1400-F-A-NR-14003-NP, Post-LOCA Long Term Cooling Evaluation Model, Rev. 1 (March 2017).

xxiii. APR1400-H-N-NR-14012-NP, Mechanical Analysis for New and Spent Fuel Storage Racks, Rev. 3 (August 2017).

xxiv. APR1400-K-I-NR-14005-NP, Staffing and Qualifications Implementation Plan, Rev. 1 (February 2018).

xxv. APR1400-K-I-NR-14009-NP, Design Implementation Plan, Rev. 1 (February 2017).

xxvi. APR1400-Z-A-NR-14006-NP, Non-LOCA Safety Analysis Methodology, Rev. 1 (February 2017).

xxvii. APR1400-Z-A-NR-14007-NP, Mass and Energy Release Methodologies for LOCA and MSLB, Rev. 2 (May 2018).

xxviii. APR1400-Z-A-NR-14011-NP, Criticality Analysis of New and Spent Fuel Storage Racks, Rev. 3 (May 2018).

xxix. APR1400-A-N-NR-17001-NP (WCAP-17889-P), Validation of SCALE 6.1.2 with 238-Group ENDF/B-VII.0 Cross Section Library for APR1400 Design Certification, Rev. 0 (June 2014).

xxx. APR1400-Z-J-NR-14001-NP, Safety I&C System, Rev. 3 (May 2018).

xxxi. APR1400-Z-J-NR-14003-NP, Software Program Manual, Rev. 3 (May 2018).

xxxii. APR1400-E-J-NR-17001-NP, Secure Development and Operational Environment for APR1400 Computer-Based I&C Safety Systems, Rev. 0 (September 2017).

xxxiii. APR1400-Z-J-NR-14004-NP, Uncertainty Methodology and Application for Instrumentation, Rev. 2 (January 2018).

xxxiv. APR1400-Z-J-NR-14005-NP, Setpoint Methodology for Safety-Related Instrumentation, Rev. 2 (January 2018).

xxxv. APR1400-E-J-NR-14001-NP, Component Interface Module, Rev. 1 (March 2017).

xxxvi. APR1400-F-C-NR-14003-NP, Functional Design Requirements for a Core Protection Calculator System for APR1400, Rev. 1 (March 2017).

xxxvii. APR1400-Z-A-NR-14019-NP, CCF Coping Analysis, Rev. 3 (July 2018).

xxxviii. APR1400-Z-J-NR-14002-NP, Diversity and Defense-in-Depth, Rev. 3 (May 2018).

xxxix. APR1400-Z-J-NR-14012-NP, Control System CCF Analysis, Rev. 3 (May 2018).

xl. APR1400-Z-J-NR-14013-NP, Response Time Analysis of Safety I&C System, Rev. 2 (January 2018).

xli. APR1400-Z-M-NR-14008-NP, Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown, Rev. 1 (January 2018).

xlii. APR1400-F-C-NR-14001-NP, CPC Setpoint Analysis Methodology for APR1400, Rev. 3 (June 2018).

xliii. APR1400-F-C-NR-14002-NP, Functional Design Requirements for a Core Operating Limit Supervisory System for APR1400, Rev. 1 (February 2017).

xliv. APR1400-E-B-NR-16001-NP, Evaluation of Main Steam and Feedwater Piping Applied to the Graded Approach for the APR1400, Rev. 0 (July 2017).

xlv. APR1400-E-B-NR-16002-NP, Evaluation of Safety Injection and Shutdown Cooling Piping Applied to the Graded Approach for the APR1400, Rev. 1 (May 2018).

xlvi. APR1400-H-N-NR-14005-NP, Summary Stress Report for Primary Piping, Rev. 2 (September 2016).

xlvii. APR1400-E-X-NR-14001-NP, Equipment Qualification Program, Rev. 4 (July 2018).

Westinghouse Topical Report

xlviii. WCAP-10697-NP-A, Common Qualified Platform Topical Report, Rev. 3 (February 2013).

Combustion Engineering, Inc. Technical Reports

xlix. CEN-312-NP, Overview Description of the Core Operating Limit Supervisory System (COLSS), Rev. 1-NP (November 1986).

l. CEN-310-NP-A, CPC and Methodology for the CPC Improvement Program (April 1986).

b. Chapter 2, Site Characteristics.

c. Chapter 3, Design of Structures, Systems, Components, and Equipment.

d. Chapter 4, Reactor.

e. Chapter 5, Reactor Coolant System and Connecting Systems.

f. Chapter 6, Engineered Safety Features.

g. Chapter 7, Instrumentation and Controls.

h. Chapter 8, Electric Power.

i. Chapter 9, Auxiliary Systems.

j. Chapter 10, Steam and Power Conversion System.

k. Chapter 11, Radioactive Waste Management.

l. Chapter 12, Radiation Protection.

m. Chapter 13, Conduct of Operations.

n. Chapter 14, Verification Programs.

o. Chapter 15, Transient and Accident Analyses.

p. Chapter 16, Technical Specifications.

q. Chapter 17, Quality Assurance and Reliability Assurance.

r. Chapter 18, Human Factors Engineering.

s. Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation.

B. An applicant or licensee referencing this appendix, in accordance with Section IV of this appendix, shall incorporate by reference and comply with the requirements of this appendix except as otherwise provided in this appendix.

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

D. If there is a conflict between the generic DCD and either the application for the design certification of the APR1400 design or the NUREG, "Final Safety Evaluation Report Related to Certification of the APR1400 Standard Design," then the generic DCD controls.

E. Design activities for structures, systems, and components that are entirely 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 that wishes to reference 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 APR1400 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 items; and

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

3. Include, in the plant-specific DCD, the sensitive, unclassified, non-safeguards information (including proprietary information and security-related information) and safeguards information referenced in the APR1400 generic DCD.

4. Include, as part of its application, a demonstration that an entity other than KEPCO/KHNP is qualified to supply the APR1400 design, unless KEPCO/KHNP supplies the design for the applicant's use.

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

V. Applicable Regulations

A. Except as indicated in paragraph B of this section, the regulations that apply to the APR1400 design are in 10 CFR parts 20, 50, 52, 73, and 100, codified as of **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, that are applicable and technically relevant, as described in the final safety evaluation report.

B. The APR1400 design is exempt from portions of the following regulations:

1. Paragraph (f)(2)(iv) of 10 CFR 50.34 – Contents of Applications: Technical Information – codified as of **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**.

VI. ISSUE RESOLUTION

A. The Commission has determined that the structures, systems, and components and design features of the APR1400 design comply with the provisions of the Atomic Energy Act of 1954, as amended, and the applicable regulations identified in Section V of this appendix; and therefore, provide adequate protection to the health and safety of the public. A conclusion that a matter is resolved includes the finding that additional or alternative structures, systems, and components, design features, design criteria, testing, analyses, acceptance criteria, or justifications are not necessary for the APR1400 design.

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

1. All nuclear safety issues associated with the information in the final safety evaluation report, Tier 1, Tier 2, and the rulemaking record for certification of the APR1400 design, with the exception of generic TS and other operational requirements;

2. All nuclear safety and safeguards issues associated with the referenced information in the 53 non-public documents in Tables 1.6-1 and 1.6-2 of Tier 2 of the DCD, which contain sensitive unclassified non-safeguards information (including proprietary information and security-related information) and safeguards information and which, in context, are intended as requirements in the generic DCD for the APR1400 design;

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

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

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

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

7. All environmental issues concerning severe accident mitigation design alternatives associated with the information in the NRC’s environmental assessment for the APR1400 design (ADAMS Accession No. ML18306A607) and APR1400-E-P-NR-14006, Revision 2, ‘‘Severe Accident Mitigation Design Alternatives (SAMDAs) for the APR1400’’ (ML18235A158) for plants referencing this appendix whose site characteristics fall within those site parameters specified in APR1400-E-P-NR-14006.

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, ~~and~~ components, or design features as described in the generic DCD;

2. Provide additional or alternative structures, systems, ~~and~~ 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, ~~and~~ 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 to review portions of the design certification or references containing safeguards information or sensitive unclassified non-safeguards information (including proprietary information, such as trade secrets and commercial or financial information obtained from a person that are privileged or confidential (10 CFR 2.390 and 10 CFR part 9), and security-related information), for the purpose of participating in the hearing required by § 52.85, the hearing provided under § 52.103, or in any other proceeding relating to this appendix, in which interested persons have a right to request an adjudicatory hearing.

VII. DURATION OF THIS APPENDIX

This appendix may be referenced for a period of 15 years from **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, except as provided for in §§ 52.55(b) and 52.57(b). This appendix remains valid for an applicant or licensee who references this appendix until the application is withdrawn or the license expires, including any period of extended operation under a renewed license.

VIII. PROCESSES FOR CHANGES AND DEPARTURES

A. Tier 1 information.

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

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

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

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

B. Tier 2 information.

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

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

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

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

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

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

litigation in the same manner as other issues material to the license hearing. The granting of an exemption to a licensee must be subject to an opportunity for a hearing in the same manner as license amendments.

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

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

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

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

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

(4) Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the plant-specific DCD;

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

(6) Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any evaluated previously in the plant-specific DCD;

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

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

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

(1) There is a substantial increase in the probability of an ex-vessel severe accident such that a particular ex-vessel severe accident previously reviewed and determined to be not credible could become credible; or

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

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

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

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

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

C. Operational requirements.

1. Changes to APR1400 DC generic TS and other operational requirements that were completely reviewed and approved in the design certification rulemaking and do not require a change to a design feature in the generic DCD are governed by the requirements in 10 CFR 50.109. Changes that require a change to a design feature in the generic DCD are governed by the requirements in paragraphs A or B of this section.

2. Changes to APR1400 DC generic TS and other operational requirements are applicable to all applicants who reference this appendix, except those for which the change has been rendered technically irrelevant by action taken under paragraphs C.3 or C.4 of this section.

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

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

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

3. An applicant or licensee who references this appendix shall prepare and maintain written evaluations 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 periods of renewal).

4.a. The applicant for the APR1400 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 periods 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 departure. 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 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 50.71(e) and 52.3.

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

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

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

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

Dated at Rockville, Maryland, this xxth day of Xxxxx, 2019.

For the Nuclear Regulatory Commission.

Annette Vietti-Cook,
Secretary of the Commission.

SGB Edits

ENVIRONMENTAL ASSESSMENT BY THE

U.S. NUCLEAR REGULATORY COMMISSION

RELATING TO THE CERTIFICATION OF THE

APR1400 STANDARD PLANT DESIGN

DOCKET NO. 52-046

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UNITED STATES NUCLEAR REGULATORY COMMISSION

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

RELATING TO THE CERTIFICATION OF THE

APR1400 STANDARD PLANT DESIGN

DOCKET NO. 52-046

The U.S. Nuclear Regulatory Commission (NRC) is issuing a design certification (DC) for the Advanced Power Reactor 1400 (APR1400) standard plant design in response to an application submitted on December 23, 2014, by Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. , hereinafter referred to as KEPCO/KHNP or the applicant. The NRC has decided to adopt DC rules as appendices to Part 52 of Title 10 of the *Code of Federal Regulations* (10 CFR).

The NRC has performed the following environmental assessment of the environmental impacts of the new rule and has documented its finding of no significant impact in accordance with the requirements of 10 CFR 51.21 and the National Environmental Policy Act of 1969, as amended. This environmental assessment addresses the severe accident mitigation design alternatives (SAMDAs) that the NRC has considered for the APR1400 standard plant design. This environmental assessment does not address the site-specific environmental impacts of constructing and operating any facility that references the APR1400 DC at a particular site; those impacts will be evaluated as part of any application(s) for the siting, construction, or operation of such a facility.

As discussed in Section 5.0 of this environmental assessment, the NRC has determined that issuing this DC does not constitute a major Federal action significantly affecting the quality of the human environment. This finding is based on the generic finding made in 10 CFR 51.32(b)(1) that there is no significant environmental impact associated with the certification of a standard plant design under 10 CFR Part 52, Subpart B. The action does not authorize the siting, construction, or operation of a facility using the APR1400 standard plant design. Rather, it merely codifies the APR1400 standard plant design in a rule that could be referenced in a future construction permit (CP), combined license (COL), or operating license (OL) application. Furthermore, because the certification is a rule rather than a physical action, it does not involve commitment of any resources that have alternative uses. As explained in the statements of consideration for “Licenses, Certifications, and Approvals for Nuclear Power Plants; Final Rule,” (72 FR 49352, 49427; August 28, 2007), the 10 CFR 51.32(b)(1) generic finding of no significant impact is legally equivalent to a categorical exclusion. Therefore, the NRC has not prepared an environmental impact statement for the action.

Under 10 CFR 51.30(d), an environmental assessment for a DC must identify the proposed action and is otherwise limited to consideration of the costs and benefits of SAMDAs and the bases for not incorporating SAMDAs in the DC. As discussed in Section 4.0 of this environmental assessment, the NRC also reviewed KEPCO/KHNP’s assessment of SAMDAs that generically apply to the APR1400 standard plant design. The NRC finds that KEPCO/KHNP’s assessment took into consideration a reasonable set of SAMDAs, and that no additional SAMDAs beyond those currently incorporated into the APR1400 standard plant design would be cost-beneficial. This finding is applicable whether SAMDAs are considered at the time of the certification of the APR1400 standard plant design or are considered with respect to licensing a potential future facility referencing the APR1400 DC rule,~~. In Appendix F to 10 CFR Part 52, a~~provided that the plant referencing the APR1400 DC rule ~~should be~~is sited at a location with site characteristics that are encompassed by the postulated site parameters for the DC reference plant site in APR1400-K-X-ER-14001-NP, Revision 2, "Applicant's Environmental Report – Standard Design Certification," issued August 2018 and in the supporting documents.

ENVIRONMENTAL ASSESSMENT

1.0 Identification of the Proposed Action

The proposed action is to certify the APR1400 standard plant design in Appendix F to 10 CFR Part 52. The new rule allows applicants to reference the certified APR1400 standard plant design as part of a COL application under 10 CFR Part 52, or may allow this for a CP application under 10 CFR Part 50.

2.0 Need for the Proposed Action

The proposed action is to issue a rule amending 10 CFR Part 52 to certify the APR1400 standard plant design. The amendment allows an applicant to reference the certified APR1400 standard plant design as part of a COL application under 10 CFR Part 52, or may allow this for a CP application under 10 CFR Part 50. Those portions of the APR1400 standard plant design included in the scope of the certification rulemaking are not subject to further safety review or approval in a COL proceeding. In addition, the DC rule could resolve SAMDAs for any future applications for facilities that reference the certified APR1400 standard plant design.

3.0 Environmental Impact of the Proposed Action

The proposed action constitutes issuance of the DC to 10 CFR Part 52 to certify the APR1400 standard plant design. As stated in 10 CFR 51.32(b)(1), the NRC has determined that there is no significant environmental impact associated with the issuance of a DC. The DC merely codifies the NRC's approval of the APR1400 standard plant design through its final safety evaluation report on the design issued during rulemaking (Agencywide Documents

Access and Management System (ADAMS) Accession No. ML18087A364). Furthermore, because the certification of the design constitutes only a rule rather than a physical action, it would not involve the commitment of any resources that have alternative uses.

As described in Section 4.0 of this environmental assessment, the NRC reviewed various alternative design features for preventing and mitigating severe accidents. The National Environmental Policy Act of 1969, as amended, requires consideration of alternatives to show that the DC rule is the appropriate course of action. The NRC's regulations at 10 CFR 51.55(a) ensure that the design referenced in rulemaking does not exclude any cost beneficial design changes related to the prevention and mitigation of severe accidents.

Through its own independent analysis, the NRC concludes that KEPCO/KHNP adequately considered an appropriate set of SAMDAs and that none met the cost beneficial criteria. Although KEPCO/KHNP made no design changes as a result of considering SAMDAs, KEPCO/KHNP had already incorporated certain features in the APR1400 standard plant design on the basis of probabilistic risk assessment (PRA) results. Section 4.2 of this environmental assessment gives examples of these features. These design features relate to severe accident prevention and mitigation, but were not considered in the SAMDA evaluation because they were already part of the APR1400 standard plant design (refer to Sections 19.2.2 and 19.2.3 of the design control document, "Severe Accident Prevention" and "Severe Accident Mitigation," respectively).

Finally, the DC rule, itself, does not authorize the siting, construction, or operation of a nuclear power plant. An applicant for a CP, early site permit, COL, or OL that references the APR1400 standard plant design will be required to address the environmental impacts of construction and operation for its specific site. The NRC will then evaluate the environmental impacts for that particular site and issue an environmental impact statement in accordance with 10 CFR Part 51. However, the SAMDA analysis that has been completed as part of this environmental assessment can be incorporated by reference into an environmental impact statement related to an application for siting, construction, or operation of a nuclear plant that references the APR1400 standard plant design.

4.0 Severe Accident Mitigation Design Alternatives

The proposed action provides finality in licensing proceedings on an application referencing the APR1400 DC rule and proposing a plant located on a site whose site characteristics fall within the postulated site parameters of the DC referenced plant site (i.e., the Surry Power Station site), as described in APR1400-K-X-ER-14001-NP and supporting documents.

This section provides a summary of the NRC's review of KEPCO/KHNP's Standard Design Certification Environmental Report and the related APR1400 SAMDAs, as provided in APR1400-K-X-ER-14001-NP and supporting documents. The specific details of the NRC's evaluation, summarized in this environmental assessment, are provided in a technical analysis report under ADAMS Accession No. ML18096A697.

4.1. Severe Accident Mitigation Design Alternatives

Consistent with the Commission's objectives of standardization and early resolution of design issues, the SAMDAs are being evaluated as part of the DC for the APR1400 standard plant design. In a 1985 policy statement (50 FR 32138; August 8, 1985), the Commission defined the term severe accident as an event that is beyond the substantial coverage of design-basis events, including events where there is substantial damage to the reactor core (whether or not there are serious offsite consequences). Design-basis events are events analyzed in accordance with the NRC's Standard Review Plan (NUREG-0800) and documented in Chapter 15, "Safety Analysis," of the design control document.

As part of its DC application, KEPCO/KHNP performed a PRA for the APR1400 standard plant design to achieve the following objectives:

- identify the dominant severe accident sequences that account for most of the core damage frequency and associated source terms for the design;
- modify the design, on the basis of PRA insights, to prevent severe accidents or mitigate their consequences and thereby reduce the risk of such accidents; and
- provide a qualitative basis for concluding that all reasonable steps have been taken to reduce the chances of severe accidents to occur and to mitigate the consequences.

KEPCO/KHNP's PRA analysis is described in Chapter 19 of the APR1400 design control document, Revision 3.

The APR1400 Level 1 and Level 2 PRA models quantified six risk categories; three for operations at-power and three for low-power and shutdown operations, namely:

- at-power internal events
- at-power internal flooding events
- at-power internal fire events
- low-power and shutdown internal events
- low-power and shutdown internal flooding events
- low-power and shutdown internal fire events

The risks from other external events, such as high winds, seismic events, external flooding, external fires, etc., were determined by the PRA models to be negligible and were not further analyzed under the SAMDA assessment.

In addition to these safety considerations, applicants for reactor DCs or COLs must also consider alternative design features for severe accidents as part of the NRC's environmental review. These requirements can be summarized as follows:

- Section 52.79(a)(46) requires a COL applicant to describe the plant-specific PRA and its results, with the aim of identifying potential improvements in the reliability of the core and containment heat removal systems that are significant and practical and, which do not impact excessively on the plant.

- Section 51.30(d) requires consideration of SAMDAs in an environmental assessment for a DC, while 10 CFR 51.50(c) sets forth the general requirements for an environmental report accompanying a COL application, including the requirement to evaluate SAMDAs.

Although these requirements are not directly related, they share common purposes, which are to consider alternatives to the proposed design, to evaluate whether potential alternative improvements in the plant design might significantly enhance safety performance during severe accidents, and to prevent reasonable alternatives from being foreclosed.

The NRC has determined that the generic evaluation of SAMDAs for the APR1400 standard design is both practical and warranted for two reasons. First, the design and construction of all plants referencing the certified APR1400 standard plant design will be governed by the rule certifying a single design. Second, the site parameters in APR1400-K-X-ER-14001-NP and supporting documents establish the consequences for a reasonable set of SAMDAs for the APR1400 standard plant design. The low residual risk of the APR1400 standard plant design and the limited potential for further risk reduction provides high confidence that additional cost-beneficial SAMDAs would not be found for sites with characteristics that fit within the site parameter envelope. If an actual characteristic for a particular site does not fall within the postulated site parameters, then SAMDAs that could be affected by the value of the site characteristic must be re-evaluated in the site-specific environmental report and the environmental impact statement prepared in connection with the application. If the actual characteristics of a proposed site fall within the postulated site parameters, then the SAMDA analysis can be incorporated by reference in the site-specific environmental impact statement, and SAMDAs need not be re-evaluated in the environmental impact statement.

4.2. Potential Design Improvements Identified by KEPCO/KHNP

In APR1400-K-X-ER-14001-NP and the supporting documents, the applicant identified 153 candidate design alternatives, or design improvements, based on a review of the standard list of design alternatives provided in Table 14 of Nuclear Energy Institute 05-01A, "Severe Accident Mitigation Alternatives (SAMA) Analysis," and several license renewal environmental reports. KEPCO/KHNP eliminated certain candidate design alternatives from further consideration on the following bases:

- they were already implemented in the APR1400 standard plant design;
- they were not applicable to the APR1400 standard plant design or to the APR1400 DC;
- they had excessive implementation costs; or
- they were of very low benefit.

There were 30 candidate design alternatives that the APR1400 standard plant design already incorporated such as the following:

- installing a gas turbine generator;
- installing an independent active or passive high pressure injection system;
- adding a diverse low pressure injection system;
- improving emergency core cooling system suction strainers;
- adding the ability to manually align the emergency core cooling system recirculation;
- adding the ability to automatically align the emergency core cooling system to recirculation mode upon refueling water storage tank depletion;
- providing an in-containment reactor water storage tank;
- creating a reactor coolant depressurization system; and

- installing an independent reactor coolant pump seal injection system, without a dedicated diesel.

The applicant initially screened the design alternatives based on their analysis in APR1400-K-X-ER-14001-NP, Section 4, "Unmitigated Risk Monetary Value." As described in Section 4.6.1 below, if the implementation costs for a SAMDA candidate exceeded the calculated maximum benefit resulting in a negative Net Present Value, the SAMDA was not considered further. This screening process eliminated 30 potential design alternatives that were identified as being unfeasible due to excessive implementation costs or that provided negligible benefit. Another 54 SAMDA candidates were identified as not applicable to the DC stage of plant development (such as procedural processes, training, or design features not applicable at the DC stage). One potential design alternative was determined to be of very low benefit. The applicant retained the remaining 38 SAMDAs for further assessment in the cost-benefit analysis.

KEPCO/KHNP also applied insights from the APR1400 PRA by applying relevant guidance from Section 5.1, Probabilistic Safety Assessment Importance, in Nuclear Energy Institute 05-01A. First, KEPCO/KHNP identified APR1400-specific dominant risk contributors, derived from the PRA, for further consideration for events. This subset of risk contributors was derived from an importance analysis of core damage cutsets using a Fussell-Vessely importance criterion of greater than 0.5 percent contribution to the total risk (i.e., the total core damage frequency). By applying this criterion, KEPCO/KHNP identified a number of basic events derived from the information in design control document Section 19.1. This process identified basic events in Section 7 of the environmental report that are associated with the six risk categories (see Tables 6a through 6f). Secondly, KEPCO/KHNP applied insights from the APR1400 PRA's top 100 cutsets by identifying any that were not included as part of the Fussell-Vessely importance analysis review. KEPCO/KHNP identified these additional at-power and low-power and shutdown basic events, as provided in Tables 7a through 7f of the environmental report, for further consideration based on the information in design control document Section 19.1.

4.3. NRC Evaluation of Potential Design Improvements

The NRC found that the set of SAMDAs and basic events evaluated by KEPCO/KHNP addressed the major contributor to core damage. KEPCO/KHNP used a systematic and comprehensive process for identifying potential plant improvements for the APR1400 standard plant design, and the set of potential plant improvements identified by KEPCO/KHNP is reasonably comprehensive and, therefore, is acceptable for further evaluation. This included reviewing insights from the plant-specific PRA study as well as assessing severe accident mitigation alternatives (SAMAs) based on accepted industry guidance.

The NRC has concluded that the applicant's assessment of the potential SAMDAs and their impacts on the APR1400 standard plant design is acceptable. The NRC's review did not reveal any additional design alternatives that the applicant should have considered.

4.4. Risk Reduction Potential of SAMDAs

4.4.1. KEPCO/KHNP Evaluation

KEPCO/KHNP evaluated the potential SAMDAs not screened out to assess their potential benefits by using bounding techniques to estimate the possible risk reduction. This is accomplished by associating the basic events identified with a Fussell-Vessely importance of greater than 0.5 percent, and from the top 100 cutsets to a particular SAMDA. This linkage to a SAMDA is provided for each basic event in APR1400-K-X-ER-14001-NP, Sections 7.1 through Section 7.19. The basic event that a potential SAMDA is associated with is also provided in the "Qualitative Screening" column of Table 5 in APR1400-K-X-ER-14001-NP.

Because there are likely several basic events that are considered under a specific SAMDA, KEPCO/KHNP applied a factor of risk reduction based on the sum of Fussell-Vessely importance values for each basic event. KEPCO/KHNP determined the sum of Fussell-Vessely values for each basic event under the six risk categories for a total risk reduction percentage associated with a particular risk category (i.e., at-power internal events, internal flooding, and internal fire; low-power and shutdown internal events, internal flooding, and internal fire). In several basic event cases, KEPCO/KHNP found that there were no Fussell-Vessely importance values; therefore the sum for a risk category would be zero. Section 4.4.2 discusses this assessment further.

4.4.2. NRC Evaluation

The NRC reviewed KEPCO/KHNP's bases for calculating the risk reduction for the various plant improvements and concludes that the rationale and assumptions for estimating risk reduction are reasonable. Specifically, the sum of Fussell-Vessely importance values for risk reductions is acceptable due to its conservatism (i.e., the estimated risk reduction is higher than what would actually be realized). Accordingly, the NRC based its estimates of averted risk for the potential SAMDAs on the resulting APR1400 risk reduction estimates.

4.5. Cost Impacts of Candidate SAMDAs

4.5.1. KEPCO/KHNP Evaluation

In performing the cost benefit analysis of the SAMDAs considered, the cost of enhancement (COE) implementation associated with potential events are estimated from available information related to similar events and components of other nuclear power plant designs. The COE values of the APR1400 SAMDAs are derived from two sources. The first source is the compilation of information from the SAMA[1] analyses performed for the license renewal applications of the presently operating nuclear power plants as documented in the licensees' renewal environmental reports and in the final supplemental environmental impact statements under NUREG-1437. The second source is an assessment by the applicant, as presented in APR1400-K-X-ER-14001-NP. The publicly available license renewal SAMA costs

[1] SAMAs are a subset of SAMDAs, which are attributes for the mitigation of severe accidents of design alternatives, procedural modifications, and training activities.

are full-cost values, while the associated SAMDA costs applied by KEPCO/KHNP were conservatively set to half of the license renewal values based on an assumption that half of the cost would be from engineering and procedure updates. However, it is important to note that for license renewal SAMA evaluations, the full SAMA costs were applied in their cost-benefit analyses.

4.5.2. NRC Evaluation

On the basis of the analyses performed by KEPCO/KHNP, the NRC has concluded that the applicant's estimates of potential costs for the APR1400 SAMDAs are acceptable because the sources for the information and the cost estimates are both reasonable. First, the NRC applied this information in the cost benefit analysis by using half of the SAMDA COE implementation value, as did KEPCO/KHNP for the APR1400 evaluation presented in APR1400-K-X-ER-14001-NP. Second, if SAMDAs were not further screened out based on the conservative assumptions, then the NRC applied the full COE implementation value. This approach facilitates the cost benefit comparisons founded on a graded approach when assessing the averted costs using 7 percent and 3 percent discount rates. This approach is consistent with the guidance in Section 7.2 of Nuclear Energy Institute 05-01A.

4.6. Cost-Benefit Comparison

4.6.1. KEPCO/KHNP Evaluation

The methodology used by KEPCO/KHNP was based primarily on the NRC's guidance for performing cost-benefit analysis outlined in NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook." The guidance involves determining the net present value (NPV) for each SAMDA according to the following formula:

NPV = (APE + AOC + AOE + AOSC) - COE

Where:

NPV = Net present value of current risk ($);

APE = Present value of averted public exposure ($);

AOC = Present value of averted offsite property damage costs ($);

AOE = Present value of averted occupational exposure ($);

AOSC = Present value of averted onsite costs ($); and

COE = Cost of any enhancement implemented to reduce risk ($).

If the net present value of a SAMDA is negative, the cost of implementing the SAMDA is larger than the benefit associated with the SAMDA and it is not cost beneficial. As noted above, 30 candidate SAMDAs were screened out of further analyses for this reason. If the SAMDA benefit exceeds the estimated cost resulting in a positive NPV, the SAMDA is potentially cost-beneficial.

For the representation of the maximum benefit that could be provided, the maximum benefit is calculated to be the sum of the four averted cost categories. It is represented as:

Maximum Benefit = APE + AOC + AOE + AOSC

Table 4.6.2-1 summarizes the applicant's and the NRC's estimates for each of the associated cost elements.

Table 4.6.2-1 Calculated Total Maximum Benefit

Risk Category	KEPCO/KHNP		NRC Staff	
	7%	3%	7%	3%
APE	$49,877	$98,622	$49,872	$98,612
AOC	$63,933	$126,417	$63,941	$126,429
AOE	$3,817	$8,787	$3,818	$8,786
$AOSC_{CD}$	$116,457	$276,642	$191,035	$453,773
$AOSC_{RP}$	$675,084	$1,134,638	$706,726	$1,879,727
Total Maximum Benefit	$909,168	$1,645,106	$1,015,393	$2,567,327

It is important to note that the monetary present value estimate for each risk attribute does not represent the expected reduction in risk resulting from a single accident. Rather, it is the present value of potential losses extending over the projected lifetime (in this case, 60 years) of the facility. Therefore, it reflects the expected annual loss resulting from a single accident, the possibility that such an accident could occur at any time over the licensed life, and the effect of discounting these potential future losses to present value.

The NRC issued Revision 4 of NUREG/BR-0058, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission,” in August 2004 to reflect the agency's policy on discount rates. NUREG/BR-0058, Revision 4, states that two sets of estimates should be developed — one at 3 percent and one at 7 percent. The applicant provided estimates using both discount rates.

KEPCO/KHNP calculated the maximum benefit for at-power internal events, internal flooding events, and internal fire events; along with low-power and shutdown internal events, internal flooding events, and internal fire events for the baseline 7 percent and the sensitivity 3 percent discount rates. The results of the KEPCO/KHNP evaluation are provided in Table 4.6.2-1.

As previously discussed, 38 SAMDAs were carried to the next screening phase. In addition to these remaining SAMDAs, each basic event with a Fussell-Vessely importance of greater than 0.5 percent or part of the top 100 cutsets, if not already included as a basic event, were reviewed to identify any potential SAMDAs. KEPCO/KHNP then related each of the 38 SAMDAs back to one or more of the basic events and assessed the NPV for each basic event with the following steps:

1. Assessed the maximum benefit for each basic event applying conservative assumptions for risk reductions to the AOE and AOSC categories;
2. Conservatively assessed the COE based on half of the SAMDA values obtained from source documents; and
3. Determined the NPV.

For each of the basic events/SAMDAs applying the 7 percent and 3 percent discount rates, KEPCO/KHNP evaluated the NPV and reached a conclusion of whether the enhancements were cost beneficial. KEPCO/KHNP determined, through its ~~SAMDS~~SAMDA analyses, that there were no potentially cost-beneficial enhancements for the 7 percent discount rate analysis. KEPCO/KHNP stated that its sensitivity analysis for the 3 percent discount rate showed a higher maximum benefit over the 7 percent discount rate. However, KEPCO/KHNP concluded that no design changes would provide a positive cost-benefit for either discount rate, if included in the APR1400 standard plant design.

4.6.2. NRC Evaluation

As shown in Table 4.6.2-1, the NRC's confirmatory analysis for the 7 percent and 3 percent discount rates were in general agreement with the applicant for the offsite public exposure (i.e., APE), offsite property damage cost (i.e., AOC), and onsite occupational dose (i.e., AOE) averted costs. The NRC evaluation resulted in higher values than the applicant's evaluation for the onsite cleanup and decontamination (i.e., $AOSC_{CD}$) averted costs, with a similar higher result for the replacement power (i.e., $AOSC_{RP}$) averted costs.

In the $AOSC_{CD}$ evaluation, the NRC adjusted the base averted cost per event provided by NUREG/BR-0184, which was applied by KEPCO/KHNP, to current dollars, resulting in a higher value for the NRC's evaluation. The small difference between the NRC's and the applicant's $AOSC_{RP}$ averted costs for the 7 percent discount rate evaluation is principally due to applying different inflation factors to adjust the base replacement cost to current dollars. For the 3 percent discount rate analysis of the replacement power, KEPCO/KHNP applied a linear interpolation to the NPV for discount rates below 5 percent, as described near the end of Section 5.7.6.2 of NUREG/BR-0184 (see page 5.45 of NUREG/BR-0184). Based on NRC experience in prior regulatory rulemaking analyses, the NRC applied the same replacement cost formula for both the 7 percent and 3 percent discount rates (see the formula in Section 5.7.6.2 of NUREG/BR-0184 on page 5.44). This is viewed by the NRC as being conservative as demonstrated by the larger replacement power averted cost in the NRC evaluation in comparison to the applicant's evaluation.

In its review, the NRC noted that the applicant used two assumed conservatisms in its cost-benefit analysis. The first case of conservatism involved the total averted costs in each analysis, where the applicant did not apply the percent risk reductions for the contribution to total core damage frequency to the population dose (i.e., APE) and offsite property damage (i.e., AOC) costs. The APE and AOC were based on MELCOR Accident Consequence Code System calculations and, thus, are directly tied to the size of a release. As shown by the NRC's 3 percent discount rate analysis compared to the KEPCO/KHNP 3 percent discount rate analysis, applying this reduction to only the onsite exposure (i.e., AOE) and onsite economic costs (i.e., AOSC), results in a conservative result. Namely, it will result in a total maximum benefit that is larger than if the percentage risk reduction is applied to all cost categories. The second conservative assumption involved the use of the determined COE values, as discussed in Section 4.5.1. As assessed by the NRC staff, when the applicant applies only half of the estimated COE value, the final determination of the cost-benefit analysis could more likely provide a positive NPV.

Even with the above discussed differences in the averted cost values, the NRC's confirmatory analysis also reached the same conclusion as KEPCO/KHNP that there were no cost beneficial design alternatives when applying a 7 percent discount rate. This result is the same whether applicant's conservative assumptions were, or were not, applied in the 7 percent discount rate analysis. Based on the NRC's review of the methodology and associated analysis, KEPCO/KHNP's assessment adequately addressed the cost-benefit analysis for the 7 percent discount rate.

For the 3 percent discount rate analysis, the NRC performed a confirmatory calculation to assess the cost-benefits applying the NRC results provided in Table 4.6.2-1, without applying

KEPCO/KHNP's conservative assumptions. Specifically, the NRC also applied the risk reduction percentages to the APE and AOC, since they are also dependent on the released plume, and applied the full COE values. As a result, the NRC determined that there were no cost beneficial design alternatives when applying a 3 percent discount rate.

4.7. Conclusions on SAMDAs

The NRC reviewed KEPCO/KHNP's SAMDA analysis and concludes that the methods used and the implementation of the methods are appropriate. On the basis of the applicant's treatment of SAMDA benefits and costs, the NRC finds that the evaluation performed by KEPCO/KHNP is reasonable and sufficient. Based on its own independent evaluation, the NRC reached the same conclusion as KEPCO/KHNP that none of the possible candidate design alternatives are potentially cost beneficial for the APR1400 standard plant design. This independent evaluation was based on a reasonable treatment of costs, benefits, and sensitivities. Based on the NRC review of KEPCO/KHNP's evaluation, including KEPCO/KHNP's response to requests for additional information, the NRC concludes that KEPCO/KHNP has adequately identified areas where risk potentially could be reduced in a cost-beneficial manner and adequately assessed whether the implementation of the identified potential SAMDAs or candidate design alternatives would be cost-beneficial for the given site parameters.

Because of the magnitude of the negative NPV values, a SAMA based on operational procedures or training for an APR1400 reactor would have to cause a significant effect on the total core damage frequency and/or have a low implementation cost to become cost beneficial. Based on its evaluation, the NRC concludes that it is unlikely that any of the SAMAs based on procedures or training would reduce the risk to be cost beneficial for the given site parameters.

5.0 Finding of No Significant Impact

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC is not required to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the DC rule and the documents referenced in the statement of considerations for the final rule. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, 20852. Publicly available records will be accessible electronically from the ADAMS Public Electronic Reading Room on the NRC Web site at https://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents in ADAMS should contact the NRC PDR reference staff at 1-800-397-4209 or 301-415-4737 or send an e-mail to pdr@nrc.gov.

AFFIRMATION ITEM

RESPONSE SHEET

TO: Annette Vietti-Cook, Secretary

FROM: Commissioner Caputo

SUBJECT: SECY-19-0020: Direct Final Rule - Advanced Power Reactor 1400 Design Certification

Approved X Disapproved ____ Abstain ____ Not Participating ____

COMMENTS: Below ____ Attached X None ____

Entered in STARS

Yes X

No ____

Signature

4/ 17 /2019

Date

Commissioner Caputo's Comments on SECY-19-0020
Direct Final Rule: Advanced Power Reactor 1400 Design Certification

I appreciate the staff's dedication and hard work that have resulted in the completion of the final safety evaluation report and standard design approval for the Advanced Power Reactor 1400 (APR1400) in September 2018, the NRC's first design certification review to be completed within 42 months.

The Commission has before it in SECY-19-0020, the direct final rule APR1400 design certification for consideration. The APR1400 design is based upon the Combustion Engineering System 80+ design, which was certified in May 1997, and is similar to several plants currently operating in the United States. The design incorporates advanced design features to enhance safety and operational flexibility. During the staff's review of the design certification application, there were no safety concerns expressed by members of the public. For these reasons, the staff considers this rulemaking to be noncontroversial. The NRC staff recommends that the Commission approve for publication in the Federal Register the direct final rule and companion proposed rule for the APR1400 design certification, enclosures 1 and 2 of SECY-19-0020.

Based on the staff's completion of the safety evaluation report and standard design approval, the fact that there were no safety concerns expressed by members of the public, and the staff's analysis of options for conducting the rulemaking, I approve the staff's recommendation to publish the direct final rule and the companion proposed rule in the Federal Register.

Although I find no fault in the technical content of the rulemaking package, the processing of the package is another matter. This rulemaking package was provided to the Commission before the Office of the Federal Register (OFR) highlighted the need for significant revisions thereby adding confusion and delays. These substantive deficiencies should have been detected during the staff's review and approval prior to the OFR's review. In the same spirit that we expect our licensees to self-identify performance shortcomings, we should also. Review and approval of documents is a responsibility that should not be taken lightly.

AFFIRMATION ITEM

RESPONSE SHEET

TO: **Annette Vietti-Cook, Secretary**

FROM: **Commissioner Wright**

SUBJECT: **SECY-19-0020: Direct Final Rule - Advanced Power Reactor 1400 Design Certification**

Approved __X__ Disapproved ____ Abstain ____ Not Participating ____

COMMENTS: Below __X__ Attached __X__ None ____

I commend the staff on its thorough technical review of the Advanced Power Reactor 1400 design control document. I also agree with the staff that the direct final rule process is appropriate in this instance, and I appreciate the staff's use of this efficient and effective process of achieving the agency's mission.

I approve publication of the direct final rule and companion proposed rule in the *Federal Register*, subject to the attached edits to the direct final rule. I also approve the staff's environmental assessment, subject to the attached edits.

Entered in STARS

Yes ✓

No ____

____

Signature

3/20/2019

Date

[7590-01-P]

DAW Edits

NUCLEAR REGULATORY COMMISSION

10 CFR Part 52

[NRC-2015-0224]

RIN 3150-AJ67

Advanced Power Reactor 1400 (APR1400) Design Certification

AGENCY: Nuclear Regulatory Commission.

ACTION: Direct final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to certify the Advanced Power Reactor 1400 (APR1400) standard plant design. This action is necessary so that applicants or licensees intending to construct and operate an APR1400 standard plant design may do so by referencing this design certification (DC) rule. The applicant for the certification of the APR1400 standard plant design is Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. (KEPCO/KHNP).

DATES: The final rule is effective **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, unless significant adverse comments are received by **[INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. If the direct final rule is withdrawn as a result of such comments, timely notice of the withdrawal will be published in the *Federal Register*. The incorporation by reference of certain publications listed in this regulation is approved by

5792; February 3, 2015). KEPCO/KHNP submitted its application in accordance with Subpart B of 10 CFR part 52. On March 12, 2015, the NRC formally accepted the application as a docketed application for design certification (80 FR 13035; March 12, 2015). The pre-application information submitted before the NRC formally accepted the application can be found in ADAMS under Docket No. PROJ0782.

IV. Discussion.

Final Safety Evaluation Report

The NRC issued ~~a~~ the final safety evaluation report for the APR1400 design in September 2018. The final safety evaluation report is available in ADAMS under Accession No. ML18087A364. The NRC will publish the final safety evaluation report as a NUREG titled, "Final Safety Evaluation Report Related to the Certification of the Advanced Power Reactor 1400 Standard Design." The final safety evaluation report is based on the NRC's review of revision 3 of the APR1400 design control document.

APR1400 DC Rule

The following discussion describes the purpose and key aspects of each section of the APR1400 DC rule. All section and paragraph references are to the provisions being added as appendix F to the regulations in 10 CFR part 52, unless otherwise noted. The NRC has modeled the APR1400 DC rule on existing DC rules, with certain modifications where necessary to account for differences in the APR1400 design documentation, design features, and environmental assessment (including severe accident mitigation design alternatives). As a result, DC rules are standardized to the extent practical.

A. Introduction (Section I)

The purpose of Section I of appendix F to 10 CFR part 52 is to identify the standard plant design that would be approved by this DC rule and the applicant for certification of the standard plant design. Identification of the design certification applicant is necessary to implement appendix F to 10 CFR part 52 for two reasons. First, the implementation of § 52.63(c) depends on whether an applicant for a combined license (COL) contracts with the design certification applicant to obtain the generic design control document and supporting design information. If the COL applicant does not use the design certification applicant to provide the design information and instead uses an alternate nuclear plant vendor, then the COL applicant must meet the requirements in § 52.73. Second, paragraph X.A.1 of the rule would require that the identified design certification applicant maintain the generic design control document throughout the time that appendix F to 10 CFR part 52 may be referenced.

B. Definitions (Section II)

The purpose of Section II of appendix F to 10 CFR part 52 is to define specific terminology with respect to the design certification rule. During development of the first two DC rules, the NRC decided that there would be both generic (master) design control documents maintained by the NRC and the design certification applicant, as well as individual plant-specific design control documents maintained by each applicant or licensee that references a ~~10 CFR part 52 appendix~~certified standard design. This distinction is necessary in order to specify the relevant plant-specific requirements to applicants and licensees referencing appendix F to 10 CFR part 52. In order to facilitate the maintenance of the master design control documents, the NRC requires that each application for a standard design certification be updated to include an electronic copy of the final version of the design control document. The final version is required to which may be modified as specified in paragraph VIII.C, and the remaining site-specific information needed to complete the technical specifications. The final safety analysis report that is required by § 52.79 will consist of the plant-specific design control document, the site-specific final safety analysis report, and the plant-specific technical specifications.

The terms Tier 1, Tier 2, and COL items (license information) are defined in appendix F to 10 CFR part 52 because these concepts were not envisioned when 10 CFR part 52 was developed. The design certification applicants and the NRC use these terms in implementing the two-tiered rule structure (the DCD is divided into Tiers 1 and 2 to support the rule structure) that was proposed by representatives of the nuclear industry after publication of 10 CFR part 52. The Commission approved the use of a two-tiered rule structure in its staff requirements memorandum, dated February 15, 1991, on SECY-90-377, "Requirements for Design Certification under 10 CFR part 52," dated November 8, 1990 (ADAMS Accession No. ML003707892).

The change process for Tier 2 information is similar to, but not identical to, the change process set forth in 10 CFR 50.59. The regulations in § 50.59 describe when a licensee may make changes to a plant as described in its final safety analysis report without a license amendment. Because of some differences in how the change control requirements are structured in the DC rules, certain definitions contained in § 50.59 are not applicable to 10 CFR part 52 and are not being included in this direct final rule. The NRC is including a definition for a "*Departure from a method of evaluation described in the plant-specific DCD used in establishing the design basis or in the safety analysis*" (paragraph II.~~G~~F), which is appropriate to include in this direct final rule, so that the eight criteria in paragraph VIII.B.5.b will be implemented for new reactors, as intended.

of issues that would be resolved by this rulemaking, paragraph VI.C identifies issues that~~, which~~ are not resolved by this rulemaking, and paragraph VI.D identifies the issue finality restrictions applicable to the NRC with respect to appendix F to 10 CFR part 52.

Paragraph VI.A describes the nature of the NRC's findings in general terms and makes the findings required by § 52.54 for the NRC's approval of this DC rule.

Paragraph VI.B sets forth the scope of issues that may not be challenged as a matter of right in subsequent proceedings. The introductory phrase of paragraph VI.B clarifies that issue resolution, as described in the remainder of the paragraph, extends to the delineated NRC proceedings referencing appendix F to 10 CFR part 52. The remainder of paragraph VI.B describes the categories of information for which there is issue resolution.

Paragraph VI.C reserves the right of the NRC to impose operational requirements on applicants that reference appendix F to 10 CFR part 52. This provision reflects the fact that only some operational requirements, including portions of the generic technical specifications in Chapter 16 of the design control document, and no operational programs (e.g., operational quality assurance), were completely or comprehensively reviewed by the NRC in this design certification rulemaking proceeding. Therefore, the issue finality provisions of § 52.63 apply only to those operational requirements that either the NRC completely reviewed and approved, or formed the basis of an NRC safety finding of the adequacy of the APR1400, as documented in the NRC's final safety evaluation report. 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

H. Processes for Changes and Departures (Section VIII)

The purpose of Section VIII of appendix F to 10 CFR part 52 is to set forth the processes for generic changes to, or plant-specific departures (including exemptions) from, the design control document. The NRC adopted this restrictive change process in order to achieve a more stable licensing process for applicants and licensees that reference DC rules. Section VIII is divided into three paragraphs, which correspond to Tier 1, Tier 2, and operational requirements.

Generic *changes* (called "modifications" in § 52.63(a)(3)) must be accomplished by rulemaking because the intended subject of the change is this DC rule itself, as is contemplated by § 52.63(a)(1). Consistent with § 52.63(a)(3), any generic rulemaking changes are applicable to all plants referencing this DC rule, absent circumstances which render the change technically irrelevant. By contrast, plant-specific *departures* could be either an order to one or more applicants or licensees; or an applicant or licensee-initiated departure applicable only to that applicant's or licensee's plant(s), similar to a § 50.59 departure or an exemption. Because these plant-specific departures will result in a design control document that is unique for that plant, Section X would require an applicant or licensee to maintain a plant-specific design control document. For purposes of brevity, the following discussion refers to the processes for both generic changes and plant-specific departures as "change processes." Section VIII refers to an exemption from one or more requirements of this appendix and addresses the criteria for granting an exemption. The NRC cautions that when the exemption involves an underlying substantive requirement (i.e., a requirement outside this appendix), then the applicant or licensee requesting the exemption must demonstrate that an exemption from the underlying applicable requirement meets the criteria of § 52.7 and § 50.12.

applications identified dimensions of length to define critical structural sections as Tier 2* information. During recent construction activities for another design, actual dimensional lengths were found to be outside of their design tolerances. This variance did not necessarily reduce safety but did require additional license amendments to resolve the issue associated with the design tolerances, resulting in increased costs and possible construction schedule impacts. For the APR1400 design, the resolution was to revise Tier 1 and the ITAAC for these critical structural sections to use the design load and design load capacity in lieu of dimensions of length, as specific dimensions are not necessarily as important to safety. By focusing on important to safety parameters and including them in ITAAC, rather than in Tier 2* information (thus eliminating the need for Tier 2* information), the staff expects that the need for license amendments to address changes during construction will be greatly reduced while still maintaining reasonable assurance of adequate protection of public health and safety.

Tier 1 information

Paragraph A describes the change process for changes to Tier 1 information that are accomplished by rulemakings that amend the generic design control document and are governed by the standards in § 52.63(a)(1). A generic change under § 52.63(a)(1) will not be made to a certified design while it is in effect unless the change: 1) is necessary for compliance with NRC regulations applicable and in effect at the time the certification was issued; 2) is necessary to provide adequate protection of the public health and safety or common defense and security; 3) reduces unnecessary regulatory burden and maintains protection to public health and safety and common defense and security; 4) provides the detailed design information necessary to resolve select design acceptance criteria; 5) corrects material errors in the certification information; 6) substantially increases overall safety, reliability, or security of a facility and the costs demonstrate that the exemption complies with one of the special circumstances in regulations governing specific exemptions in § 50.12(a). In addition, the NRC would not grant requests for exemptions that may result in a significant decrease in the level of safety otherwise provided by the design. However, unlike Tier 1 changes, the special circumstances for the exemption do not have to outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. If the exemption is requested by an applicant for a license, the exemption would be subject to litigation in the same manner as other issues in the licensing hearing, consistent with § 52.63(b)(1). If the exemption is requested by a licensee, then the exemption would be subject to litigation in the same manner as a license amendment.

Paragraph B.5 would allow an applicant or licensee to depart from Tier 2 information, without prior NRC approval, if it does not involve a change to, or departure from, Tier 1 information, technical specifications, or does not require a license amendment under paragraphs B.5.b or c. The technical specifications referred to in B.5.a of this paragraph are the technical specifications in Chapter 16 of the generic design control document, including bases, for departures made prior to the issuance of the COL. After the issuance of the COL, the plant-specific technical specifications would be controlling under paragraph B.5. The requirement for a license amendment in paragraph B.5.b would be similar to the requirement in § 50.59 and would apply to all of the information in Tier 2 except for the information that resolves the severe accident issues.

Paragraph B.5.b addresses information described in the design control document to address aircraft impacts, in accordance with § 52.47(a)(28). Under § 52.47(a)(28), applicants are required to include the information required by § 50.150(b) in their design control document. 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 aircraft impact assessment required by § 50.150(a). The applicant or licensee is also required to describe in the plant-specific design control document how the modified design features and functional capabilities continue to meet the assessment requirements in § 50.150(a)(1). Submittal of this updated information is governed by the reporting requirements in Section X.B.

During an ongoing adjudicatory proceeding (e.g., for issuance of a COL) a party who believes that an applicant or licensee has not complied with paragraph B.5 when departing from Tier 2 information may petition to admit such a contention into the proceeding under paragraph B.5.g. As set forth in paragraph B.5.g, the petition would have to comply with the requirements of § 2.309 and show that the departure does not comply with paragraph B.5. If on the basis of the petition and any responses thereto, the presiding officer in the proceeding determines that the required showing has been made, the matter would be certified to the Commission for its final determination. In the absence of a proceeding, assertions of nonconformance with paragraph B.5 requirements applicable to Tier 2 departures would be treated as petitions for enforcement action under § 2.206.

Operational Requirements

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

The process in paragraph C.1 for making generic changes to the generic technical specifications in Chapter 16 of the design control document or other operational requirements in the generic design control document would be accomplished by rulemaking and governed by the backfit standards in § 50.109. The determination of whether the generic technical specifications and other operational requirements were completely reviewed and approved in the design certification rulemaking would be based upon the extent to which the NRC reached a safety conclusion in the final safety evaluation report on this matter. If a technical specification or operational requirement was completely reviewed and finalized in the design certification rulemaking, then the requirement of § 50.109 would apply because a position was taken on that safety matter. Generic changes made under paragraph VIII.C.1 would be applicable to all applicants or licensees (refer to paragraph C.2), unless the change is irrelevant because of a plant-specific departure.

Some generic technical specifications 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 APR1400 DC rule must replace the values in brackets with final plant-specific values (refer to guidance provided in Regulatory Guide 1.206, Revision 1, "Applications for Nuclear Power Plants"). The values in brackets are neither part of the DC rule nor are they binding. Therefore, the replacement of bracketed values with final plant-specific values does not require an exemption from the generic technical specifications.

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

Paragraph C.6 states that the generic technical specifications would have no further effect on the plant-specific technical specifications after the issuance of a license that references this appendix ~~and the change process~~. After a license is issued, the bases for the plant-specific ~~TS~~ technical specifications would be controlled by the bases

Nonetheless, the SUNSI and SGI were reviewed by the NRC and, as stated in paragraph VI.B.2, the NRC would consider the information to be resolved within the meaning of § 52.63(a)(5). Because this information is not in the generic design control document, this information, or its equivalent, is required to be provided by an applicant for a license referencing this DC rule. Only the generic design control document is identified and incorporated by reference into this rule. The generic design control document and the NRC-approved version of the SUNSI and SGI must be maintained by the applicant (KEPCO/KHNP) for the period of time that appendix F to 10 CFR part 52 may be referenced.

Paragraphs X.A.2 and X.A.3 place recordkeeping requirements on the applicant or licensee that references this design certification so that its plant-specific design control document accurately reflects both generic changes to the generic design control document and plant-specific departures made under Section VIII. The term "plant-specific" is used in paragraph X.A.2 and other sections of appendix F to 10 CFR part 52 to distinguish between the generic design control document that ~~would be~~is being incorporated by reference into appendix F to 10 CFR part 52, and the plant-specific design control document that the COL applicant is required to submit under paragraph IV.A. The requirement to maintain changes to the generic design control document is explicitly stated to ensure that these changes are not only reflected in the generic design control document, which will be maintained by the applicant for the design certification, but also in the plant-specific design control document. Therefore, records of generic changes to the design control document will be required to be maintained by both entities to ensure that both entities have up-to-date design control documents.

Paragraph X.A.4.a requires the DC rule applicant to maintain a copy of the aircraft impact assessment analysis for the term of the certification and any renewal.

This provision, which is consistent with § 50.150(c)(3), would 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 appendix F to 10 CFR part 52 to maintain a copy of the aircraft impact assessment performed to comply with the requirements of § 50.150(a) throughout the pendency of the application and for the term of the license and any renewal. This provision is consistent with § 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 § 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 final safety analysis report required by § 52.79 will contain the plant-specific design control document 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 final safety analysis report for a facility (required by § 52.79), but is not part of the plant-specific design control document (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 design control document is part of the final safety analysis report for the facility.

Paragraph X.B.1 requires applicants or licensees that reference this rule to submit reports that describe departures from the design control document and include a summary of the written evaluations. The requirement for the written evaluations is set forth in paragraph X.A.~~1~~3. The frequency of the report submittals is set forth in and although an Agreement State may not adopt program elements reserved to the NRC, it may wish to inform its licensees of certain requirements by a mechanism that is consistent with a particular State's administrative procedure laws, but does not confer regulatory authority on the State.

XVI. Availability of Documents.

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

Documents Related to APR1400 Design Certification Rule

DOCUMENT	**ADAMS ACCESSION NO. / WEB LINK / *FEDERAL REGISTER* CITATION**
SECY-19~~XX-XXXX~~0020, "Direct Final Rule – Advanced Power Reactor 1400 Design Certification"	ML18302A069
KEPCO/KHNP Application for Design Certification of the APR1400 Design	ML15006A037
APR1400 Design Control Document, Revision 3	ML18228A667
APR1400 Final Safety Evaluation Report	ML18087A364
APR1400 Environmental Assessment	ML18306A607
APR1400 Standard Design Approval	ML18261A187
Regulatory History of Design Certification[3]	ML003761550
KHNP Topical and Technical Reports	
APR1400-F-A-TR-12004-NP-A, Realistic Evaluation Methodology for Large-Break LOCA of the APR1400, Rev. 1 (August 2018)	ML18233A431
APR1400-F-C-TR-12002-NP-A, KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design, Rev. 0 (April 2017)	ML17115A559
APR1400-F-M-TR-13001-NP-A, PLUS7 Fuel Design for the APR1400, Rev. 1 (August 2018)	ML18232A140
APR1400-K-Q-TR-11005-NP-A, KHNP Quality Assurance Program Description (QAPD) for the APR1400 Design Certification, Rev. 2 (October 2016)	ML18085B044
APR1400-Z-M-TR-12003-NP-A, Fluidic Device Design for the APR1400, Rev. 0 (April 2017)	ML17129A597

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

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

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

1. Definitions and general provisions;

2. Design descriptions;

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

4. Significant site parameters; and

5. Significant interface requirements.

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

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

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

1. Paragraph (f)(2)(iv) of 10 CFR 50.34 – Contents of Applications: Technical Information – codified as of **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**.

VI. ISSUE RESOLUTION

A. The Commission has determined that the structures, systems, and components and design features of the APR1400 design comply with the provisions of the Atomic Energy Act of 1954, as amended, and the applicable regulations identified in Section V of this appendix; and therefore, provide adequate protection to the health and safety of the public. A conclusion that a matter is resolved includes the finding that additional or alternative structures, systems, and components, design features, design criteria, testing, analyses, acceptance criteria, or justifications are not necessary for the APR1400 design.

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

1. All nuclear safety issues associated with the information in the final safety evaluation report, Tier 1, Tier 2, and the rulemaking record for certification of the APR1400 design, with the exception of generic TS and other operational requirements;

2. All nuclear safety and safeguards issues associated with the referenced information in the 53 non-public documents in Tables 1.6-1 and 1.6-2 of Tier 2 of the DCD, which contain sensitive unclassified non-safeguards information (including proprietary information and security-related information) and safeguards information and which, in context, are intended as requirements in the generic DCD for the APR1400 design;

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

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

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

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

7. All environmental issues concerning severe accident mitigation design alternatives associated with the information in the NRC's environmental assessment for the APR1400 design (ADAMS Accession No. ML18306A607) and APR1400-E-P-NR-14006, Revision 2, "Severe Accident Mitigation Design Alternatives (SAMDAs) for the APR1400" (ML18235A158) for plants referencing this appendix whose site characteristics fall within those site parameters specified in APR1400-E-P-NR-14006.

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, and components or design features as described in the generic DCD;

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

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

C. Operational requirements.

1. Changes to APR1400 DC generic TS and other operational requirements that were completely reviewed and approved in the design certification rulemaking and do not require a change to a design feature in the generic DCD are governed by the requirements in 10 CFR 50.109. Changes that require a change to a design feature in the generic DCD are governed by the requirements in paragraphs A or B of this section.

2. Changes to APR1400 DC generic TS and other operational requirements are applicable to all applicants who reference this appendix, except those for which the change has been rendered technically irrelevant by action taken under paragraphs C.3 or C.4 of this section.

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ENVIRONMENTAL ASSESSMENT BY THE

U.S. NUCLEAR REGULATORY COMMISSION

RELATING TO THE CERTIFICATION OF THE

APR1400 STANDARD DESIGN

DOCKET NO. 52-046

UNITED STATES NUCLEAR REGULATORY COMMISSION

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

RELATING TO THE CERTIFICATION OF THE

APR1400 STANDARD DESIGN

DOCKET NO. 52-046

The U.S. Nuclear Regulatory Commission (NRC) is issuing a design certification (DC) for the Advanced Power Reactor 1400 (APR1400) standard plant design in response to an application submitted on December 23, 2014, by Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. , hereinafter referred to as KEPCO/KHNP or the applicant. The NRC has decided to adopt DC rules as appendices to Part 52 of Title 10 of the *Code of Federal Regulations* (10 CFR).

The NRC has performed the following environmental assessment of the environmental impacts of the new rule and has documented its finding of no significant impact in accordance with the requirements of 10 CFR 51.21 and the National Environmental Policy Act of 1969, as amended (NEPA). This environmental assessment addresses the severe accident mitigation design alternatives (SAMDAs) that the NRC has considered for the APR1400 standard plant design. This environmental assessment does not address the site-specific environmental impacts of constructing and operating any facility that references the APR1400 DC at a particular site; those impacts will be evaluated as part of any application(s) for the siting, construction, or operation of such a facility.

characteristics that are encompassed by the postulated site parameters for the DC reference plant site in APR1400-K-X-ER-14001-NP, Revision 2, "Applicant's Environmental Report – Standard Design Certification," issued August 2018 and in the supporting documents.

ENVIRONMENTAL ASSESSMENT

1.0 Identification of the Proposed Action

The proposed action is to certify the APR1400 standard plant design in Appendix F to 10 CFR Part 52. The new rule allows applicants to reference the certified APR1400 standard plant design as part of a COL application under 10 CFR Part 52, or may allow this for a CP application under 10 CFR Part 50.

2.0 Need for the Proposed Action

The proposed action is to issue a rule amending 10 CFR Part 52 to certify the APR1400 standard plant design. The amendment allows an applicant to reference the certified APR1400 standard plant design as part of a COL application under 10 CFR Part 52, or may allow this for a CP application under 10 CFR Part 50. Those portions of the APR1400 standard plant design included in the scope of the certification rulemaking are not subject to further safety review or approval in a COL proceeding. In addition, the DC rule could resolve SAMDAs for any future applications for facilities that reference the certified APR1400 standard plant design.

3.0 Environmental Impact of the Proposed Action

The proposed action constitutes issuance of the DC as an amendment to 10 CFR Part 52 to certify the APR1400 standard plant design. As stated in 10 CFR 51.32(b)(1), the NRC has determined that there is no significant environmental impact associated with the issuance of a DC. The DC merely codifies the NRC's approval of the APR1400 standard plant design through its final safety evaluation report on the design issued during rulemaking (Agencywide Documents Access and Management System (ADAMS) Accession No.

ML18087A364). Furthermore, because the certification of the design constitutes only a rule rather than a physical action, it would not involve the commitment of any resources that have alternative uses.

As described in Section 4.0 of this environmental assessment, the NRC reviewed various alternative design features for preventing and mitigating severe accidents. ~~The National Environmental Policy Act of 1969, as amended~~NEPA~~,~~ requires consideration of alternatives to show that the DC rule is the appropriate course of action. The NRC's regulations at 10 CFR 51.55(a) ensure that the design referenced in rulemaking does not exclude any cost beneficial design changes related to the prevention and mitigation of severe accidents.

Through its own independent analysis, the NRC concludes that KEPCO/KHNP adequately considered an appropriate set of SAMDAs and that none met the cost beneficial criteria. Although KEPCO/KHNP made no design changes as a result of considering SAMDAs, KEPCO/KHNP had already incorporated certain features in the APR1400 standard plant design on the basis of probabilistic risk assessment (PRA) results. Section 4.2 of this environmental assessment gives examples of these features. These design features relate to severe accident prevention and mitigation, but they were not considered in the SAMDA evaluation because they were already part of the APR1400 standard plant design (refer to Sections 19.2.2 and 19.2.3 of the design control document, "Severe Accident Prevention" and "Severe Accident Mitigation," respectively).

Finally, the DC rule, itself, does not authorize the siting, construction, or operation of a nuclear power plant. An applicant for a CP, early site permit, COL, or OL that references the APR1400 standard plant design will be required to address the environmental impacts of construction and operation for its specific site. The NRC will then evaluate the environmental impacts for that particular site and issue an environmental impact statement in accordance with 10 CFR Part 51. However, the SAMDA analysis that has been completed as part of this environmental assessment can be incorporated by reference into an environmental impact statement related to an application for siting, construction, or operation of a nuclear plant that references the APR1400 standard plant design.

4.0 Severe Accident Mitigation Design Alternatives

The proposed action provides finality in licensing proceedings on an application referencing the APR1400 DC rule and proposing a plant located on a site whose site characteristics fall within the postulated site parameters of the DC referenced plant site (i.e., the Surry Power Station site), as described in APR1400-K-X-ER-14001-NP and supporting documents.

This section provides a summary of the NRC's review of KEPCO/KHNP's Standard Design Certification Environmental Report and the related APR1400 SAMDAs, as provided in APR1400-K-X-ER-14001-NP and supporting documents. The specific details of the NRC's evaluation, summarized in this environmental assessment, are provided in a technical analysis report under ADAMS Accession No. ML18096A697.

4.1. Severe Accident Mitigation Design Alternatives

Consistent with the Commission's objectives of standardization and early resolution of design issues, the SAMDAs are being evaluated as part of the DC for the APR1400 standard plant design. In a 1985 policy statement (50 FR 32138; August 8, 1985), the Commission defined the term severe accident as an event that is beyond the substantial coverage of design-basis events, including events where there is substantial damage to the reactor core (whether or not there are serious offsite consequences). Design-basis events are events analyzed in accordance with the NRC's Standard Review Plan (NUREG-0800) and documented in Chapter 15, "~~Safety Analysis~~Transient and Accident Analysis," of the design control document.

- Section 52.79(a)(46) requires a COL applicant to describe the plant-specific PRA and its results, with the aim of identifying potential improvements in the reliability of the core and containment heat removal systems that are significant and practical and, which do not impact excessively on the plant.
- Section 51.30(d) requires consideration of SAMDAs in an environmental assessment for a DC, while 10 CFR 51.50(c) sets forth the general requirements for an environmental report accompanying a COL application, including the requirement to evaluate SAMDAs.

Although these requirements are not directly related, they share common purposes:~~,~~ ~~which are~~ to consider alternatives to the proposed design, to evaluate whether potential alternative improvements in the plant design might significantly enhance safety performance during severe accidents, and to prevent reasonable alternatives from being foreclosed.

The NRC has determined that the generic evaluation of SAMDAs for the APR1400 standard design is both practical and warranted for two reasons. First, the design and construction of all plants referencing the certified APR1400 standard plant design will be governed by the rule certifying a single design. Second, the site parameters in APR1400-K-X-ER-14001-NP and supporting documents establish the consequences for a reasonable set of SAMDAs for the APR1400 standard plant design. The low residual risk of the APR1400 standard plant design and the limited potential for further risk reduction provides high confidence that additional cost-beneficial SAMDAs would not be found for sites with characteristics that fit within the site parameter envelope. If an actual characteristic for a particular site does not fall within the postulated site parameters, then SAMDAs that could be affected by the value of the site characteristic must be re-evaluated in the site-specific environmental report and the environmental impact statement prepared in connection with the application. If the actual characteristics of a proposed site fall within the postulated site report, for further consideration based on the information in design control document Section 19.1.

4.3. NRC Evaluation of Potential Design Improvements

The NRC found that the set of SAMDAs and basic events evaluated by KEPCO/KHNP addressed the major contributor to core damage. KEPCO/KHNP used a systematic and comprehensive process for identifying potential plant improvements for the APR1400 standard plant design, and the set of potential plant improvements identified by KEPCO/KHNP is reasonably comprehensive and, therefore, is acceptable for further evaluation. This included reviewing insights from the plant-specific PRA study as well as assessing severe accident mitigation alternatives (SAMAs) based on accepted industry guidance.

The NRC has concluded that the applicant's assessment of the potential SAMDAs and their impacts on the APR1400 standard plant design is acceptable. The NRC's review did not reveal any additional design alternatives that the applicant should have considered.

4.4. Risk Reduction Potential of SAMDAs

4.4.1. KEPCO/KHNP Evaluation

KEPCO/KHNP evaluated the potential SAMDAs not screened out to assess their potential benefits by using bounding techniques to estimate the possible risk reduction. This ~~is~~ was accomplished by associating the basic events identified with a Fussell-Vessely importance of greater than 0.5 percent, and from the top 100 cutsets to a particular SAMDA. This linkage to a SAMDA is provided for each basic event in APR1400-K-X-ER-14001-NP, Sections 7.1 through Section 7.19. The basic event that a potential SAMDA is associated with is also provided in the "Qualitative Screening" column of Table 5 in APR1400-K-X-ER-14001-NP.

Because there are likely several basic events that are considered under a specific SAMDA, KEPCO/KHNP applied a factor of risk reduction based on the sum of Fussell-Vessely importance values for each basic event. KEPCO/KHNP determined the sum of Fussell-Vessely accident, the possibility that such an accident could occur at any time over the licensed life, and the effect of discounting these potential future losses to present value.

The NRC issued Revision 4 of NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," in August 2004 to reflect the agency's policy on discount rates. NUREG/BR-0058, Revision 4, states that two sets of estimates should be developed — one at 3 percent and one at 7 percent. The applicant provided estimates using both discount rates.

KEPCO/KHNP calculated the maximum benefit for at-power internal events, internal flooding events, and internal fire events; along with low-power and shutdown internal events, internal flooding events, and internal fire events for the baseline 7 percent and the sensitivity 3 percent discount rates. The results of the KEPCO/KHNP evaluation are provided in Table 4.6.2-1.

As previously discussed, 38 SAMDAs were carried to the next screening phase. In addition to these remaining SAMDAs, each basic event with a Fussell-Vessely importance of greater than 0.5 percent or part of the top 100 cutsets, if not already included as a basic event, ~~were~~ was reviewed to identify any potential SAMDAs. KEPCO/KHNP then related each of the 38 SAMDAs back to one or more of the basic events and assessed the NPV for each basic event with the following steps:

1. Assessed the maximum benefit for each basic event applying conservative assumptions for risk reductions to the AOE and AOSC categories;
2. Conservatively assessed the COE based on half of the SAMDA values obtained from source documents; and
3. Determined the NPV.

For each of the basic events/SAMDAs applying the 7 percent and 3 percent discount rates, KEPCO/KHNP evaluated the NPV and reached a conclusion of whether the enhancements were cost beneficial. KEPCO/KHNP determined, through its SAMDAS analyses, that there were no potentially cost-beneficial enhancements for the 7 percent discount rate analysis. KEPCO/KHNP stated that its sensitivity analysis for the 3 percent discount rate showed a higher maximum benefit over the 7 percent discount rate. However, KEPCO/KHNP concluded that no design changes would provide a positive cost-benefit for either discount rate, if included in the APR1400 standard plant design.

4.6.2. NRC Evaluation

As shown in Table 4.6.2-1, the NRC's confirmatory analysis for the 7 percent and 3 percent discount rates were in general agreement with the applicant for the offsite public exposure (i.e., APE), offsite property damage cost (i.e., AOC), and onsite occupational dose (i.e., AOE) averted costs. The NRC evaluation resulted in higher values than the applicant's evaluation for the onsite cleanup and decontamination (i.e., $AOSC_{CD}$) averted costs, with a similar higher result for the replacement power (i.e., $AOSC_{RP}$) averted costs.

In the $AOSC_{CD}$ evaluation, the NRC adjusted the base averted cost per event provided by NUREG/BR-0184, which was applied by KEPCO/KHNP, to current dollars, resulting in a higher value for the NRC's evaluation. The small difference between the NRC's and the applicant's $AOSC_{RP}$ averted costs for the 7 percent discount rate evaluation is principally due to applying different inflation factors to adjust the base replacement cost to current dollars. For the 3 percent discount rate analysis of the replacement power, KEPCO/KHNP applied a linear interpolation to the NPV for discount rates below 5 percent, as described near the end of Section 5.7.6.2 of NUREG/BR-0184 (see page 5.45 of NUREG/BR-0184). Based on NRC experience in prior regulatory rulemaking analyses, the NRC applied the same replacement cost formula for both the 7 percent and 3 percent discount rates (see the formula in Section 5.7.6.2 of NUREG/BR-0184 on page 5.44). This is viewed by the NRC as being conservative as demonstrated by the larger replacement power averted cost in the NRC evaluation in comparison to the applicant's evaluation.

In its review, the NRC noted that the applicant used two assumed conservatisms in its cost-benefit analysis. The first case of conservatism involved the total averted costs in each analysis, where the applicant did not apply the percent risk reductions for the contribution to total core damage frequency to the population dose (i.e., APE) and offsite property damage (i.e., AOC) costs. The APE and AOC were based on MELCOR Accident Consequence Code System calculations and, thus, are directly tied to the size of a release. As shown by the NRC's 3 percent discount rate analysis compared to the KEPCO/KHNP 3 percent discount rate analysis, applying this reduction to only the onsite exposure (i.e., AOE) and onsite economic costs (i.e., AOSC), results in a conservative result. Namely, it will result in a total maximum benefit that is larger than if the percentage risk reduction is applied to all cost categories. The second conservative assumption involved the use of the determined COE values, as discussed in Section 4.5.1. As assessed by the NRC staff, when the applicant applies only half of the estimated COE value, the final determination of the cost-benefit analysis could more likely provide a positive NPV.

Even with the above discussed differences in the averted cost values, the NRC's confirmatory analysis also reached the same conclusion as KEPCO/KHNP that there were no cost beneficial design alternatives when applying a 7 percent discount rate. This result ~~is~~ was the same regardless of whether the applicant's conservative assumptions were~~, or were not,~~ applied in the 7 percent discount rate analysis. Based on the NRC's review of the methodology and associated analysis, KEPCO/KHNP's assessment adequately addressed the cost-benefit analysis for the 7 percent discount rate.

For the 3 percent discount rate analysis, the NRC performed a confirmatory calculation to assess the ~~cost~~ costs and benefits applying the NRC results provided in Table 4.6.2-1,