RULEMAKING ISSUE
(Affirmation)

November 16, 1999

SECY-99-268

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: FINAL RULE — AP600 DESIGN CERTIFICATION

PURPOSE:

To obtain the Commission's approval to publish in the Federal Register the attached final rule that would amend 10 CFR Part 52 to certify the AP600 standard plant design.

BACKGROUND:

Westinghouse Electric Company submitted an application for certification of its AP600 standard plant design on June 26, 1992. The NRC staff completed its review of the AP600 design and issued NUREG-1512, "Final Safety Evaluation Report related to Certification of the AP600 Standard Design," in September 1998 (see COMSECY-98-025). The NRC staff issued a final design approval to Westinghouse on September 3, 1998, that signified completion of the technical review phase and readiness for the rulemaking phase of Westinghouse's application.

DISCUSSION:

On May 20, 1999 (64 FR 27626), the NRC published a proposed design certification rule (DCR) for the AP600 standard plant design. The Federal Register notice gave the public an opportunity to comment on the proposed DCR; the AP600 Design Control Document (DCD) (3/99 revision); and the environmental assessment. The Federal Register notice also gave the public an opportunity to request an informal hearing under 10 CFR 52.51(b) and Section II of the notice. In addition, the Federal Register notice was sent to all State liaison officers and public utility commissions. The NRC did not receive any requests for an informal hearing. Only one comment was submitted in response to the Federal Register notice and it did not address the proposed DCR, the AP600 DCD, or the environmental assessment. Therefore, the NRC staff did not change the proposed rule for final rule consideration.

CONTACT:
Jerry N. Wilson, NRR/DRIP
415-3145

The final rule is nearly identical to the two previously issued DCRs for the U.S. advanced boiling-water reactor (ABWR) and System 80+ designs (Appendices A and B to 10 CFR Part 52, respectively). The staff believes that the AP600 DCR should emulate the existing DCRs for the ABWR and the System 80+, inasmuch as the three designs were reviewed contemporaneously against the same technical requirements. Furthermore, many of the procedural issues and their resolutions for the ABWR and the System 80+ DCRs (e.g., the two-tier structure, Tier 2*, and the scope of issue resolution) were developed after extensive discussions with nuclear industry representatives, and Westinghouse participated in those discussions. It was the NRC's intent and Westinghouse's expectation that the resolutions for these issues in the ABWR and System 80+ rulemakings would also be applied to the AP600 design. Accordingly, the staff has modeled the AP600 DCR on the existing DCRs for the ABWR and System 80+, with certain departures. The departures from these DCRs were necessary to account for different applicants; design documentation, including Tier 2* information; design features; and the environmental assessment. The only significant change was the inclusion of the investment protection short-term availability controls in Sections II, III, and VI of the AP600 DCR. Westinghouse was notified by letters dated June 9 and August 2, 1997, that these availability controls would be binding on applicants and licensees that reference the AP600 DCR. The NRC staff's evaluation of Westinghouse's changes to the DCD will be provided in Supplement No. 1 to the FSER.

By letter dated September 29, 1999, Westinghouse submitted documentation changes to the AP600 DCD as a result of a final review that was performed to check the consistency of the implementation of approved design change proposals. Also, DCD changes resulted from ongoing detailed design work on systems, structures, and components outside the scope of design certification and commitments made in the final stages of the design certification review. In addition, changes to the description of the plates that protect the recirculation pump screens was necessitated by the location of other nearby hardware. The NRC staff determined that the documentation changes are acceptable for the AP600 standard plant design and do not affect the staff's findings in NUREG-1512. Therefore, the final design certification rule incorporates the AP600 DCD (9/99 revision) by reference.

COORDINATION:

The Office of the General Counsel has no legal objection to this paper. The Office of the Chief Financial Officer has reviewed this Commission Paper for resource implications and has no objection. The Office of the Chief Information Officer has reviewed this final rule for information technology and information management implications and concurs in it. A copy of the attached Federal Register notice was sent to the ACRS for its consideration.

The NRC staff is preparing a letter to the Director, Office of the Federal Register (OFR), requesting approval of the AP600 DCD for incorporation by reference. The letter will be sent to OFR before we request publication of the Federal Register notice and will address the criteria for approval of documents for incorporation by reference.

RECOMMENDATION:

That the Commission

1. Approve Attachment 1 for publication in the Federal Register.

2. Approve the final environmental assessment in Attachment 2.

3. Certify (in order to satisfy requirements of the Regulatory Flexibility Act, 5 U.S.C. 605(b)) that this rule, if promulgated, will not have a negative economic impact on a substantial number of small entities.

4. Note:

   a. This rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). This rule has been reviewed by the Office of Management and Budget, and the paperwork requirements were approved on August 10, 1999;

   b. The Chief Counsel for Advocacy of the Small Business Administration will be informed of the certification regarding the economic impact on small entities and the reasons for it as required by the Regulatory Flexibility Act (Section VII of the Federal Register notice);

   c. The appropriate congressional committees will be informed; and

   d. The staff does not believe that this rule falls within the Office of Management and Budget's definition of a "major" rule and, therefore, this rule may become effective without a 60-day Congressional review period.

   e. The Office of Public Affairs has prepared a press release.

William Travers

William D. Travers
Executive Director
for Operations

Attachments:
1. Federal Register notice
2. Environmental Assessment
3. Congressional Review Act Form

Commissioners' completed vote sheets/comments should be provided directly to the Office of the Secretary by COB Friday, December 3, 1999.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT November 26, 1999, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional review and comment, the Commissioenrs and the Secretariat should be apprised of when comments may be expected.

This paper is tentatively scheduled for affirmation at an Open Meeting during the Week of December 6, 1999. Pleae refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

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[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR PART 52

RIN 3150 - AG23

AP600 Design Certification

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC or Commission) is amending its regulations to certify the AP600 standard plant design under Subpart B of 10 CFR Part 52. This action is necessary so that applicants or licensees intending to construct and operate an AP600 design may do so by referencing this regulation [AP600 design certification rule (DCR)]. The applicant for certification of the AP600 design was Westinghouse Electric Company LLC (hereinafter referred to as Westinghouse).

EFFECTIVE DATE: The effective date of this rule is [**Insert date 30 days after publication in the Federal Register**]. The incorporation by reference of certain documents listed in this regulation is approved by the Director of the Office of the Federal Register as of [**Insert date 30 days after publication in the Federal Register**].

FOR FURTHER INFORMATION CONTACT: Jerry N. Wilson, Mail Stop O-12 G15, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or telephone (301) 415-3145, or e-mail: jnw@nrc.gov.

SUPPLEMENTARY INFORMATION:

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VII. Regulatory Flexibility Act certification.
VIII. Backfit analysis.
IX. Small Business Regulatory Enforcement Fairness Act.
X. National Technology Transfer and Advancement Act.

I. Background

The NRC added 10 CFR Part 52 to its regulations to provide for the issuance of early site permits, standard design certifications, and combined licenses for nuclear power reactors. Subpart B of 10 CFR Part 52 established the process for obtaining design certifications. On June 26, 1992, Westinghouse tendered its application for certification of the AP600 design with the NRC. Westinghouse submitted this application in accordance with Subpart B and Appendix O of 10 CFR Part 52. The NRC formally accepted the application as a docketed application for design certification (Docket No. 52-003) on December 31, 1992 (58 FR 3982, January 12, 1993). Information submitted before that date can be found under Project No. 676.

The NRC staff issued a final safety evaluation report (FSER) related to certification of the AP600 standard plant design in September 1998 (NUREG-1512, 63 FR 48772). The FSER documents the results of the staff's safety review of the AP600 design against the requirements of 10 CFR Part 52, Subpart B, and delineates the scope of the technical details considered in evaluating the design. Subsequently, Westinghouse submitted some changes to the AP600 design documentation [AP600 Design Control Document (DCD) 9/99 revision]. The NRC staff reviewed these changes and determined that they did not affect the findings in the FSER. The NRC staff's evaluation of the changes to the DCD will be provided in Supplement No. 1 to the FSER. A notice of availability for Supplement No. 1 will be published in the Federal Register. The FSER and Supplement No. 1 provide the bases for Commission approval of the AP600 standard plant design through design certification. A copy of the FSER may be obtained from the Superintendent of Documents, U. S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 or the National Technical Information Service, Springfield, VA 22161-0002. The final design approval (FDA) for the AP600 design was issued on September 3, 1998, and published in the Federal Register on September 11, 1998 (63 FR 48772).

II Public Comment

Subpart B of 10 CFR Part 52 provides for Commission approval of standard designs for nuclear power facilities (e.g., design certification) through rulemaking. In accordance with the Administrative Procedure Act (APA), Part 52 provides the opportunity for the public to submit written comments on the proposed design certification rule. However, Part 52 goes beyond the requirements of the APA by providing the public with an opportunity to request a hearing before the Atomic Safety and Licensing Board Panel in a design certification rulemaking. Therefore, on May 20, 1999, the NRC published a proposed rule in the Federal Register (64 FR 27626) that invited public comment and provided the public with the opportunity to request an informal hearing before an Atomic Safety and Licensing Board.

The period for requesting an informal hearing or submitting comments on the proposed DCR, AP600 DCD, or draft environmental assessment expired on August 3, 1999. The NRC did not receive any requests for an informal hearing during this period, but it did receive a comment from a member of the public. This individual did not comment on the AP600 DCD, draft environmental assessment, or proposed DCR. Rather, the commenter expressed views on new nuclear power plants and nuclear waste. Therefore, the Commission did not change the proposed DCR, AP600 DCD, or draft environmental assessment [except for editorial revisions] and has adopted this rule [Appendix C to 10 CFR Part 52] as final.

III. Section-By-Section Discussion of Design Certification Rule

The final rule for the AP600 standard plant design is nearly identical to the two design certification rules (DCRs) for the U.S. ABWR and the System 80+ designs, which the NRC previously adopted. These DCRs are set forth in 10 CFR Part 52, Appendix A (U.S. ABWR, 62 FR 25800, May 12, 1997) and Appendix B (System 80+, 62 FR 27840, May 21, 1997). The AP600 DCR emulates the U.S. ABWR and System 80+ DCRs, inasmuch as the three designs were reviewed contemporaneously against the same technical requirements. Furthermore, many of the procedural issues and their resolutions for the ABWR and the System 80+ DCRs (e.g., the two-tier structure, Tier 2*, the scope of issue resolution) were developed after extensive discussions with nuclear industry representatives, and Westinghouse participated in those discussions. It was the NRC's intent and Westinghouse's expectation that the resolutions for these issues in the ABWR and System 80+ rulemakings would also be applied to the AP600 design certification. Accordingly, the NRC has modeled the AP600 DCR on the existing DCRs for the ABWR and System 80+ designs, with certain departures. These departures were necessary to acknowledge that Westinghouse is the applicant for the AP600 DCR, and to account for differences in the AP600 design documentation (including Tier 2* information), design features, and environmental assessment (including severe accident mitigation design alternatives). The only significant change was the inclusion of the investment protection short-term availability controls in Sections II, III, and VI of the AP600 DCR.

The following discussion sets forth the purpose and key aspects of each portion of the final AP600 design certification rule. All section, paragraph, and subparagraph references are to the provisions in Appendix C to 10 CFR Part 52.

A. Introduction.

The purpose of Section I of Appendix C to 10 CFR Part 52 ("this appendix") is to identify the standard plant design that is approved by this design certification rule and the applicant for certification of the standard design. Identification of the design certification applicant is necessary to implement this appendix, for two reasons. First, the implementation of 10 CFR 52.63(c) depends on whether an applicant for a combined license (COL) contracts with the design certification applicant to provide the generic DCD and supporting design information. If the COL applicant does not use the design certification applicant to provide this information, then the COL applicant must meet the requirements in 10 CFR 52.63(c). Also, subparagraph X.A.1 of this appendix imposes a requirement on the design certification applicant to maintain the generic DCD throughout the time period in which this appendix may be referenced.

B. Definitions.

The terms Tier 1, Tier 2, Tier 2*, and COL action items (license information) are defined in this appendix because these concepts were not envisioned when 10 CFR Part 52 was developed. The design certification applicants and the NRC staff used these terms in implementing the two-tiered rule structure that was proposed by representatives of the nuclear industry after issuance of 10 CFR Part 52. During consideration of the comments received on Appendices A and B to Part 52, the Commission determined that it would be useful to distinguish between the "plant-specific DCD" and the "generic DCD," the latter of which is incorporated by reference into this appendix and remains unaffected by plant-specific departures. This distinction is necessary in order to clarify the obligations of applicants and licensees that reference this appendix. Also, the technical specifications that are located in Section 16.1 of the generic DCD are designated as "generic technical specifications" in order to facilitate the special treatment of this information under this appendix. Therefore, appropriate definitions for these additional terms are included in this appendix.

The Tier 1 portion of the design-related information contained in the DCD is *certified* by this appendix and, therefore, subject to the special backfit provisions in paragraph VIII.A of this appendix. An applicant who references this appendix is required to incorporate by reference and comply with Tier 1, under paragraph III.B and subparagraph IV.A.1 of this appendix. This information consists of an introduction to Tier 1, the system based and non-system based design descriptions and corresponding inspections, tests, analyses, and acceptance criteria (ITAAC), significant interface requirements, and significant site parameters for the design. The design descriptions, interface requirements, and site parameters in Tier 1 were derived entirely from Tier 2, but may be more general than the Tier 2 information. The NRC staff's evaluation of the Tier 1 information is provided in Section 14.3 of the FSER. Changes to or departures from the Tier 1 information must comply with paragraph VIII.A of this appendix.

The Tier 1 design descriptions serve as design commitments for the lifetime of a facility referencing the design certification. The ITAAC verify that the as-built facility conforms with the approved design and applicable regulations. In accordance with 10 CFR 52.103(g), the Commission must find that the acceptance criteria in the ITAAC are met before operation. After the Commission has made the finding required by 10 CFR 52.103(g), the ITAAC do not constitute regulatory requirements for licensees or for renewal of the COL. However, subsequent modifications to the facility must comply with the design descriptions in the plant-specific DCD unless changes are made in accordance with the change process in Section VIII of this appendix. The Tier 1 interface requirements are the most significant of the interface requirements for systems that are wholly or partially outside the scope of the standard design, which were submitted in response to 10 CFR 52.47(a)(1)(vii) and must be met by the site-specific design features of a facility that references this appendix. The Tier 1 site parameters are the most significant site parameters, which were submitted in response to 10 CFR 52.47(a)(1)(iii). An application that references this appendix must demonstrate that the site parameters (both Tier 1 and Tier 2) are met at the proposed site (refer to III.D of this SOC).

Tier 2 is the portion of the design-related information contained in the DCD that is *approved* by this appendix but is not certified. Tier 2 information is subject to the backfit provisions in paragraph VIII.B of this appendix. Tier 2 includes the information required by 10 CFR 52.47 (with the exception of generic technical specifications, conceptual design information, and the evaluation of severe accident mitigation design alternatives) and the supporting information on inspections, tests, and analyses that will be performed to demonstrate that the acceptance criteria in the ITAAC have been met. As with Tier 1, paragraph III.B and subparagraph IV.A.1 of this appendix require an applicant who references this appendix to incorporate Tier 2 by reference and to comply with Tier 2, except for the COL action items, including the investment protection short-term availability controls in Section 16.3 of the generic DCD. The definition of Tier 2 makes clear that Tier 2 information has been determined by the Commission, by virtue of its inclusion in this appendix and its designation as Tier 2 information, to be an approved ("sufficient") method for meeting Tier 1 requirements. However, there may be other acceptable ways of complying with Tier 1. The appropriate criteria for departing from Tier 2 information are specified in paragraph VIII.B of this appendix. Departures from Tier 2 do not negate the requirement in paragraph III.B to reference Tier 2.

A definition of "combined license (COL) action items" (combined license information), which is part of the Tier 2 information, has been added to clarify that COL applicants, who reference this appendix, are required to address these matters in their license application, but the COL action items are not the only acceptable set of information. An applicant may depart from or omit these items, provided that the departure or omission is identified and justified in the

FSAR. After issuance of a construction permit or combined license, these items are not requirements for the licensee unless such items are restated in its FSAR.

The investment protection short-term availability controls, which are set forth in Section 16.3 of the generic DCD, were added to the list of information that is part of Tier 2. This set of requirements was added to Tier 2 to make it clear that the availability controls are not operational requirements for the purposes of paragraph VIII.C of this appendix. Rather, the availability controls are associated with specific design features, and the availability controls may be changed in the same manner as other Tier 2 information.

Certain Tier 2 information has been designated in the generic DCD with brackets and italicized text as "Tier 2*" information and, as discussed in greater detail in the section-by-section explanation for paragraph VIII.B, a plant-specific departure from Tier 2* information requires prior NRC approval. However, the Tier 2* designation expires for some of this information when the facility first achieves full power after the finding required by 10 CFR 52.103(g). The process for changing Tier 2* information and the time at which its status as Tier 2* expires is set forth in subparagraph VIII.B.6 of this appendix. Some Tier 2* requirements, concerning special preoperational tests, are designated to be performed only for the first plant or first three plants referencing the AP600 DCR. The Tier 2* designation for these selected tests will expire after the first plant or first three plants complete the specified tests. However, a COL action item requires that subsequent plants shall also perform the tests or justify that the results of the first-plant-only or first-three-plants-only tests are applicable to the subsequent plant. The Commission is interested in comments addressing whether the first-plant-only or first-three-plants-only limitations should be part of the Tier 2* information for these specified tests.

During development of Appendices A and B to Part 52, the Commission decided that there would be both generic (master) DCDs maintained by the NRC and the design certification applicant, as well as individual plant-specific DCDs, maintained by each applicant and licensee who references this appendix. The generic DCDs (identical to each other) would reflect generic changes to the version of the DCD approved in this design certification rulemaking. The generic changes would occur as the result of generic rulemaking by the Commission (subject to the change criteria in Section VIII of this appendix). In addition, the Commission understood that each applicant and licensee referencing this appendix would be required to submit and maintain a plant-specific DCD. This plant-specific DCD would contain (not just incorporate by reference) the information in the generic DCD. The plant-specific DCD would be updated as necessary to reflect the generic changes to the DCD that the Commission may adopt through rulemaking, any plant-specific departures from the generic DCD that the Commission imposed on the licensee by order, and any plant-specific departures that the licensee chose to make in accordance with the relevant processes in Section VIII of this appendix. Thus, the plant-specific DCD would function akin to an updated Final Safety Analysis Report, in the sense that it would provide the most complete and accurate information on a plant's licensing basis for that part of the plant within the scope of this appendix. Therefore, this appendix defines both a generic DCD and plant-specific DCD. Also, the Commission decided to treat the technical specifications in Section 16.1 of the generic DCD as a special category of information and to designate them as generic technical specifications. A COL applicant must submit plant-specific technical specifications that consist of the generic technical specifications, which may be modified under paragraph VIII.C of this appendix, and the remaining plant-specific information needed to complete the technical specifications, including bracketed values. The Final Safety Analysis Report (FSAR) that is required by § 52.79(b) will consist of the plant-specific DCD, the site-specific portion of the FSAR, and the plant-specific technical specifications.

C. Scope and contents.

The purpose of Section III of this appendix is to describe and define the scope and contents of this design certification and to set forth how documentation discrepancies or inconsistencies are to be resolved. Paragraph A of this section is the required statement of the Office of the Federal Register (OFR) for approval of the incorporation by reference of Tier 1, Tier 2, and the generic technical specifications into this appendix and paragraph B requires COL applicants and licensees to comply with the requirements of this appendix. The legal effect of incorporation by reference is that the material is treated as if it were published in the Federal Register. This material, like any other properly-issued regulation, has the force and effect of law. Tier 1 and Tier 2 information, as well as the generic 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. The generic DCD was prepared to meet the requirements of the OFR for incorporation by reference (1 CFR Part 51). One of the requirements of OFR for incorporation by reference is that the design certification applicant must make the generic DCD available upon request after the final rule becomes effective. Therefore, paragraph III.A of this appendix identifies a representative of Westinghouse who can be contacted to obtain a copy of the generic DCD.

Paragraphs A and B of Section III also identify the investment protection short-term availability controls in Section 16.3 of the generic DCD as part of the Tier 2 information. During its review of the AP600 design, the NRC determined that residual uncertainties associated with passive safety system performance increased the importance of non-safety-related active systems in providing defense-in-depth functions that back-up the passive systems. As a result, Westinghouse developed some administrative controls to provide a high level of confidence that active systems having a significant safety role are available when challenged. Westinghouse named these additional controls "investment protection short-term availability controls," and the Commission included this statement in Section III to ensure that these availability controls are binding on applicants and licensees that reference this appendix and will be enforceable by the NRC. The NRC's evaluation of the availability controls is provided in Chapter 22 of the FSER.

The generic DCD (master copy) for this design certification will be archived at NRC's central file with a matching copy at OFR. Copies of the up-to-date generic DCD will also be available at the NRC's Public Document Room. Questions concerning the accuracy of information in an application that references this appendix will be resolved by checking the master copy of the generic DCD in NRC's central file. If a generic change (rulemaking) is made to the DCD pursuant to the change process in Section VIII of this appendix, then at the completion of the rulemaking the NRC will request approval of the Director, OFR for the changed incorporation by reference and change its copies of the generic DCD and notify the OFR and the design certification applicant to change their copies. The Commission is requiring that the design certification applicant maintain an up-to-date copy under subparagraph X.A.1 of this appendix because it is likely that most applicants intending to reference the standard design will obtain the generic DCD from the design certification applicant. Plant-specific changes to and departures from the generic DCD will be maintained by the applicant or licensee that references this appendix in a plant-specific DCD, under subparagraph X.A.2.

In addition to requiring compliance with this appendix, paragraph B clarifies that the conceptual design information and Westinghouse's evaluation of severe accident mitigation design alternatives are not considered to be part of this appendix. The conceptual design information is for those portions of the plant that are outside the scope of the standard design and are intermingled throughout Tier 2. As provided by 10 CFR 52.47(a)(1)(ix), these conceptual designs are not part of this appendix and, therefore, are not applicable to an application that references this appendix. Therefore, the applicant does not need to conform with the conceptual design information that was provided by the design certification applicant. The conceptual design information, which consists of site-specific design features, was required to facilitate the design certification review. Conceptual design information is neither Tier 1 nor Tier 2. Section 1.8 of Tier 2 identifies the location of the conceptual design information. Westinghouse's evaluation of various design alternatives to prevent and mitigate severe accidents does not constitute design requirements. The Commission's assessment of this information is discussed in Section IV of this SOC on environmental impacts. The detailed methodology and quantitative portions of the design-specific probabilistic risk assessment (PRA), as required by 10 CFR 52.47(a)(1)(v), were not included in the generic DCD, as requested by NEI and the applicant for design certification. The NRC agreed with the request to delete this information because conformance with the deleted portions of the PRA is not necessary. Also, the NRC's position is predicated in part upon NEI's acceptance, in conceptual form, of a future generic rulemaking that will require a COL applicant or licensee to have a plant-specific PRA that updates and supersedes the design-specific PRA supporting this rulemaking and maintain it throughout the operational life of the facility.

Paragraphs C and D of Section III set forth the manner in which potential conflicts are to be resolved. Paragraph C establishes the Tier 1 description in the DCD as controlling in the event of an inconsistency between the Tier 1 and Tier 2 information in the DCD. Paragraph D establishes the generic DCD as the controlling document in the event of an inconsistency between the DCD and either the application for certification of the AP600 design (AP600 Standard Safety Analysis Report) or the final safety evaluation report for the certified standard design.

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

D. Additional requirements and restrictions.

Section IV of this appendix sets forth additional requirements and restrictions imposed upon an applicant who references this appendix. Paragraph IV.A sets forth the information requirements for these applicants. This appendix distinguishes between information and/or documents which must actually be *included* in the application or the DCD, versus those which may be *incorporated by reference* (i.e., referenced in the application as if the information or documents were actually included in the application), thereby reducing the physical bulk of the application. 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 by reference.

Subparagraph A.1 requires an applicant who references this appendix to incorporate by reference this appendix in its application. The legal effect of such incorporation by reference is that this appendix is legally binding on the applicant or licensee. Subparagraph A.2.a is intended to make clear that the initial application must include a plant-specific DCD. This assures, among other things, that the applicant commits to complying with the DCD. This paragraph also requires the plant-specific DCD to use the same format as the generic DCD and to reflect the applicant's proposed departures and exemptions from the generic DCD as of the time of submission of the application. The Commission expects that the plant-specific DCD will become the plant's final safety analysis report (FSAR), by including within its pages, at the appropriate points, information such as site-specific information for the portions of the plant outside the scope of the referenced design, including related ITAAC, and other matters required to be included in an FSAR by 10 CFR 50.34 and 52.79. Integration of the plant-specific DCD and remaining site-specific information into the plant's FSAR, will result in an application that is easier to use and should minimize "duplicate documentation" and the attendant possibility for confusion. Subparagraph A.2.a is also intended to make clear that the initial application must include the reports on departures and exemptions as of the time of submission of the application.

Subparagraph A.2.b requires that the application include the reports required by paragraph X.B of this appendix for exemptions and departures proposed by the applicant as of the date of submission of its application. Subparagraph A.2.c requires submission of plant-specific technical specifications for the plant that consists of the generic technical specifications from Section 16.1 of the DCD, with any changes made under paragraph VIII.C of this appendix, and the technical specifications for the site-specific portions of the plant that are either partially or wholly outside the scope of this design certification. The applicant must also provide the plant-specific information designated in the generic technical specifications, such as bracketed values.

Subparagraph A.2.d makes it clear that the applicant must provide information demonstrating that the proposed site falls within the site parameters for this appendix and that the plant-specific design complies with the interface requirements, as required by 10 CFR 52.79(b). If the proposed site has a characteristic that exceeds one or more of the site parameters in the DCD, then the proposed site is unacceptable for this design unless the applicant seeks an exemption under Section VIII of this appendix and justifies why the certified design should be found acceptable on the proposed site. Subparagraph A.2.e requires submission of information addressing COL Action Items, which are identified in the generic DCD as Combined License Information, in the application. The Combined License Information identifies matters that need to be addressed by an applicant that references this appendix, as required by Subpart C of 10 CFR Part 52. An applicant may depart from or omit these items, provided that the departure or omission is identified and justified in its application (FSAR). Subparagraph A.2.f requires that the application include the information required by 10 CFR 52.47(a) that is not within the scope of this rule, such as generic issues that must be addressed, in whole or in part, by an applicant that references this rule. Subparagraph IV.A.3 requires the applicant to physically include, not simply reference, the proprietary and safeguards information referenced in the DCD, or its equivalent, to assure that the applicant has actual notice of these requirements.

Paragraph IV.B reserves to the Commission the right to determine in what manner this design certification may be referenced by an applicant for a construction permit or operating license under 10 CFR Part 50. This determination may occur in the context of a subsequent rulemaking modifying 10 CFR Part 52 or this design certification 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 design certifications were not implemented in the manner that was originally envisioned at the time that 10 CFR Part 52 was created. The Commission's concern is with the manner in which ITAAC were developed and the lack of experience with design certifications in license proceedings. Therefore, it is appropriate to have some uncertainty regarding the manner in which this appendix could be referenced in a 10 CFR Part 50 licensing proceeding.

E. Applicable regulations.

The purpose of Section V of this appendix is to specify the regulations that were applicable and in effect at the time that 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 are not applicable to this certified design (exempt).

Paragraph V.A identifies the regulations in 10 CFR Parts 20, 50, 73, and 100 that are applicable to the AP600 design. After the NRC staff issued its FSER for the AP600 design in September 1998, the Commission amended several existing regulations and adopted several new regulations. The Commission has reviewed these regulations to determine if they are applicable to this design and, if so, to determine if the design meets these regulations. The Commission finds that the AP600 design either meets the requirements of these regulations or that these regulations are not applicable to the design, as discussed below. The Commission's determination of the applicable regulations was made as of the date specified in paragraph V.A of this appendix. The specified date is the date that this appendix was approved by the Commission and signed by the Secretary of the Commission.

10 CFR 20, Transfer for Disposal and Manifests; Minor Technical Conforming Amendment (63 FR 50127; September 21, 1998).

This amendment to Part 20 removed expired provisions from the regulations on low-level waste shipment manifest information. The previous regulation included dual implementation procedures that allow use of one of two manifesting procedures. This is a procedural requirement that applies to licensees and, therefore, is not applicable to either NRC issuance of design certification or applicants for design certification.

10 CFR 30 & 50, Financial Assurance Requirements for Decommissioning Nuclear Power Reactors (63 FR 50465; September 22, 1998).

This amendment to the regulations requires power reactor licensees to report periodically on the status of their decommissioning funds, and on changes in their external trust agreements and other financial assurance mechanisms. This regulation applies to licensees and, therefore, is not applicable to either NRC issuance of design certification or applicants for design certification.

10 CFR 50 & 70, Criticality Accident Requirements (63 FR 63127; November 12, 1998).

This amendment to the regulations provides licensees of light-water nuclear reactors with greater flexibility in meeting the requirement to maintain a criticality monitoring system in each area in which special nuclear material is handled, used, or stored. The criticality monitoring system is not considered to be part of the plant design and, therefore, is not applicable to either NRC issuance of design certification or applicants for design certification.

10 CFR 50, Changes to Quality Assurance Programs (64 FR 9030; February 23, 1999).

This amendment to 10 CFR 50.54(a) allows licensees to make routine or administrative quality assurance (QA) program changes, which do not have an adverse impact on the effectiveness of their QA program, without obtaining NRC approval in advance. This is a procedural requirement that can be utilized after issuance of a license and, therefore, is not applicable to either NRC issuance of design certification or applicants for design certification.

10 CFR 50 & 73, Frequency of Reviews and Audits for Emergency Preparedness Programs, Safeguards Contingency Plans, and Security Programs for Nuclear Power

Reactors (64 FR 14814; March 29, 1999).

This amendment to the regulations allows licensees to change the frequency of independent reviews and audits of their emergency preparedness programs, safeguards contingency plans, and security programs. This is a procedural requirement that can be utilized after issuance of a license and, therefore, is not applicable to either NRC issuance of design certification or applicants for design certification.

10 CFR 50, Codes and Standards: IEEE National Consensus Standard (64 FR 17944; April 13, 1999).

This amendment to 10 CFR 50.55a incorporates IEEE Std. 603-1991 by reference, a national consensus standard for power, instrumentation, and control portions of safety systems in nuclear power plants. The NRC staff reviewed the AP600 design against this IEEE standard and, therefore, the Commission has determined that the AP600 design meets the applicable portions of this new requirement.

10 CFR 50, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (64 FR 38551; July 19, 1999).

This amendment to 10 CFR 50.65 requires nuclear power plant licensees to assess the effect of equipment maintenance on the plant's capability to perform safety functions before beginning maintenance activities on structures, systems, and components within the scope of the maintenance rule. This requirement applies only to licensees and, therefore, is not applicable to either NRC issuance of design certification or applicants for design certification.

In paragraph V.B of this appendix, the Commission identified the regulations that do not apply to the AP600 design. The Commission has determined that the AP600 design should be exempt from portions of 10 CFR 50.34, 50.62, and Appendix A to Part 50, as described in the FSER (NUREG-1512) and summarized below:

(1) Paragraph (a)(1) of 10 CFR 50.34 - whole body dose criterion.

This regulation sets forth dose criteria to be used in siting determinations. The NRC staff performed its evaluation of the radiological consequences of postulated design basis accidents for the AP600 design against the dose criterion specified in 10 CFR 50.34(a)(1)(ii)(D) because it was the Commission's intent that the new dose criterion be used for future nuclear power plants. However, when the NRC codified the new reactor site criteria for nuclear power plants (61 FR 65157; December 11, 1996), it made an error in the assignment of applicants that could use the new dose criterion [25 rem TEDE], versus those that must use the whole body criterion. The assignment of applicants in 10 CFR 50.34(a)(1), who must use the whole body criterion, should not have included applicants for a design certification or combined license who applied prior to January 10, 1997 (refer to 61 FR 65158). The Commission adopted 25 rem TEDE as the new dose criterion for future plant evaluation purposes, because this value is essentially the same level of risk as the current criterion (61 FR 65160). Therefore, the Commission has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that application of the 25 rem whole body criterion is not necessary to achieve the underlying purpose of the rule because 25 rem TEDE is essentially the same level of risk. On this basis, the Commission concludes that the AP600 design review can be performed pursuant to the new dose criterion [25 rem TEDE] and an exemption from the requirements of 10 CFR 50.34(a)(1) is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security.

(2) Paragraph (f)(2)(iv) of 10 CFR 50.34 - Plant Safety Parameter Display Console.

10 CFR 50.34(f)(2)(iv) requires that an application provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, be capable of displaying a full range of important plant parameters and data trends on demand, and be capable of indicating when process limits are being approached or exceeded. Westinghouse answered this requirement, in Section 18.8.2 of the DCD, with an integrated design rather than a stand-alone, add-on system, as is used at most current operating plants. Specifically, Westinghouse integrated the SPDS requirements into the design requirements for the alarm and display systems. In NUREG-0800, the NRC staff indicated that, for applicants who are in the early stages of the control room design, the "function of a separate SPDS may be integrated into the overall control room design" (p. 18.0-1). Therefore, the Commission has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that the requirement for an SPDS console need not be applied in this particular circumstance to achieve the underlying purpose because Westinghouse has provided an acceptable alternative that accomplishes the intent of the regulation. On this basis, the Commission concludes that an exemption from the requirements of 10 CFR 50.34(f)(2)(iv) is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security.

(3) Paragraphs (f)(2)(vii), (viii), (xxvi), and (xxviii) of 10 CFR 50.34 - Accident Source Terms in TID 14844.

Pursuant to 10 CFR 52.47(a)(ii), an applicant for design certification must demonstrate compliance with any technically relevant TMI requirements in 10 CFR 50.34(f). The TMI requirements in 10 CFR 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii) refer to the accident source term in TID 14844. Specifically, 10 CFR 50.34(f)(2)(xxviii) requires the evaluation of pathways that may lead to control room habitability problems "under accident conditions resulting in a TID 14844 source term release." Similar wording appears in requirements (vii), (viii), and (xxvi). Westinghouse has adopted the new source term technology summarized in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," dated February 1995, not the old TID 14844 source term cited in 10 CFR Part 50.34(f). The new source term is a more realistic representation of the source term resulting from postulated design basis accidents, therefore, the Commission has determined that the special circumstances described in 10 CFR 50.12(a)(ii) exist in that these regulations need not be applied in this particular circumstance to achieve the underlying purpose because Westinghouse has adopted acceptable alternatives that accomplish the underlying intent of the regulations that specify TID 14844. On this basis, the Commission concludes that a partial exemption from the requirements of paragraphs (f)(2)(vii), (viii), (xxvi), and (xxviii) of 10 CFR 50.34 is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security.

(4) Paragraph (c)(1) of 10 CFR 50.62 - Auxiliary feedwater system.

The AP600 design relies on the passive residual heat removal system (PRHR) in lieu of an auxiliary or emergency feedwater system as its safety-related method of removing decay heat. Westinghouse requested an exemption from a portion of 10 CFR 50.62(c)(1), which requires auxiliary or emergency feedwater as an alternate system for decay heat removal during an ATWS event. The NRC staff concluded that Westinghouse met the intent of the rule by relying on the PRHR system to remove the decay heat and, thereby, met the underlying purpose of the rule. Therefore, the Commission has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that the requirement for an auxiliary or emergency feedwater system is not necessary to achieve the underlying purpose of 10 CFR 50.62(c)(1), because Westinghouse has adopted acceptable alternatives that accomplish the intent of this regulation, and the exemption is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security.

(5) Appendix A to 10 CFR Part 50, GDC 17 - Offsite Power Sources.
Westinghouse requested a partial exemption from the requirement in GDC 17 for a second offsite power supply circuit. The AP600 plant design relies on safety-related "passive" systems. Unlike operating plants with active safety-related systems, the AP600 safety-related systems only require a small amount of electric power for valves and related instrumentation. The onsite Class 1E batteries and associated dc and ac distribution systems can provide the power for these valves and instrumentation. In addition, if no offsite power is available, it is expected that the non-safety-related onsite diesel generators would be available for important plant functions; however, this non-safety- related ac power is not relied on to maintain core cooling or containment integrity. Therefore, the Commission has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that the requirement need not be applied in this particular circumstance to achieve the underlying purpose of having two offsite power sources because the AP600 design includes an acceptable alternative approach to accomplish safety functions that does not rely on power from the offsite system and, therefore, accomplishes the intent of the regulation. On this basis, the Commission concludes that a partial exemption from the requirements of GDC 17 is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security.

(6) Appendix A to 10 CFR Part 50, GDC 19 - whole body dose criterion.
The NRC staff used a criterion of 5 rem TEDE for evaluating the radiological consequences of design basis accidents in the control room of the AP600 design. The NRC staff used the 5 rem TEDE criterion to be consistent with the new reactor site criteria in 10 CFR 50.34(a)(1) [61 FR 65157], although GDC 19 specifies . . . "5 rem whole body, or its equivalent to any part of the body". . . The Commission has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that application of the 5 rem whole body criterion is not necessary to achieve the underlying purpose of the rule because a TEDE dose provides essentially the same level of risk as a whole body dose (see 61 FR 65160). On this basis, the Commission concludes that a partial exemption from GDC 19 is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security.

F. Issue resolution.

The purpose of Section VI of this appendix is to identify the scope of issues that are resolved by the Commission in this rulemaking and; therefore, are "matters resolved" within the meaning and intent of 10 CFR 52.63(a)(4). The section is divided into five parts: (A) the Commission's safety findings in adopting this appendix, (B) the scope and nature of issues which are resolved by this rulemaking, (C) issues which are not resolved by this rulemaking, (D) the backfit restrictions applicable to the Commission with respect to this appendix, and (E) the availability of secondary references.

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

Paragraph B sets forth the scope of issues which may not be challenged as a matter of right in subsequent proceedings. The introductory phrase of paragraph B clarifies that issue resolution as described in the remainder of the paragraph extends to the delineated NRC proceedings referencing this appendix. The remainder of paragraph B describes the categories of information for which there is issue resolution. Specifically, subparagraph B.1 provides that all nuclear safety issues arising from the Atomic Energy Act of 1954, as amended, that are associated with the information in the NRC staff's FSER (NUREG-1512) and Supplement No. 1, the Tier 1 and Tier 2 information (including the availability controls in Section 16.3 of the generic DCD), and the rulemaking record for this appendix are resolved within the meaning of § 52.63(a)(4). These issues include the information referenced in the DCD that are requirements (i.e., "secondary references"), as well as all issues arising from proprietary and safeguards information which are intended to be requirements. Subparagraph B.2 provides for issue preclusion of proprietary and safeguards information. Subparagraphs B.3, B.4, B.5, and B.6 clarify that approved changes to and departures from the DCD which are accomplished in compliance with the relevant procedures and criteria in Section VIII of this appendix continue to be matters resolved in connection with this rulemaking. Subparagraph B.7 provides that, for those plants located on sites whose site parameters do not exceed those assumed in Westinghouse's evaluation of severe accident mitigation design alternatives (SAMDAs), all issues with respect to SAMDAs arising under the National Environmental Policy Act of 1969 associated with the information in the Environmental Assessment for this design and the information regarding SAMDAs in Appendix 1B of the generic DCD are also resolved within the meaning and intent of § 52.63(a)(4). In the event an exemption from a site parameter is granted, the exemption applicant has the initial burden of demonstrating that the original SAMDA analysis still applies to the actual site parameters but, if the exemption is approved, requests for litigation at the COL stage must meet the requirements of § 2.714 and present sufficient information to create a genuine controversy in order to obtain a hearing on the site parameter exemption.

Paragraph C reserves the right of the Commission to impose operational requirements on applicants that reference this appendix. This provision reflects the fact that operational requirements, including generic technical specifications in Section 16.1 of the DCD, were not completely or comprehensively reviewed at the design certification stage. Therefore, the special backfit provisions of § 52.63 do not apply to operational requirements. However, all design changes will be controlled by the appropriate provision in Section VIII of this appendix. Although the information in the DCD that is related to operational requirements was necessary to support the NRC staff's safety review of this design, the review of this information was not sufficient to conclude that the operational requirements are fully resolved and ready to be assigned finality under § 52.63. As a result, if the NRC wanted to change a temperature limit and that operational change required a consequential change to a design feature, then the temperature limit backfit would be controlled by Section VIII (paragraph A or B) of this appendix. However, changes to other operational issues, such as in-service testing and in-service inspection programs, post-fuel load verification activities, and shutdown risk that do not require a design change would not be restricted by § 52.63 (see paragraph VIII.C of this appendix). Paragraph VI.C does allow the NRC to impose future operational requirements (distinct from design matters) on applicants who reference this design certification. Also, license conditions for portions of the plant within the scope of this design certification, e.g. start-up and power ascension testing, are not restricted by § 52.63. The requirement to perform these testing programs is contained in 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. Therefore, another regulatory vehicle is necessary to ensure that licensees comply with the matters contained in the license conditions. License conditions for these areas cannot be developed now because this requires the type of detailed design information that will be developed after design certification. In the absence of detailed design information to evaluate the need for and develop specific post-fuel load verifications for these matters, the Commission is reserving the right to impose license conditions by rule for post-fuel load verification activities for portions of the plant within the scope of this design certification.

Paragraph D reiterates the restrictions (contained in Section VIII of this appendix) placed upon the Commission when ordering generic or plant-specific modifications, changes or additions to structures, systems or components, design features, design criteria, and ITAAC (subparagraph VI.D.3 addresses ITAAC) within the scope of the certified design.

Paragraph E provides the procedure for an interested member of the public to obtain access to proprietary or safeguards information for the AP600 design, in order to request and participate in proceedings identified in paragraph VI.B of this appendix, viz., proceedings involving licenses and applications which reference this appendix. As set forth in paragraph VI.E, access must first be sought from the design certification applicant. If Westinghouse refuses to provide the information, the person seeking access shall request access from the Commission or the presiding officer, as applicable. Access to the proprietary or safeguards information may be ordered by the Commission, but must be subject to an appropriate non-disclosure agreement.

G. Duration of this appendix.

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

H. Processes for changes and departures.

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

Both Section VIII of this appendix and this SOC refer to an "exemption" from one or more requirements of this appendix and the criteria for granting an exemption. The Commission cautions that where the exemption involves an underlying substantive requirement (applicable regulation), then the applicant or licensee requesting the exemption must also show that an exemption from the underlying applicable requirement meets the criteria of 10 CFR 50.12.

Tier 1 information

The change processes for Tier 1 information are covered in paragraph VIII.A. Generic changes to Tier 1 are accomplished by rulemaking that amends the generic DCD and are governed by the standards in 10 CFR 52.63(a)(1). This provision provides that the Commission may not modify, change, rescind, or impose new requirements by rulemaking except where necessary either to bring the certification into compliance with the Commission'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. The rulemakings must include an opportunity for hearing with respect to the proposed change, as required by 10 CFR 52.63(a)(1), and the Commission expects such hearings to be conducted in accordance with 10 CFR Part 2, Subpart H. Departures from Tier 1 may occur in two ways: (1) the Commission may *order* a licensee to depart from Tier 1, as provided in subparagraph A.3; or (2) an applicant or licensee may request an *exemption* from Tier 1, as provided in subparagraph A.4. If the Commission seeks to order a licensee to depart from Tier 1, subparagraph A.3 requires that the Commission find both that the departure is necessary for adequate protection or for compliance, and that special circumstances are present. Subparagraph A.4 provides that exemptions from Tier 1 requested by an applicant or licensee are governed by the requirements of 10 CFR 52.63(b)(1) and 52.97(b), which provide an opportunity for a hearing. In addition, the Commission will not grant requests for exemptions that may result in a significant decrease in the level of safety otherwise provided by the design.

Tier 2 information

The change processes for the three different categories of Tier 2 information, viz., Tier 2, Tier 2*, and Tier 2* with a time of expiration, are set forth in paragraph VIII.B. The change process for Tier 2 has the same elements as the Tier 1 change process, but some of the standards for plant-specific orders and exemptions are different. The Commission adopted a "50.59-like" change process in accordance with its SRMs on SECY-90-377 and SECY-92-287A.

The process for generic Tier 2 changes (including changes to Tier 2* and Tier 2* with a time of expiration) tracks the process for generic Tier 1 changes. As set forth in subparagraph B.1, generic Tier 2 changes are accomplished by rulemaking amending the generic DCD, and are governed by the standards in 10 CFR 52.63(a)(1). This provision provides that the Commission may not modify, change, rescind or impose new requirements by rulemaking except where necessary either to bring the certification into compliance with the Commission's regulations applicable and in effect at the time of approval of the design certification or to assure adequate protection of the public health and safety or common defense and security. If a generic change is made to Tier 2* information, then the category and expiration, if necessary, of the new information would also be determined in the rulemaking and the appropriate change process for that new information would apply.

Departures from Tier 2 may occur in five ways: (1) the Commission may order a plant-specific departure, as set forth in subparagraph B.3; (2) an applicant or licensee may request an exemption from a Tier 2 requirement as set forth in subparagraph B.4; (3) a licensee may make a departure without prior NRC approval in accordance with subparagraph B.5 [the "50.59-like" process]; (4) the licensee may request NRC approval for proposed departures which do not meet the requirements in subparagraph B.5 as provided in subparagraph B.5.d; and (5) the licensee may request NRC approval for a departure from Tier 2* information under subparagraph B.6.

Similar to Commission-ordered Tier 1 departures and generic Tier 2 changes, Commission-ordered Tier 2 departures cannot be imposed except where necessary either to bring the certification into compliance with the Commission'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 subparagraph B.3. However, the special circumstances for the Commission-ordered Tier 2 departures do not have to outweigh any decrease in safety that may result from the reduction in standardization caused by the plant-specific order, as required by 10 CFR 52.63(a)(3). The Commission determined that it was not necessary to impose an additional limitation similar to that imposed on Tier 1 departures by 10 CFR 52.63(a)(3) and (b)(1). This type of additional limitation for standardization would unnecessarily restrict the flexibility of applicants and licensees with respect to Tier 2, which by its nature is not as safety significant as Tier 1.

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

Subparagraph B.5 allows an applicant or licensee to depart from Tier 2 information, without prior NRC approval, if the proposed departure does not involve a change to or departure from Tier 1 or Tier 2* information, technical specifications, or involves an unreviewed safety question (USQ) as defined in B.5.b and B.5.c of this paragraph. The technical specifications referred to in B.5.a and B.5.b of this paragraph are the technical specifications in Section 16.1 of the generic DCD, including bases, for departures made prior to issuance of the COL. After issuance of the COL, the plant-specific technical specifications are controlling under subparagraph B.5. The bases for the plant-specific technical specifications will be controlled by the bases control procedures for the plant-specific technical specifications (analogous to the bases control provision in the Improved Standard Technical Specifications). The definition of a USQ in B.5.b of this paragraph is similar to the definition in 10 CFR 50.59 and it applies to all information in Tier 2 except for the information that resolves the severe accident issues. The process for evaluating proposed tests or experiments not described in Tier 2 will be incorporated into the change process for the portion of the design that is outside the scope of this design certification. Although subparagraph B.5 does not specifically state, the Commission has determined that departures must also comply with all applicable regulations unless an exemption or other relief is obtained.

The Commission believes that it is important to preserve and maintain the resolution of severe accident issues just like all other safety issues that were resolved during the design certification review (refer to SRM on SECY-90-377). However, because of the increased uncertainty in severe accident issue resolutions, the Commission has adopted separate criteria in B.5.c for determining whether a departure from information that resolves severe accident issues constitutes a USQ. For purposes of applying the special criteria in B.5.c, severe accident resolutions are limited to design features when the intended function of the design feature is relied upon to resolve postulated accidents where the reactor core has melted and exited the reactor vessel and the containment is being challenged (severe accidents). These design features are identified in Section 1.9.5 of the DCD, with other issues, and are described in other sections of the DCD. Therefore, the location of design information in the DCD is not important to the application of this special procedure for severe accident issues. However, the special procedure in B.5.c does not apply to design features that resolve so-called beyond design basis accidents or other low probability events. The important aspect of this special procedure is that it is limited solely to severe accident design features, as defined above. Some design features may have intended functions to meet "design basis" requirements and to resolve "severe accidents." If these design features are reviewed under subparagraph VIII.B.5, then the appropriate criteria from either B.5.b or B.5.c are selected depending upon the function being changed.

An applicant or licensee that plans to depart from Tier 2 information, under subparagraph VIII.B.5, must prepare a safety evaluation which provides the bases for the determination that the proposed change does not involve an unreviewed safety question, a change to Tier 1 or Tier 2* information, or a change to the technical specifications, as explained above. In order to achieve the Commission's goals for design certification, the evaluation needs to consider all of the matters that were resolved in the DCD, such as generic issue resolutions that are relevant to the proposed departure. The benefits of the early resolution of safety issues would be lost if departures from the DCD were made that violated these resolutions without appropriate review. The evaluation of the relevant matters needs to consider the proposed departure over the full range of power operation from startup to shutdown, as it relates to anticipated operational occurrences, transients, design basis accidents, and severe accidents. The evaluation must also include a review of all relevant secondary references from the DCD because Tier 2 information intended to be treated as requirements is contained in the secondary references. The evaluation should consider Tables 14.3-1 through 14.3-8 and 19.59-29 of the generic DCD to ensure that the proposed change does not impact Tier 1. These tables contain various cross-references from the safety analyses and probabilistic risk assessment in Tier 2 to the important parameters that were included in Tier 1. Although many issues and analyses could have been cross-referenced, the listings in these tables were developed only for key analyses for the AP600 design. Westinghouse provided more detailed cross-references for important analysis assumptions that are included in Tier 1 in its revised response to RAI 640.60 (DCP/NRC 1440 - September 15, 1998).

If a proposed departure from Tier 2 involves a change to or departure from Tier 1 or Tier 2* information, technical specifications, or otherwise constitutes a USQ, then the applicant or licensee must obtain NRC approval through the appropriate process set forth in this appendix before implementing the proposed departure. The NRC does not endorse NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations," for performing safety evaluations required by subparagraph VIII.B.5 of this appendix. However, the NRC will work with industry, if it is desired, to develop an appropriate guidance document for processing proposed changes under paragraph VIII.B of this appendix.

A party to an adjudicatory proceeding (*e.g.*, for issuance of a combined license) who believes that an applicant or licensee has not complied with subparagraph VIII.B.5 when departing from Tier 2 information, may petition to admit such a contention into the proceeding under B.5.f. This provision was included because an incorrect departure from the requirements of this appendix essentially places the departure outside of the scope of the Commission's safety finding in the design certification rulemaking. Therefore, it follows that properly-founded contentions alleging such incorrectly-implemented departures cannot be considered "resolved" by this rulemaking. As set forth in B.5.fof paragraph VIII.B, the petition must comply with the requirements of § 2.714(b)(2) and show that the departure does not comply with subparagraph B.5. Any other party may file a response to the petition. If on the basis of the petition and any responses, the presiding officer in the proceeding determines that the required showing has been made, the matter shall be certified to the Commission for its final determination. In the absence of a proceeding, petitions alleging non-conformance with subparagraph B.5 requirements applicable to Tier 2 departures will be treated as petitions for enforcement action under 10 CFR 2.206.

Subparagraph B.6 provides a process for departing from Tier 2* information. The creation of and restrictions on changing Tier 2* information resulted from the development of the Tier 1 information for the ABWR design. During this development process, the applicants for design certification requested that the amount of information in Tier 1 be minimized to provide additional flexibility for an applicant or licensee who references this appendix. Also, many codes, standards, and design processes, which were not specified in Tier 1, that are acceptable for meeting ITAAC were specified in Tier 2. The result of these actions is that certain significant information only exists in Tier 2 and the Commission does not want this significant information to be changed without prior NRC approval. This Tier 2* information is identified in the generic DCD with italicized text and brackets.

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

Certain preoperational tests in B.6.c of paragraph VIII.B are designated to be performed only for the first plant or first three plants that reference this appendix. Westinghouse's basis for performing these "first-plant-only" and "first-three-plants-only" preoperational tests is provided in Section 14.2.5 of the DCD. The NRC staff found Westinghouse's basis for performing these tests and its justification for only performing the tests on the first-plant or first-three-plants acceptable. The NRC staff's decision was based on the need to verify that plant-specific manufacturing and/or construction variations do not adversely impact the predicted performance of certain passive safety systems, while recognizing that these special tests will result in significant thermal transients being applied to critical plant components. The NRC staff believes that the range of manufacturing or construction variations that could adversely affect the relevant passive safety systems will be adequately disclosed after performing the designated tests on the first plant, or the first three plants, as applicable. The COL action item in Section 14.4.6 of the DCD states that subsequent plants shall either perform these preoperational tests or justify that the results of the first-plant-only or first-three-plant-only tests are applicable to the subsequent plant. The Tier 2* designation for these tests will expire after the first plant or first three plants complete these tests, as indicated in B.6.c of paragraph VIII.B.

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

Operational Requirements

The change process for technical specifications and other operational requirements in the DCD is set forth in paragraph VIII.C of this appendix. This change process has elements similar to the Tier 1 and Tier 2 change process in paragraphs VIII.A and VIII.B, but with significantly different change standards. Because of the different finality status for technical specifications and other operational requirements (refer to III.F of this SOC), the Commission decided to designate a special category of information, consisting of the technical specifications and other operational requirements, with its own change process in paragraph VIII.C. The key to using the change processes in Section VIII is to determine if the proposed change or departure requires a change to a design feature described in the generic DCD. If a design change is required, then the appropriate change process in paragraph VIII.A or VIII.B applies. However, if a proposed change to the technical specifications or other operational requirements does not require a change to a design feature in the generic DCD, then paragraph VIII.C applies. The language in paragraph VIII.C also distinguishes between generic (Section 16.1 of DCD) and plant-specific technical specifications to account for the different treatment and finality accorded technical specifications before and after a license is issued.

The process in subparagraph VIII.C.1 for making generic changes to the generic technical specifications in Section 16.1 of the DCD or other operational requirements in the generic DCD is accomplished by rulemaking and governed by the backfit standards in 10 CFR 50.109. The determination of whether the generic technical specifications and other operational requirements were completely reviewed and approved in the design certification rulemaking is based upon the extent to which an NRC safety conclusion in the FSER is being modified or changed. If it cannot be determined that the technical specification or operational requirement was comprehensively reviewed and finalized in the design certification rulemaking, then there is no backfit restriction under 10 CFR 50.109 because no prior position was taken on this safety matter. Some generic technical specifications contain bracketed values, which clearly indicate that the NRC staff's review was not complete. Generic changes made under subparagraph VIII.C.1 are applicable to all applicants or licensees (refer to subparagraph VIII.C.2), unless the change is irrelevant because of a plant-specific departure.

Plant-specific departures may occur by either a Commission order under subparagraph VIII.C.3 or an applicant's exemption request under subparagraph VIII.C.4. The basis for determining if the technical specification or operational requirement was completely reviewed and approved for these processes is the same as for subparagraph VIII.C.1. If the technical specification or operational requirement was comprehensively reviewed and finalized in the design certification rulemaking, then the Commission must demonstrate that special circumstances are present before ordering a plant-specific departure. If not, there is no restriction on plant-specific changes to the technical specifications or operational requirements, prior to issuance of a license, provided a design change is not required. Although the generic technical specifications were reviewed by the NRC staff to facilitate the design certification review, the Commission 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 is similar to other issues in a licensing hearing, except that the petitioner must also demonstrate why special circumstances are present (subparagraph VIII.C.5).

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

I. Inspections, tests, analyses, and acceptance criteria (ITAAC).

The purpose of Section IX of this appendix is to set forth how the ITAAC in Tier 1 of this design certification rule are to be treated in a license proceeding. Paragraph A restates the responsibilities of an applicant or licensee for performing and successfully completing ITAAC, and notifying the NRC of such completion. Subparagraph A.1 makes it clear that an applicant may proceed at its own risk with design and procurement activities subject to ITAAC, and that a licensee may proceed at its own risk with design, procurement, construction, and preoperational testing activities subject to an ITAAC, even though the NRC may not have found that any particular ITAAC has been successfully completed. Subparagraph A.2 requires the licensee to notify the NRC that the required inspections, tests, and analyses in the ITAAC have been completed and that the acceptance criteria have been met.

Subparagraphs B.1 and B.2 essentially reiterate the NRC's responsibilities with respect to ITAAC as set forth in 10 CFR 52.99 and 52.103(g). Finally, subparagraph B.3 states that ITAAC do not, by virtue of their inclusion in the DCD, constitute regulatory requirements after the licensee has received authorization to load fuel or for renewal of the license. However, subsequent modifications must comply with the design descriptions in the DCD unless the applicable requirements in 10 CFR 52.97 and Section VIII of this appendix have been complied with. As discussed in paragraph III.D of this SOC, the Commission will defer a determination of the applicability of ITAAC and their effect in terms of issue resolution in 10 CFR Part 50 licensing proceedings to such time that a Part 50 applicant decides to reference this appendix.

J. Records and Reporting.

The purpose of Section X of this appendix is to set forth the requirements for maintaining records of changes to and departures from the generic DCD, which are to be reflected in the plant-specific DCD. Section X also sets forth the requirements for submitting reports (including updates to the plant-specific DCD) to the NRC. This section of the appendix is similar to the requirements for records and reports in 10 CFR Part 50, except for minor differences in information collection and reporting requirements, as discussed in V of this SOC. Subparagraph X.A.1 of this appendix requires that a generic DCD and the proprietary and safeguards information referenced in the generic DCD be maintained by the applicant for this rule. The generic DCD was developed, in part, to meet the requirements for incorporation by reference, including availability requirements. Therefore, the proprietary and safeguards information could not be included in the generic DCD because it is not publicly available. However, the proprietary and safeguards information was reviewed by the NRC and, as stated in subparagraph VI.B.2 of this appendix, the Commission considers the information to be resolved within the meaning of 10 CFR 52.63(a)(4). Because this information is not in the generic DCD, the proprietary and safeguards information, or its equivalent, is required to be provided by an applicant for a license. Therefore, to ensure that this information will be available, a requirement for the design certification applicant to maintain the proprietary and safeguards information was added to subparagraph X.A.1 of this appendix. The acceptable version of the proprietary and safeguards information is identified (referenced) in the version of the DCD that is incorporated into this rule. The generic DCD and the acceptable version of the proprietary and safeguards information must be maintained for the period of time that this appendix may be referenced.

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

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

Subparagraphs B.1 and B.2 of this appendix establish reporting requirements for applicants or licensees that reference this rule that are similar to the reporting requirements in 10 CFR Part 50. For currently operating plants, a licensee is required to maintain records of the basis for any design changes to the facility made under 10 CFR 50.59. Section 50.59(b)(2) requires a licensee to provide a summary report of these changes to the NRC annually, or along with updates to the facility final safety analysis report under 10 CFR 50.71(e). Section 50.71(e)(4) requires that these updates be submitted annually, or 6 months after each refueling outage if the interval between successive updates does not exceed 24 months.

The reporting requirements in subparagraph B.3 of this appendix vary according to four different time periods during a facilities' lifetime. Under B.3.a of paragraph X.B, if an applicant that references this rule decides to make departures from the generic DCD, then the departures and any updates to the plant-specific DCD must be submitted with the initial application for a license. Under B.3.b of paragraph X.B, the applicant may submit any subsequent reports and updates 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. Under B.3.c of paragraph X.B, summary reports must be submitted quarterly during the period of facility construction. This increase in frequency of summary reports of departures from the plant-specific DCD is in response to the Commission's guidance on reporting frequency in its SRM on SECY-90-377, dated February 15, 1991.

Quarterly reporting of design changes during the period of construction is necessary to closely monitor the status and progress of the construction of the plant. To make its finding under 10 CFR 52.99, the NRC must monitor the design changes made in accordance with Section VIII of this appendix. The ITAAC verify that the as-built facility conforms with the approved design and emphasizes design reconciliation and design verification. Quarterly reporting of design changes is particularly important in times where the number of design changes could be significant, such as during the procurement of components and equipment, detailed design of the plant at the start of construction, and during preoperational testing. The frequency of updates to the plant-specific DCD is not increased during facility construction. After the facility begins operation, the frequency of reporting reverts to the requirement in X.B.3.d of paragraph X.B, which is consistent with the requirement for plants licensed under 10 CFR Part 50.

IV. Finding of No Significant Environmental Impact: Availability

The Commission has determined under the National Environmental Policy Act of 1969, as amended (NEPA), and the Commission's regulations in 10 CFR Part 51, Subpart A, that this design certification rule is not a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement (EIS) is not required. The basis for this determination, as documented in the final environmental assessment, is that this amendment to 10 CFR Part 52 does not authorize the siting, construction, or operation of a facility using the AP600 design; it only codifies the AP600 design in a rule. The NRC will evaluate the environmental impacts and issue an EIS, as appropriate, in accordance with NEPA as part of the application(s) for the construction and operation of a facility.

In addition, as part of the final environmental assessment for the AP600 design, the NRC reviewed Westinghouse's evaluation of various design alternatives to prevent and mitigate severe accidents in Appendix 1B of the AP600 Standard Safety Analysis Report (SSAR). The Commission finds that Westinghouse's evaluation provides a reasonable assurance that certifying the AP600 design will not exclude severe accident mitigation design alternatives for a future facility that would prove cost beneficial had they been considered as part of the original design certification application. These issues are considered resolved for the AP600 design.

The final environmental assessment (EA), upon which the Commission's finding of no significant impact is based, and the AP600 SSAR are available for examination and copying at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC. Single copies of the EA are also available from Jerry N. Wilson, Mailstop O-12 G15, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

V. Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These requirements were approved by the Office of Management and Budget (OMB) on August 10, 1999 (OMB #3150-0151). If an application is submitted, the additional public reporting burden for this information collection is estimated to average 8 person-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.

Send comments on any aspect of this information collection, including suggestions for reducing the burden, to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail at BJS1@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0151), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

VI. Regulatory Analysis

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

VII. Regulatory Flexibility Act Certification

As required by the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission certifies that this final rule will not have a significant economic impact upon a substantial number of small entities. The final rule provides for certification of a nuclear power plant design. Neither the design certification applicant, nor prospective nuclear power plant licensees who reference this design certification rule, fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121.

VIII. Backfit Analysis

The Commission has determined that the backfit rule, 10 CFR 50.109, does not apply to this amendment because it does not impose new or changed requirements on existing 10 CFR Part 50 licensees. Therefore, a backfit analysis was not prepared for this rule.

IX. Small Business Regulatory Enforcement Fairness Act

As required by the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs, OMB.

X. National Technology Transfer and Advancement Act

The National Technology and Transfer Act of 1995 (Act), Public Law 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. This rule provides for certification of a nuclear power plant design. 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 Commission approvals of specific nuclear power plant designs by rulemaking. Furthermore, design certification rulemakings are initiated by an applicant for a design certification, rather than the NRC. For these reasons, the Commission concludes that the Act does not apply to this rule.

List of Subjects in 10 CFR Part 52

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

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

Part 52 - Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants

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

AUTHORITY: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1243, 1244, 1246, 1246, as amended (42 U.S.C. 5841, 5842, 5846).

2. In § 52.8, paragraph (b) is revised to read as follows:

§ 52.8 Information collection requirements: OMB approval.

* * * * *

(b) The approved information collection requirements contained in this part appear in §§ 52.15, 52.17, 52.29, 52.35, 52.45, 52.47, 52.51, 52.57, 52.63, 52.75, 52.77, 52.78, 52.79, 52.89, 52.91, 52.99, and appendices A, B, and C.

3. A new Appendix C to 10 CFR Part 52 is added to read as follows:

Appendix C To Part 52 - Design Certification Rule for the AP600 Design

I. INTRODUCTION

Appendix C constitutes the standard design certification for the AP600[1] design, in accordance with 10 CFR Part 52, Subpart B. The applicant for certification of the AP600 design is Westinghouse Electric Company LLC.

II. DEFINITIONS

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

B. *Generic technical specifications* 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 the document, maintained by an applicant or licensee who references this appendix, consisting of the information in the generic DCD, as modified and supplemented by the plant-specific departures and exemptions made under Section VIII of this appendix.

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

1. Information required by 10 CFR 52.47, with the exception of generic technical specifications and conceptual design information;
2. Information required for a final safety analysis report under 10 CFR 50.34;
3. 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
4. Combined license (COL) action items (combined license information), which identify certain matters that shall be addressed in the site-specific portion of the final safety analysis report (FSAR) by an applicant who references this appendix. These items constitute information requirements but are not the only acceptable set of information in the FSAR. An applicant may depart from or omit these items, provided that the departure or omission is identified and justified in the FSAR. After issuance of a construction permit or COL, these items

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[1]AP600 is a trademark of Westinghouse Electric Company LLC

are not requirements for the licensee unless such items are restated in the FSAR.

5. The investment protection short-term availability controls in Section 16.3 of the DCD.

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

G. All other terms in this appendix have the meaning set out in 10 CFR 50.2, 10 CFR 52.3, or Section 11 of the Atomic Energy Act of 1954, as amended, as applicable.

III. SCOPE AND CONTENTS

A. Tier 1, Tier 2 (including the investment protection short-term availability controls in Section 16.3), and the generic technical specifications in the AP600 DCD (9/99 revision) are approved for incorporation by reference by the Director of the Office of the Federal Register on **[Insert date of approval]** in accordance with 5 U.S.C. 552(a) and 1 CFR Part 51. Copies of the generic DCD may be obtained from Mr. Brian A. McIntyre, Manager, Advanced Plant Safety and Licensing, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, PA 15230-0355. A copy of the generic DCD is available for examination and copying at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC 20555-0001. Copies are also available for examination at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20582 and the Office of the Federal Register, 800 North Capitol Street, NW., Suite 700, Washington, DC.

B. An applicant or licensee referencing this appendix, in accordance with Section IV of this appendix, shall incorporate by reference and comply with the requirements of this appendix, including Tier 1, Tier 2 (including the investment protection short-term availability controls in Section 16.3), and the generic technical specifications except as otherwise provided in this appendix. Conceptual design information in the generic DCD and the evaluation of severe accident mitigation design alternatives in Appendix 1B of the generic DCD are not part of this appendix.

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

D. If there is a conflict between the generic DCD and either the application for design certification of the AP600 design or NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," (FSER), then the generic DCD controls.

E. Design activities for structures, systems, and components that are wholly outside the scope of this appendix may be performed using site-specific design parameters, 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 license that wishes to reference this appendix shall, in addition to complying with the requirements of 10 CFR 52.77, 52.78, and 52.79, 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 information and utilizing the same organization and numbering as the AP600 DCD, 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 X.B of this appendix;

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

d. Information demonstrating compliance with the site parameters and interface requirements;

e. Information that addresses the COL action items; and

f. Information required by 10 CFR 52.47(a) that is not within the scope of this appendix.

3. Physically include, in the plant-specific DCD, the proprietary and safeguards information referenced in the AP600 DCD.

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 Part 50.

V. APPLICABLE REGULATIONS

A. Except as indicated in paragraph B of this section, the regulations that apply to the AP600 design are in 10 CFR Parts 20, 50, 73, and 100, codified as of **[insert date that rule is signed]**, that are applicable and technically relevant, as described in the FSER (NUREG-1512).

B. The AP600 design is exempt from portions of the following regulations:

1. Paragraph (a)(1) of 10 CFR 50.34 - whole body dose criterion;
2. Paragraph (f)(2)(iv) of 10 CFR 50.34 - Plant Safety Parameter Display Console;
3. Paragraphs (f)(2)(vii), (viii), (xxvi), and (xxviii) of 10 CFR 50.34 - Accident Source Term in TID 14844;
4. Paragraph (c)(1) of 10 CFR 50.62 - Auxiliary (or emergency) feedwater system;
5. Appendix A to 10 CFR Part 50, GDC 17 - Offsite Power Sources; and
6. Appendix A to 10 CFR Part 50, GDC 19 - whole body dose criterion.

VI. ISSUE RESOLUTION

A. The Commission has determined that the structures, systems, components, and design features of the AP600 design comply with the provisions of the Atomic Energy Act of 1954, as amended, and the applicable regulations identified in Section V of this appendix; and therefore, provide adequate protection to the health and safety of the public. A conclusion that a matter is resolved includes the finding that additional or alternative structures, systems, components, design features, design criteria, testing, analyses, acceptance criteria, or justifications are not necessary for the AP600 design.

B. The Commission considers the following matters resolved within the meaning of 10 CFR 52.63(a)(4) in subsequent proceedings for issuance of a combined license, amendment of a combined license, or renewal of a combined license, proceedings held pursuant to 10 CFR 52.103, and enforcement proceedings involving plants referencing this appendix:

1. All nuclear safety issues, except for the generic technical specifications and other operational requirements, associated with the information in the FSER, Tier 1, Tier 2 (including referenced information, which the context indicates is intended as requirements, and the investment protection short-term availability controls in Section 16.3), and the rulemaking record for certification of the AP600 design;

2. All nuclear safety and safeguards issues associated with the information in proprietary and safeguards documents, referenced and in context, are intended as requirements in the generic DCD for the AP600 design;

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

4. All exemptions from the DCD pursuant to and in compliance with the change processes in Sections VIII.A.4 and VIII.B.4 of this appendix, but only for that proceeding;

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

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

7. All environmental issues concerning severe accident mitigation design alternatives (SAMDAs) associated with the information in the NRC's environmental assessment for the AP600 design and Appendix 1B of the generic DCD, for plants referencing this appendix whose site parameters are within those specified in the SAMDA evaluation.

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 10 CFR 52.63(a)(4). 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 in accordance with the change processes in Section VIII of this appendix, the Commission may not require an applicant or licensee who references this appendix to:

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

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

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

E.1. Persons who wish to review proprietary and safeguards information or other secondary references in the AP600 DCD, in order to request or participate in the hearing required by 10 CFR 52.85 or the hearing provided under 10 CFR 52.103, or to request or participate in any other hearing relating to this appendix in which interested persons have adjudicatory hearing rights, shall first request access to such information from Westinghouse. The request must state *with particularity*:

a. The nature of the proprietary or other information sought;

b. The reason why the information currently available to the public in the NRC's public document room is insufficient;

c. The relevance of the requested information to the hearing issue(s) which the person proposes to raise; and

d. A showing that the requesting person has the capability to understand and utilize the requested information.

2. If a person claims that the information is necessary to prepare a request for hearing, the request must be filed no later than 15 days after publication in the Federal Register of the notice required either by 10 CFR 52.85 or 10 CFR 52.103. If Westinghouse declines to provide the information sought, Westinghouse shall send a written response within ten (10) days of receiving the request to the requesting person setting forth with particularity the reasons for its refusal. The person may then request the Commission (or presiding officer, if a proceeding has been established) to order disclosure. The person shall include copies of the original request (and any subsequent clarifying information provided by the requesting party to the applicant) and the applicant's response. The Commission and presiding officer shall base their decisions *solely* on the person's original request (including any clarifying information provided by the requesting person to Westinghouse), and Westinghouse's response. The Commission and presiding officer may order Westinghouse to provide access to some or all of the requested information, subject to an appropriate non-disclosure agreement.

VII. DURATION OF THIS APPENDIX

This appendix may be referenced for a period of 15 years from **[Insert date 30 days after publication of the final rule in the Federal Register]**, except as provided for in 10 CFR 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 10 CFR 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 10 CFR 52.63(a)(3).

4. Exemptions from Tier 1 information are governed by the requirements in 10 CFR 52.63(b)(1) and § 52.97(b). The Commission will deny a request for an exemption from Tier 1, if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design.

B. Tier 2 information.

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

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

3. The Commission may not require new requirements on Tier 2 information by plant-specific order while this appendix is in effect under §§ 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 assure adequate protection of the public health and safety or the common defense and security; and

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

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

5.a. An applicant or licensee who references this appendix may depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the technical specifications, or involves an unreviewed safety question as defined in paragraphs B.5.b and 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, involves an unreviewed safety question if --

(1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the plant-specific DCD may be increased;

(2) A possibility for an accident or malfunction of a different type than any evaluated previously in the plant-specific DCD may be created; or

(3) The margin of safety as defined in the basis for any technical specification is reduced.

c. A proposed departure from Tier 2 affecting resolution of a severe accident issue identified in the plant-specific DCD, involves an unreviewed safety question if --

(1) There is a substantial increase in the probability of a severe accident such that a particular 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 severe accident previously reviewed.

d. If a departure involves an unreviewed safety question as defined in paragraph B.5 of this section, it is governed by 10 CFR 50.90.

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

f. A party to an adjudicatory proceeding for either the issuance, amendment, or renewal of a license or for operation under 10 CFR 52.103(a), who believes that an applicant or licensee who references this appendix has not complied with VIII.B.5 of this appendix when departing from Tier 2 information, may petition to admit into the proceeding such a contention. In addition to compliance with the general requirements of 10 CFR 2.714(b)(2), the petition must demonstrate that the departure does not comply with VIII.B.5 of this appendix. Further, the petition must demonstrate that the change bears on an asserted noncompliance with an ITAAC acceptance criterion in the case of a 10 CFR 52.103 preoperational hearing, or that the change bears directly on the amendment request in the case of a hearing on a license amendment. Any other party may file a response. If, on the basis of the petition and any response, the presiding officer determines that a sufficient showing has been made, the presiding officer shall certify the matter directly to the Commission for determination of the admissibility of the contention. The Commission may admit such a contention if it determines the petition raises a genuine issue of fact regarding compliance with VIII.B.5 of this appendix.

6.a. An applicant who references this appendix may not depart from Tier 2* information, which is designated with italicized text or brackets and an asterisk in the generic DCD, without NRC approval. The departure will not be considered a resolved issue, within the meaning of Section VI of this appendix and 10 CFR 52.63(a)(4).

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

(1) Maximum fuel rod average burn-up.

(2) Fuel principal design requirements.

(3) Fuel criteria evaluation process.

(4) Fire areas.

(5) Human factors engineering.

c. A licensee who references this appendix may not, before the plant first achieves full power following the finding required by 10 CFR 52.103(g), depart from the following Tier 2* matters except in accordance with paragraph B.6.b of this section. After the plant first achieves full power, the following Tier 2* matters revert to Tier 2 status and are thereafter subject to the departure provisions in paragraph B.5 of this section.

(1) Nuclear Island structural dimensions.

(2) ASME Boiler & Pressure Vessel Code, Section III, and Code Case N-284.

(3) Design Summary of Critical Sections.

(4) ACI 318, ACI 349, and ANSI/AISC –690.

(5) Definition of critical locations and thicknesses.

(6) Seismic qualification methods and standards.

(7) Nuclear design of fuel and reactivity control system, except burn-up limit.

(8) Motor-operated and power-operated valves.

(9) Instrumentation & control system design processes, methods, and standards.

(10) PRHR natural circulation test (first plant only).

(11) ADS and CMT verification tests (first three plants only).

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

C. Operational requirements.

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

2. Generic changes to generic technical specifications and other operational requirements 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 C.3 or C.4 of this section.

3. The Commission may require plant-specific departures on generic technical specifications 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 2.758(b) are present. The Commission may modify or supplement generic technical specifications and other operational requirements that were not completely reviewed and approved or require additional technical specifications 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 technical specifications or other operational requirements. 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 grant of an exemption must be subject to litigation in the same manner as other issues material to the license hearing.

5. A party to an adjudicatory proceeding for either the issuance, amendment, or renewal of a license or for operation under 10 CFR 52.103(a), who believes that an operational requirement approved in the DCD or a technical specification derived from the generic technical specifications must be changed may petition to admit into the proceeding such a contention. Such petition must comply with the general requirements of 10 CFR 2.714(b)(2) and must demonstrate why special circumstances as defined in 10 CFR 2.758(b) are present, or for 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 thereto. 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 technical specifications or other operational requirements are subject to a hearing as part of the license proceeding.

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

IX. INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA (ITAAC)

A.1 An applicant or licensee who references this appendix shall perform and demonstrate conformance with the ITAAC before fuel load. With respect to activities subject to an ITAAC, an applicant for a license may proceed at its own risk with design and procurement activities, and a licensee may proceed at its own risk with design, procurement, construction, and preoperational activities, even though the NRC may not have found that any particular ITAAC has been satisfied.

2. The licensee who references this appendix shall notify the NRC that the required inspections, tests, and analyses in the ITAAC have been successfully completed and that the corresponding acceptance criteria have been met.

3. In the event that an activity is subject to an ITAAC, and the applicant or licensee who references this appendix has not demonstrated that the ITAAC has been satisfied, the applicant or licensee may either take corrective actions to successfully complete that ITAAC, request an exemption from the ITAAC in accordance with Section VIII of this appendix and 10 CFR 52.97(b), or petition for rulemaking to amend this appendix by changing the requirements of the ITAAC, under 10 CFR 2.802 and 52.97(b). Such rulemaking changes to the ITAAC must meet the requirements of paragraph VIII.A.1 of this appendix.

B.1 The NRC shall ensure that the required inspections, tests, and analyses in the ITAAC are performed. The NRC shall verify that the inspections; tests, and analyses referenced by the licensee have been successfully completed and, based solely thereon, find the prescribed acceptance criteria have been met. At appropriate intervals during construction, the NRC shall publish notices of the successful completion of ITAAC in the *Federal Register.*

2. In accordance with 10 CFR 52.99 and 52.103(g), the Commission shall find that the acceptance criteria in the ITAAC for the license are met before fuel load.

3. After the Commission has made the finding required by 10 CFR 52.103(g), the ITAAC do not, by virtue of their inclusion within the DCD, constitute regulatory requirements either for licensees or for renewal of the license; except for specific ITAAC, which are the subject of a Section 103(a) hearing, their expiration will occur upon final Commission action in such proceeding. However, subsequent modifications must comply with the Tier 1 and Tier 2 design descriptions in the plant-specific DCD unless the licensee has complied with the applicable requirements of 10 CFR 52.97 and Section VIII of this appendix.

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 to Tier 1 and Tier 2. The applicant shall maintain the proprietary 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 pursuant to Section VIII of this appendix throughout the period of application

and for the term of the license (including any period of renewal).

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

B. Reporting.

1. An applicant or licensee who references this appendix shall submit a report to the NRC containing a brief description of any departures from the plant-specific DCD, including a summary of the safety evaluation of each. This report must be filed in accordance with the filing requirements applicable to reports in 10 CFR 50.4.

2. An applicant or licensee who references this appendix shall submit updates to its plant-specific DCD, which reflect the generic changes to the generic DCD and the plant-specific departures made pursuant to Section VIII of this appendix. These updates shall be filed in accordance with the filing requirements applicable to final safety analysis report updates in 10 CFR 50.4 and 50.71(e).

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

a. On the date that an application for a license referencing this appendix is submitted, the application shall include the report and any updates to the plant-specific DCD.

b. During the interval from the date of application to the date of issuance of a license, the report and any updates to the plant-specific DCD must be submitted annually and may be submitted along with amendments to the application.

c. During the interval from the date of issuance of a license to the date the Commission makes its findings under 10 CFR 52.103(g), the report must be submitted quarterly. Updates to the plant-specific DCD must be submitted annually.

d. After the Commission has made its finding under 10 CFR 52.103(g), reports and updates to the plant-specific DCD may be submitted annually or along with updates to the site-specific portion of the final safety analysis report for the facility at the intervals required by 10 CFR 50.71(e), or at shorter intervals as specified in the license.

Dated at Rockville, Maryland, this **day of** 1999.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,
Secretary of the Commission

ENVIRONMENTAL ASSESSMENT BY THE U.S. NUCLEAR REGULATORY COMMISSION RELATING TO THE CERTIFICATION OF THE AP600 STANDARD PLANT DESIGN DOCKET NO. 52-003

Attachment 2

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1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC or Commission) has issued a design certification for the Advanced Passive 600 (AP600) design in response to an application submitted on June 16, 1992, by Westinghouse Electric Company LLC (hereinafter referred to as Westinghouse). A design certification is a rulemaking that amends 10 CFR Part 52.

This report presents an environmental assessment (EA) for this rulemaking that the NRC has prepared in accordance with the requirements of 10 CFR Part 51 and the National Environmental Policy Act of 1969 (NEPA), as amended. This EA addresses the environmental impacts of issuing a design certification. In addition, this report addresses severe accident mitigation design alternatives (SAMDAs), which the NRC has decided to consider as part of this final EA for the AP600 design. This report does not address the environmental impacts of constructing and operating a facility, which references the AP600 design certification, at a particular site; such impacts will be evaluated as part of any application(s) for the siting, construction, or operation of a facility.

As detailed in Section 4.0 of this report, the NRC determined that issuing this design certification does not constitute a major Federal action significantly affecting the quality of the human environment. This finding of no significant impact is based on the fact that the design certification would not independently authorize the siting, construction, or operation of a facility of an AP600 reactor design. Rather, the certification would merely codify the AP600 design in a rule that could be referenced in a construction permit (CP), early site permit (ESP), combined operating license (COL), or operating license (OL) application. Further, because the certification is a rule, it does not involve any resources that would have alternative uses. Therefore, the NRC has decided not to prepare an environmental impact statement (EIS) in connection with this action.

In addition, the NRC also reviewed Westinghouse's evaluation of SAMDAs that generically apply to the AP600 design. On that basis, the NRC found that the evaluation provides reasonable assurance that certifying the AP600 design would not exclude SAMDAs for a future facility that could prove cost beneficial had they been considered as part of the original design certification application.

2.0 THE NEED FOR THE PROPOSED ACTION

The NRC has long sought the safety benefits of commercial nuclear power plant standardization, as well as the early resolution of design issues and the finality of these resolutions. The NRC plans to achieve these benefits by certifying nuclear plant designs. Subpart B to 10 CFR Part 52 allows for certification in the form of rulemaking of an essentially complete nuclear plant design. The proposed action would amend 10 CFR Part 52 to certify the AP600 design. The amendment would allow prospective licensees to reference the certified AP600 design as part of an ESP or a COL application under 10 CFR Part 52.

3.0 ALTERNATIVES TO THE PROPOSED ACTION

The NRC had two alternatives to certifying the AP600 design in an amendment to 10 CFR Part 52. The NRC could take no action to approve the design or issue a final design approval

(FDA) without certifying the design. As with the proposed action, these alternatives would not have a significant impact on the quality of the human environment because they would not authorize the siting, construction, or operation of a facility.

In the first case, the NRC would not approve the design. Therefore, a facility that would be built using the AP600 design would require licensing under 10 CFR Part 50 or 10 CFR Part 52, Subpart C, as a custom plant application. Moreover, all design issues would have to be considered as part of each application to construct and operate an AP600 facility at a particular site. As a result, this alternative would neither achieve the benefits of standardization, provide for early resolution of design issues, nor permit finality for the resolved design issues.

In the second case, the NRC would issue an FDA under Appendix O to 10 CFR Part 52, but would not certify the design in a rulemaking. Therefore, although the NRC would have approved the design, the design could be modified and, thus, would require reevaluation as part of each application to construct and operate a facility of an AP600 design at a particular site. This alternative would provide for early internal NRC resolution of design issues, to the extent that the design remains unchanged at the facility application stage, but would not achieve the benefits of standardization nor permit overall finality for the resolved design issues.

The NRC sees no advantage in either alternative compared to the design certification rulemaking proposed for the AP600 design. Although neither the alternatives nor the proposed design certification rulemaking would significantly affect the quality of the human environment, the rulemaking achieves the benefits of standardization, permits early resolution of design issues, and provides finality for the resolved design issues (including SAMDAs) that are within the scope of the design certification. Therefore, the NRC concludes that the alternatives to rulemaking would not achieve the objectives that the Commission intended by certifying the AP600 design pursuant to 10 CFR Part 52, Subpart B.

3.1 Severe Accident Mitigation Design Alternatives (SAMDAs)

Consistent with its objectives of standardization and early resolution of design issues, the Commission decided to evaluate SAMDAs as part of the design certification for the AP600 design. In a 1985 policy statement, 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 considered to be those analyzed in accordance with the NRC's Standard Review Plan (NUREG-0800) and documented in Chapter 15 of the AP600 Design Control Document (DCD).

As part of its design certification application, Westinghouse performed a probabilistic risk assessment (PRA) for the AP600 design to achieve the following objectives:

- Identify the dominant severe accident sequences and associated source terms for the design.
- Modify the design, on the bases of PRA insights, to prevent or mitigate severe accidents and reduce the risk of severe accidents.
- Provide a basis for concluding that all reasonable steps have been taken to reduce the chances of occurrence, and to mitigate the consequences, of severe accidents.

Westinghouse's PRA analysis is presented in Chapter 19 of the AP600 Standard Safety Analysis Report (SSAR).

In addition to considering alternatives to the rulemaking process as discussed in Section 3.0, applicants for reactor design certification or CPs must also consider alternative design features for severe accidents consistent with the requirements of 10 CFR Part 50, as well as a court ruling related to NEPA. These requirements can be summarized as follows:

- 10 CFR 50.34(f)(1)(i) requires the applicant to perform a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.

- The U.S. Court of Appeals decision, in *Limerick Ecology Action v. NRC,* 869 F.2d 719 (3rd Cir. 1989), effectively requires the NRC to include consideration of certain SAMDAs in the environmental impact review performed under Section 102(2)(c) of NEPA.

Although these two requirements are not directly related, they share a common purpose to consider alternatives to the proposed design, to evaluate potential alternative improvements in the plant design which increase safety performance during severe accidents, and to prevent reasonable alternatives from being foreclosed. It should be noted that the Commission is not required to consider alternatives to the design in this EA; however, as a matter of discretion, the Commission has determined that considering SAMDAs concomitant with the rulemaking is consistent with the intent of 10 CFR Part 52 for early resolution of issues, finality of design issues resolution, and achieving the benefits of standardization.

In its decision in *Limerick Ecology Action v. NRC,* the Court of Appeals for the Third Circuit expressed its opinion that it would likely be difficult to evaluate SAMDAs for NEPA purposes on a generic basis. However, the NRC has determined that generic evaluation of SAMDAs for the AP600 standard design is warranted for two significant reasons. First, the design and construction of all plants referencing the certified AP600 design will be governed by the rule certifying a single design. Second, the site parameters specified in the rule and the AP600 DCD establish the consequences for a reasonable enveloping set of SAMDAs for the AP600 design. The low residual risk of the AP600 design and the limited potential for further risk reductions provides high confidence that additional cost beneficial SAMDAs would not be found for those sites within the site parameter envelope assumed for the AP600 environmental SAMDA assessment. If the actual parameters for a particular site exceed those assumed in the rule and the SSAR, then SAMDAs would have to be reevaluated in the site-specific environmental report and the EIS. If the actual parameters for a postulated site are bounded by those assumed in the rule and the DCD, then the SAMDA analysis can be incorporated by reference in the site-specific EIS.

3.2 Potential SAMDAs Identified by Westinghouse

To identify candidate design alternatives, Westinghouse reviewed the design alternatives for other plants including Limerick, Comanche Peak, and the ABB-Combustion Engineering, Inc. (ABB-CE) System 80+ design. Westinghouse also reviewed the results of the AP600 PRA and design alternatives suggested by AP600 design personnel.

Appendix 1B of the AP600 standard safety analysis report (SSAR) does not explicitly state whether Westinghouse's evaluation included plant improvements considered as part of the NRC's Containment Performance Improvement (CPI) program (NUREG/CR-5562, -5567, -5575, and -5630). However, Westinghouse stated that the types of design changes identified in the CPI program had been incorporated into the design or considered as design alternatives. The improvements identified in the CPI program were also evaluated in other documents reviewed by Westinghouse, including the System 80+ design alternative evaluations.

Westinghouse eliminated certain SAMDAs from further consideration on the basis that they are already incorporated in the AP600 design. Such features include the following:

- hydrogen ignition system
- reactor cavity flooding system
- reactor coolant pump seal cooling (AP600 has canned motor pumps)
- reactor coolant system depressurization
- external reactor vessel cooling
- nonsafety-grade containment sprays

On the basis of the screening, Westinghouse retained 14 potential SAMDAs for further consideration. These SAMDAs, described in Section 1B.7 of the SSAR, are summarized below:

(1) Upgrade the Chemical and Volume Control System (CVCS) for Small Loss-of-Coolant Accidents (LOCAs): The CVCS is currently capable of maintaining the reactor cooling system (RCS) inventory for LOCAs with effective break sizes up to 0.97 cm (3/8 in.) in diameter. This design alternative would extend the capability of the CVCS so that it could maintain the RCS inventory during small and intermediate LOCAs up to an effective break size of 15.2 cm (6 in.) in diameter. Implementation of this design alternative would require installation of in-containment refueling water storage tank (IRWST) / containment recirculation connections to the CVCS, as well as the addition of a second line from the CVCS pumps to the RCS. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 5.5E-04 person-rem/yr.

(2) Filtered Containment Vent: This design alternative would involve the installation of a filtered containment vent, including all associated piping and penetrations. This modification would provide a means to vent the containment to prevent catastrophic overpressure failures, as well as a filtering capability for source term release. The filtered vent would reduce the risk associated with late containment failures that might occur after failure of the passive containment cooling system (PCS). However, even if the PCS fails, air cooling would be expected to limit the containment pressure to less than the ultimate pressure. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 1.0E-03 person-rem/reactor-year.

(3) Self-Actuating Containment Isolation Valves: Self-actuating containment isolation valves could increase the likelihood of successful containment isolation during a severe accident. This design alternative would involve adding a self-actuating valve or enhancing the existing containment isolation valves on containment penetrations that are normally-open. (Specifically, penetrations that provide normally open pathways to

the environment during power and normal shutdown conditions.) This would permit automatic self-actuation in the event that containment conditions indicate a severe accident. Closed systems inside and outside containment, such as residual heat removal system and component cooling, would be excluded from this design alternative. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 7.4E-04 person-rem/yr.

(4) Passive Containment Sprays: Installing a passive safety-related containment spray system could result in the following risk benefits:

(a) Scrub fission products, primarily for containment isolation failure.
(b) Provide an alternative means to flood the reactor vessel (in-vessel retention).
(c) Control containment pressure for cases in which the PCS has failed.

Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 6.9E-03 person-rem/yr, which would represent eliminating all release categories except containment bypass.

(5) Active High-Pressure Safety Injection System: Adding a safety-related, active high-pressure safety injection system would enable the reactor to prevent a core melt in all events except excessive LOCAs and anticipated transients without scram. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 6.1E-03 person-rem/yr.

(6) Steam Generator Shell-Side Heat Removal System: This design alternative would involve installing a passive safety-related heat removal system to the secondary side of the steam generators. This enhancement would provide closed loop secondary system cooling via the use of natural circulation and stored water cooling, thereby preventing loss of the primary heat sink given loss of startup feedwater and the passive residual heat removal heat exchanger. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 5.3E-04 person-rem/yr.

(7) Direct Steam Generator Safety and Relief Valve Flow to the IRWST: Flow from the steam generator safety and relief valves could be directed to the IRWST to prevent or reduce fission product release from bypassing containment during an steam generator tube rupture (SGTR) event. An alternative, lower cost approach to this design alternative would be to redirect the flow only from the first stage safety valve to the IRWST. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 4.2E-04 person-rem/yr.

(8) Increased Steam Generator Pressure Capability: In lieu of design alternative (7) above, fission product release bypassing containment could be prevented or reduced by increasing the steam generator secondary side and safety valve set point to a level high enough so that an SGTR will not cause the secondary system safety valve to open. Although detailed analyses have not been performed, it is estimated that the secondary-side design pressure would have to be increased by several hundred psi to make this alternative effective. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 4.2E-04 person-rem/yr.

(9) Secondary Containment Filtered Ventilation: This design alternative would involve installing a passive charcoal and high-efficiency particulate air filter system for the middle and lower annulus region of the secondary concrete containment (below Elevation 135'-3"). Drawing a partial vacuum on the middle annulus via an eductor with motive power from compressed gas tanks would operate the filter system. This design alternative would reduce particulate fission product release from any failed containment penetrations. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 7.4E-04 person-rem/yr.

(10) Diverse IRWST Injection Valves: In the current design, a squib valve in series with a check valve isolates each of the four IRWST injection paths. To provide diversity, the design could be modified so that a different vendor provides the valves in two of the lines. This enhancement would reduce the likelihood of common cause failures of the four IRWST injection paths. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 5.3E-03 person-rem/reactor-year, which would represent eliminating all core damage sequences resulting from a failure of IRWST injection (3BE sequences).

(11) Diverse Containment Recirculation Valves: In the current design, a squib valve isolates each of the four containment recirculation paths. In two of the four paths, each of the squib valves is in series with a check valve. In the remaining two paths, each squib valve is in series with a motor-operated valve (MOV). To provide diversity, the design could be modified so that a different vendor provides the squib valves in two lines. This enhancement would reduce the likelihood of common cause failures of the four containment recirculation paths. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 1.5E-04 person-rem/reactor-year, which would represent eliminating all core damage sequences resulting from a failure of containment recirculation (3BL sequences).

(12) Ex-Vessel Core Catcher: This design alternative would inhibit core-concrete interaction (CCI), even in cases where the debris bed dries out. The enhancement would involve designing of a structure in the containment cavity or using a special concrete or coating. The current AP600 design incorporates a wet cavity design in which ex-vessel cooling is used to maintain core debris within the vessel. In cases where reactor vessel flooding has failed, the PRA assumes that containment failure occurs from an ex-vessel steam explosion or CCI. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 6.1E-03 person-rem/reactor-year.

(13) High Pressure Containment Design: The proposed high-pressure containment design would have a design pressure of approximately 300 psi, and would include a passive cooling feature similar to the existing containment design. This design would reduce the likelihood containment failures from severe accident phenomena such as steam explosions and hydrogen detonation. However, this alternative would not reduce the frequency or magnitude of releases from an unisolated containment. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 6.1E-03 person-rem/reactor-year.

(14) Increased Reliability of the Diverse Actuation System (DAS): This design alternative would involve improving the reliability of the DAS. The DAS is a non-safety system that can automatically trip the reactor and turbine and actuate certain engineered safety features (ESF) equipment if the protection and safety monitoring system is unable to perform these functions. In addition, the DAS provides diverse monitoring of selected plant parameters to guide manual operation and confirm reactor trip and ESF actuations. Westinghouse estimated that implementing this design alternative would reduce plant risk by at most 2.2E-04 person-rem/reactor-year.

3.3 Evaluation

The NRC reviewed the set of potential SAMDAs identified by Westinghouse and found it to be reasonably complete. The activity was accomplished by reviewing design alternatives associated with the following plants: Limerick (NUREG-0974), Comanche Peak (NUREG-0775), System 80+ design (NUREG-1462), Watts Bar (NUREG-0498), and the U.S. ABWR design (NUREG-1503). Also surveyed were accident management strategies (NUREG/CR-5474), and alternatives identified through the Containment Performance Improvement (CPI) Program (NUREG/CR-5567, -5575, -5630, and -5562). In addition, the review approach is consistent with the guidance outlined in the Environmental Standard Review Plan (NUREG-1555).

The results of the NRC's assessment are summarized in Appendix A to the "Review of Severe Accident Mitigation Design Alternatives (SAMDAs) for the Westinghouse AP600 Design" (SEA 97-2708-010-A;1). The appendix briefly summarizes each of the design alternatives identified in the foregoing references. In addition, the Westinghouse AP600 design alternatives, which are discussed in Appendix 1B of the SSAR, are included. In all, the staff reviewed more than 120 possible design alternatives including most improvements identified as part of the NRC's CPI program. Specific improvements considered applicable to the AP600 included a filtered containment vent and a flooded rubble bed core-retention device, two improvements specifically mentioned in NUREG-0660 for evaluation as part of Three Mile Island (TMI) Item II.B.8. The list of 120 also included potential SAMDAs oriented toward reducing the risk from major contributors to risk for AP600, including SGTR events.

Although the Westinghouse analysis did not consider several design alternatives, in most instances the excluded alternatives are either already included in the design or bounded in terms of risk reduction by one or more of the design alternatives that were included in the Westinghouse analysis. In other cases, the design alternatives were pertinent only to boiling water reactors. The NRC's preliminary review did not reveal any additional design alternatives that obviously should have been considered by Westinghouse. Also, Westinghouse considered some potential design alternatives as accident management strategies rather than design alternatives.

The NRC noted that the set of SAMDAs reviewed by Westinghouse is not all inclusive, in that additional SAMDAs could be postulated. However, potential benefits accrued by any additional modifications would not likely exceed those for the modifications evaluated, and the costs of such alternative improvements are not expected to be less than those of the least expensive improvements evaluated, when the subsidiary costs associated with maintenance, procedures, and training are considered.

The discussions in Appendix 1B of the SSAR do not specify the basis or the process that Westinghouse used to screen the many possible design alternatives to arrive at the final list of 14 selected for further evaluation. Similarly, Westinghouse's responses to the staff's request for additional information (RAI) provided few additional insights into the process. Nevertheless, as noted above, the staff's independent review of the more than 120 candidate designs did not identify any new alternatives likely to be more cost-beneficial than those included in Westinghouse's evaluation of AP600 design alternatives.

3.4 Risk Reduction Potential of SAMDAs

3.4.1 Westinghouse Evaluation

In its evaluation, Westinghouse assumed that each design alternative would work perfectly to completely eliminate the respective accident sequence(s). This assumption is conservative from a cost-benefit perspective, as it maximizes the benefit of each design alternative, which is measured on the basis of risk reduction (i.e., the risk reduction assigned to passive containment sprays assumes that all release categories except containment bypass are eliminated). In each case, Westinghouse used analytical models and results contained in the AP600 PRA to estimate the risk reduction for each design alternative. Westinghouse then expressed the risk reduction in terms of whole body person-rem per year received by the total population within a radius of 80.5 km (50 miles) of the AP600 plant site. Each of the 14 design alternatives was evaluated separately.

Table 1 of this EA summarizes Westinghouse's risk reduction estimates by comparing the benefits of averting offsite exposure using each potential design alternative. The bases for these estimates are provided in section 1.B.7 of the AP600 SSAR.

3.4.2 Evaluation

The NRC reviewed Westinghouse's bases for estimating the risk reduction associated with the various SAMDAs, and concluded that Westinghouse used a reasonable, and generally conservative set of assumptions as the bases for the risk reduction estimates regarding each design alternative. The level of risk reduction estimated for the various SAMDAs is driven by two underlying assumptions of the methodology. Specifically, Westinghouse's risk reduction estimates reflect only the contribution from internal events initiated at power, and they are point estimate (mean) values without consideration of uncertainties in core damage frequency (CDF) or offsite consequences. Although this is consistent with the approach followed in previous design alternative evaluations, further consideration of these factors could lead to significantly higher risk reduction values, given the extremely small CDF and risk estimates in the baseline PRA for internal events.

In assessing the risk reduction potential of SAMDAs for the AP600 design, the NRC staff considered Westinghouse's risk reduction estimates for the various alternatives, in conjunction with supplementary parametric analyses to evaluate the potential impact of external events and uncertainties. These analyses are further discussed in Section 3.7 of this EA.

Table 1 Comparison of Estimated Benefits from Averted Offsite Exposure

Severe Accident Mitigation Design Alternative	Estimated Capital Cost, $	Averted Risk, person-rem per year	Westing ... Benefits*, $	Staff Benefits** @ $2000/ person-rem, 1996$	Staff Benefits** @ $5000/ person-rem, 1996$
Upgrade Chemical and Volume Control System for	1,500,000.00	0.00055	4	17	39
Filtered Containment Vent	5,000,000.00	0.00100	6	30	70
Self-Actuating Containment Isolation Valves	33,000.00	0.00074	5	22	52
Passive Safety Grade In-Containment Sprays	3,900,000.00	0.00690	44	207	484
Active High-Pressure Safety Injection System	20,000,000.00	0.00610	39	183	428
Steam Generator Shell-Side Passive Heat Removal	1,300,000.00	0.00053	3	16	37
Direct Steam Generator Safety and Relief Valve	620,000.00	0.00042	3	13	29
Increased Steam Generator Pressure Capability	8,200,000.00	0.00042	3	13	29
Secondary Containment Filtered Ventilation	2,200,000.00	0.00074	5	22	52
Diverse IRWST Injection Valves	570,000.00	0.00530	34	159	372
Diverse Containment Recirculation Valves	150,000.00	0.00015	1	5	11
Ex-Vessel Core Catcher	1,660,000.00	0.00610	39	183	428
High Pressure Containment Design	50,000,000.00	0.00610	39	183	428
Increase Reliability of Diverse Actuation System	470,000.00	0.00022	2	7	15
100% Effective Design Alternative***		0.00734	47	221	551

* Benefits account only for offsite effects, 15.7% effective discount rate, 30-yr plant life, $1000/person-rem

** Benefits account only for offsite effects, 7% effective discount rate, 60-yr plant life

*** See discussion in Section 3.6.2

3.5 Cost Impacts of Candidate SAMDAs

3.5.1 Westinghouse Evaluation

Sections 1B.4.2, 1B.4.3, and 1B.8 of the SSAR discuss the capital cost estimates for the AP600 design alternatives evaluated by Westinghouse, and Table 1B.8-1 of the SSAR presents the results of the cost evaluations. Specifically, for each design alternative, Table 1B.8-1 lists the potential risk reduction, the capital benefit (assuming the design alternative was highly effective in reducing accident risks), the capital cost, and the net capital benefit. Notably, Westinghouse's cost evaluations did not account for factors such as design engineering, testing, and maintenance associated with each design alternative. If included, these factors would increase the overall costs and decrease the capital benefits of each alternative. Thus, this approach is conservative.

3.5.2 NRC Evaluation

To gauge the reasonableness of the cost estimates that Westinghouse presented in the SSAR, the NRC compared the capital costs for the AP600 design alternatives with those evaluated for the ABWR and System 80+ designs. There is not an exact match in the design alternatives among the reactor designs, therefore, only broad comparisons are possible.

For example, the AP600 active high-pressure safety injection system, which is estimated to cost $20 million, adds an active high-pressure safety injection pump and associated piping, valves, and supports, thus adding an entirely new safety-related system to the AP600 design. This alternative can be compared to the alternative high-pressure safety injection for the System 80+ design, which is estimated to cost $2.2 million. However, the design alternative for the System 80+ design simply adds parallel piping and valves to an existing system, which would be expected to cost only a fraction of the total system price.

Similarly, the filtered containment vent for AP600 can be compared to systems with similar functions for the ABWR and System 80+ designs. The AP600 design included a filtered containment vent and all associated piping and penetrations. The ABWR design added an ex-containment filter system to an existing venting system. The System 80+ design included a filtered containment vent similar to the multi-venturi scrubbing systems implemented in some European plants. The estimated costs for the three venting systems ($5 million, $3 million, and $10 million, respectively) reasonably agree with each other given the differences in the designs.

The costs for the containment spray (non-safety-related system) for the AP600 design, which was evaluated in an earlier version of the SSAR (Section 1B) before it was incorporated into the AP600 design, can be compared to the reactor building sprays for the ABWR design and the alternative containment spray for System 80+ design. This AP600 design alternative involves adding piping and spray headers inside containment, and connects to an existing fire water system. Similarly, for the ABWR, the existing in-containment fire spray system would be modified to provide sprays in areas vulnerable to fission product release. The ABWR design modification would thus be limited to providing sprays only to selected areas of containment. For the System 80+ design, this alternative involves adding piping to connect to the existing in-containment spray system, together with new pumps to supply the water. Estimated costs for these three spray systems were $415,000 for the AP600 design, $100,000 for the ABWR design, and $1.5 million for the System 80+ design. In light of the scope differences among these design alternatives, the estimates for the AP600 spray system appear to be reasonable.

The above cost comparisons illustrate that the cost estimates for several of the AP600 design alternatives reasonably agree with the costs for roughly similar design alternatives evaluated for the ABWR and the System 80+ designs.

The NRC developed independent cost estimates for one particular design alternative, the active, non-safety-related containment spray system, to further assess the reasonableness of the AP600 design alternative cost estimates (This analysis was performed before the system was incorporated into the AP600 design and deleted from SSAR Section 1B). The assessment assumed the addition of fire protection system grade spray headers and supply piping inside containment (carbon steel), and the addition of control valves and piping outside containment and connected to the existing fire water supply system. The resulting costs for the containment spray system ranged from about $300,000 to $350,000 (1996 dollars), depending on the assumptions made regarding the required pipe size. These independent estimates did not include design engineering; first-of-a-kind costs; or allowances for associated personnel training, procedure development, or recurring operations and maintenance costs. This approach is similar to that used in Westinghouse's cost estimates. Thus, Westinghouse's estimate of $415,000 for this design alternative reasonably agrees with the independent estimate. In addition, the NRC developed an independent cost estimate for a containment spray system similar to that described above, but with increased pumping capacity. (The increased pumping capacity is needed because Westinghouse's letter of March 13, 1997, indicated that the currently designed fire water supply system is capable of delivering less than 1.89 kL/min (500 gpm) to the proposed containment spray system.) The system evaluated for this alternative would increase the fire water pump capacity so that each pump would be capable of delivering 11.36 kL/min (3000 gpm) to the containment sprays against a containment pressure of 310.3 kPa (30 psig). The piping used to supply fire water to the containment in the current design would be increased in size to reduce the flow resistance. This modification to the AP600 design was estimated to cost about $370,000 (1996 dollars). As with the foregoing estimate, no allowance was made for personnel training, procedure development, or recurring operations and maintenance costs.

On the basis of this audit, the NRC viewed Westinghouse's approximate cost estimates as adequate, given the uncertainties surrounding the underlying cost estimates, and the level of precision necessary given the greater uncertainty inherent on the benefit side.

3.6 Cost-Benefit Comparison

3.6.1 Westinghouse Evaluation

After considering the risk reduction potential and cost impact of the various SAMDAs, Westinghouse performed a cost-benefit comparison to determine whether any of the potential severe-accident mitigation design features would be justified. To do so, Westinghouse assessed the benefits of each design alternative in terms of potential risk reduction, which was defined as the reduction in whole-body person-rem per year received by the total population within a radius of 80.5 km (50 miles) of the AP600 plant site. Westinghouse then assigned a value of $1000 to each person-rem of averted offsite exposure, which was assumed to account for both health effects and offsite property damage. This value was treated as the annual levelized benefit for averted risk.

Westinghouse divided the annual levelized benefit by the annual levelized fixed charge rate to determine the maximum expenditure justified by a given reduction in risk ("maximum capital benefit"). The annual levelized fixed charge rate was determined to be 15.7 percent in current U.S. dollars on the basis of factors and methods provided in documents developed by the Electric Power Research Institute (EPRI P-6587-L) and the U.S. Department of Energy (DOE/NE-0095). Westinghouse calculated the fixed charge rate using a component "book life" of 30 years. The use of a high charge rate tends to minimize the capital benefit associated with each design alternative. Nevertheless, the 30-year life used in the calculations makes little difference in the economic benefit compared to the more typical 60-year life, particularly when the high levelized annual fixed charge rate of 15.7 percent is used.

The Westinghouse approach for calculating the benefits or reduced risk from each individual design alternative also does not give credit for averted onsite property damage and replacement energy costs which are realized through a reduction in accident frequency. The onsite property damage and replacement energy costs may have been neglected because the estimated CDF is very low. However, as indicated below, these onsite considerations can substantially add to the benefits that may be achieved using design alternatives.

Table 1 of this EA reports Westinghouse's cost-benefit estimates for each potential SAMDA using a screening criterion of $1000/person-rem-averted to identify whether any of the SAMDAs could be cost effective. As shown in Table 1, the highest capital benefit calculated by Westinghouse for any design alternative is about $50, while the capital cost for the least expensive design alternative is $33,000. On this basis, Westinghouse concluded that no additional modifications to the AP600 design are warranted.

3.6.2 NRC Evaluation

The NRC updated its recommended approach for the monetary conversion of radiation exposures. Previous guidance specified that 1 person-rem of exposure should be valued at $1000. This conversion factor for offsite doses was intended to account for both health effects and offsite property damage, and exposures incurred in future years were not to be discounted. The guidance given in the NRC's regulatory analysis guidelines (NUREG/BR-0058, Revision 2), recommends using $2000 per person-rem of exposure as the monetary conversion factor. In addition, future exposures are to be discounted to arrive at their present worth to assess values and impacts. Offsite property damage from nuclear accidents is to be valued separately, and is not part of the $2000 per person-rem value.

Evaluations recently performed by Brookhaven National Laboratory for the NRC assessed total costs associated with offsite releases, including both health effects and property damage/loss effects (NUREG/CR-6349). Costs were assessed for each of the five NUREG-1150 plants (Grand Gulf, Peach Bottom, Sequoyah, Surry and Zion). The results indicated that overall costs associated with offsite releases of radioactive materials, presented on a cost per person-rem of offsite exposure, ranged from about $2000 to more than $5000 per person-rem, depending on factors such as the assumed interdiction criteria. A criterion of $3000 per person-rem averted was added to account for offsite property damage and other related costs for severe accidents. Thus, Westinghouse's cost-benefit evaluation approach used for the design alternatives is not consistent with the approach recommended in NUREG/BR-0058, Revision 2. The key differences are summarized in Table 2 of this EA, and the NRC staff's independent evaluation is found below.

Table 2 Key Differences Between the Westinghouse Approach and NUREG/BR-0058

Westinghouse's SAMDA Approach	NUREG/BR-0058 Recommended Approach
$1000 per person-rem averted (for valuing risk reduction)	$2000 per person-rem averted to account for health effects, plus $3000 per person-rem averted to account for other offsite effects and related costs
15.7% discount rate	7% discount rate
No accounting for benefits of averted onsite cleanup and decontamination costs	Consideration given for benefits of averted onsite cleanup and decontamination costs
No accounting for benefits of averted replacement energy costs	Consideration given for benefits of averted replacement energy costs

The NRC staff applied the recommended approach in NUREG/BR-0058, Revision 2 to the design alternatives identified for the AP600 design to arrive at a baseline potential benefit from the reduction in offsite risk. This EA used a discount rate of 7% and assumed a reactor life of 60 years. The averted risk for each design alternative was taken from Table 1B.8-1 of the SSAR. In addition, the NRC staff used two monetary conversion factors for offsite exposures. The first is the $2000/person-rem recommended in NUREG/BR-0058, Revision 2. The second, $5000/person-rem, is intended to account for offsite property damage and health effects. The results for each design alternative are shown in columns 5 and 6 of Table 1. For comparison purposes, Westinghouse's estimates of the capital cost, averted risk, and capital benefit for each design alternative are also presented (columns 2, 3, and 4 of Table 1). A 100% effective design alternative would reduce the CDF and/or offsite releases to zero. Estimated benefits from a 100% effective design alternative are also shown for each of the alternative cost bases (last row of Table 1).

The results shown in Table 1 indicate that the benefits calculated using a 7% discount rate, a 60-year plant life, and a $2000/person-rem conversion factor is about a factor of four higher than those calculated by Westinghouse. The benefits calculated using $5000/person-rem are about a factor of 10 higher than those estimated by Westinghouse. The highest capital benefit shown in Table 1 amounts to less than $500, while the capital cost for the least expensive design alternative is $33,000. Thus, even with the highest benefit basis ($5000/person-rem, 7% discount rate, 60-year life), the calculated benefits are almost two orders of magnitude too small to justify the addition of any of the design alternatives listed. It should be noted, however, that this assessment neglected the benefits from averted onsite costs, which are relevant for design alternatives that reduce core damage frequency. Dollar savings derived from averted onsite costs are treated as an offset or reduction in the capital cost of the design alternative in the NRC staff's analysis. Averted onsite costs are significant for certain design alternatives and are considered below.

3.7 Further Considerations

The estimates of potential design alternative benefits listed in Table 1 of this EA reflect Westinghouse's estimates of averted risk and neglect the benefits from averted onsite costs. As mentioned in Section 3.2 of this EA, Westinghouse's risk estimates do not account for uncertainties either in the CDF or in the offsite exposures resulting from a core damage event. The uncertainties in both of these key elements are fairly large because key safety features of the AP600 design are unique, and the reliability of the safety features has been evaluated through analysis and testing rather than operating experience. In addition, the estimates of CDF and offsite exposures do not account for the added risk from external events, i.e. earthquakes.

The NRC staff screened the candidate SAMDAs to determine whether any of the design alternatives could be cost-beneficial when the cost-benefit analysis incorporates uncertainties, added risk from external events, and averted onsite costs. The staff performed a more detailed assessment for those design alternatives having potentially favorable cost-benefit factors under these more limiting considerations. These analyses are discussed in Sections 3.7.1 – 3.7.3.

3.7.1 Uncertainties in Core Damage Frequency and Accident-Related Exposures

Revision 8 to the PRA discussed the uncertainty in the estimated CDF for the AP600 design. Specifically, the CDF uncertainty distribution was characterized by an error factor (EF) of about 5.7. Assuming a log normal distribution, the EF is the ratio of the 95th percentile to the median, and also the ratio of the median to the 5th percentile. Thus, the CDF for internal events could be a factor of six higher or lower than assumed in the analysis discussed above.

Additional factors that could substantially increase the estimated CDF for the AP600 design include the contributions from events and accident sequences that have not yet been identified, as well as identified sequences that have not yet been analyzed in the PRA. Examples of the latter include external events such as fires and earthquakes. Notably, the CDF base estimate of 1.7E-07/reactor-year does not include the contribution of external events. In the PRA, Westinghouse indicated that external events, in particular internal fires, are estimated to increase the CDF by about a factor of four. However, the PRA available for this study did not define the potential contributions from seismic events, which could readily increase the CDF by an order of magnitude or more. These external events can also degrade the containment performance, so that the releases from containment may also be higher than for accidents associated with internal events.

The potential increases in CDF attributed to accident sequences that have not yet been identified is difficult to estimate. The contributions from such sequences should be small if Westinghouse performed the PRA in a thorough and systematic manner. For the purposes of the present analysis, the effects of these sequences are assumed to be enveloped by the potential increase in CDF attributed to external events.

Section 1B.6 of the SSAR presented Westinghouse's estimates of offsite exposures for the major release categories (RCs) defined for the AP600 design. On the basis of the CDF reference value of 1.7E-07/reactor-year and the total risk of 7.3E-03 person-rem/reactor-year, Westinghouse estimated that the "average" offsite exposure is of the order of 50,000 person-rem per core damage event. However, Westinghouse's documentation did not indicate the uncertainty in the estimated releases. The average offsite exposure of 50,000 person-rem per AP600 core damage event estimated by Westinghouse is a factor of 2.7 lower than the average offsite exposures calculated for the five current-generation nuclear plants addressed in NUREG-1150 (after scaling the NUREG-1150 plant releases to that of a 600-MWe plant). The improved performance may be attributed, in part, to methods and assumptions for defining source terms, as well as the high likelihood of successful RCS depressurization and in-vessel retention of damaged fuel in the AP600 design.

Uncertainties in the offsite exposure estimates for the AP600 design are significant. As described in Section 19.1.3.3.3 of the AP600 Final Safety Evaluation Report (FSER), the AP600 risk profile is shaped by the following major assumptions regarding containment failure modes and release characteristics:

- conservative assumptions regarding early containment failure from ex-vessel phenomena
- optimistic assumptions that external reactor vessel cooling will always prevent reactor pressure vessel (RPV) breach
- substantial credit for additional aerosol removal in SGTR events

If early containment failure is avoided (as suggested by deterministic calculations performed subsequent to the PRA), and reactor pressure vessel breach instead results in a more benign release (e.g., a containment failure in the intermediate time frame), overall risk for internal events would be reduced by about a factor of two. By contrast, if credit for external reactor vessel cooling (ERVC) is reduced or eliminated, containment failure frequency would increase proportionally, since all RPV breaches are assumed to lead to early containment failure in the baseline PRA. Under the most limiting assumption that ERVC always fails and leads to early containment failure, the containment failure frequency would approach the core melt frequency and risk would increase by a factor of 20 (to about 0.16 person-rem/yr). Similarly, offsite risk can be significantly impacted if the design fails to realize the decontamination factor (DF) of 100 applied to aerosol release fractions for SGTR events predicted by the modular accident analysis program (MAAP) to account for fission product removal by impaction on steam generator tubes. With this credit for aerosol removal, the risk contribution from a containment bypass is minimal (6% of the total). Without this credit, overall risk for internal events would increase by a factor of seven and would be dominated by containment bypass releases. Finally, the PRA did not credit the impact of the non-safety-related containment spray system on fission product releases. Containment sprays could significantly reduce the estimated risk in the baseline PRA (by perhaps a factor of 2), since the sprays would be effective in reducing the source terms in the risk-dominant RCs such as early containment failure (CFE) and containment isolation failure. However, sprays would not impact releases attributed to SGTR events or containment bypass.

In summary, the actual offsite exposure could range from a factor of two lower to an order of magnitude higher than the Westinghouse estimate, given the uncertainties in the underlying analyses of containment performance. This uncertainty range was factored into the NRC staff's reassessment discussed below.

3.7.2 Reassessment of Design Alternative Cost-Benefit Relationships in Light of Uncertainties

The NRC staff performed independent analyses to reassess the benefits of potential AP600 design alternatives by incorporating uncertainties in the estimated CDF, offsite releases of radioactive material given a severe accident, and effects of external events. The NRC staff estimated the *maximum* benefits that can be achieved with AP600 design alternatives, assuming that an alternative can either completely eliminate all core damage events or completely eliminate offsite releases of radioactive materials if a severe accident does occur. The staff used the FORECAST code (NUREG/CR-5595, Revision 1, "FORECAST: Regulatory Effects Cost Analysis Software Manual, Version 4.1," Science and Engineering Associates, Inc., July 1996) to calculate estimated benefits. FORECAST allows the use of uncertainty ranges for all key parameters and provides a means to combine uncertainties in these parameters. It also provides a distribution for the bottom line costs or benefits, and thus presents a picture of the uncertainty in the "bottom line" figures. Table 3 of this EA presents the key parameters used in evaluating the maximum potential benefit.

For the purposes of estimating the maximum potential benefit from design alternatives, the NRC staff assumed that external events and accident sequences not previously considered in the PRA would increase the reference CDF by two orders of magnitude, (i.e., a factor of 100), with an EF of six used for this higher CDF. The staff then evaluated cases assuming the reference value of 50,000 person-rem per accident. Table 4 of this EA presents the results of this analysis.

The entries in Table 4 indicate that design alternatives which prevent accidents (reduce the accident frequency to zero) are much more cost-effective than design alternatives which reduce or eliminate offsite releases but have no effect on accident frequency. This is because of the fairly large benefits associated with averted onsite cleanup and decontamination costs, and avoided replacement energy costs, neither of which are assumed to be impacted by design alternatives which do not reduce accident frequency.

Table 3 Key Parameters Used by FORECAST in Evaluating Maximum SAMDA Benefits

Parameter	Value
Reference AP600 core damage frequency	1.7E-7/reactor-year (EF=5.7)
Average public radiation exposure per accident:	43,200 person-rem (rounded to 50,000) (assumed error factor: 5)
Plant lifetime	60 years
Discount rate	7%
Conversion factor[1]	$5000/person-rem
Replacement energy costs	$277,000/day of downtime
Averted cleanup and decontamination costs[2]	$1,690,000,000 /major accident
Averted replacement energy costs[3]	$20,200,000,000/major accident

[1] Based on NUREG/CR-6349; accounts for both offsite health effects and offsite property damage effects

[2] Based on guidance provided in NUREG/BR-0184 (not adjusted for AP600-specific features)

[3] Based on average replacement energy costs for pressurized water reactors in the 500 – 1000 MWe range

Table 4 SAMDA Benefits (1996$) Accounting for Uncertainties and External Events Effects

Case No.	Description	5% Confidence Level	Mean	95% Confidence Level
1	Base CDF (1.7E-07/yr) and reference offsite release (50,000 person-rem); design alternatives which reduce the accident frequency to zero	$1100	$8000	$26,600
2	Base CDF increased by factor of 100 to account for external events and other accident sequences not yet accounted for; other factors same as Case 1; design alternatives which reduce the accident frequency to zero	$90,500	$647,000	$2,257,000
3	Base CDF increased by factor of 100 to account for external events; other factors same as Case 1; design alternatives which reduce the offsite releases to zero, but do not change the accident frequency	$1700	$49,000	$223,000

Case 1 is the reference case utilizing the base CDF and Westinghouse-estimated offsite exposures. In this case, the estimated benefits are considerably higher than those cited in Table 1 of this EA, primarily because the benefits include averted onsite cleanup and decontamination costs, as well as averted replacement energy costs.

Cases 2 and 3 illustrate the effects of the higher CDF associated with external events, but do not include the effects of possible higher releases from containment attributed to such events (In other words, these cases retain the base offsite exposure of 50,000 person-rem/event). These cases may be used as the baseline benefits that include external events and assume that external events would not impact containment performance. Case 2 shows the potential benefit range for a design alternative that could reduce the accident frequency to zero. Case 3 applies to a design alternative that would eliminate all offsite releases, but would not impact the CDF.

Table 5 of this EA combines the information in Tables 1 and 4 to estimate the total benefit possible from specific design alternatives. The design alternatives are divided between those that impact the CDF and those that impact containment performance but not the CDF. Benefits have been estimated by taking the fractional reduction in risk for each design alternative (compared to the AP600 baseline risk as defined by Westinghouse) and applying that fraction to the mean benefits illustrated in Table 4. Design alternatives that reduce the CDF were applied to the Case 2 mean benefit, while those that only effect containment performance were applied to the Case 3 mean benefit. The values shown in Columns 4 through 7 of Table 5 reflect benefits calculated using the mean values. By contrast, values shown in Columns 8 through 11 were calculated using the 95^{th}-percentile values (In other words, there is only a 5% chance that the actual benefits will be greater than the values shown in Columns 8-11).

The use of the maximum benefits typically improves the cost-benefit ratio by a factor of approximately five, but does not alter any of the overall conclusions about design alternatives that have acceptable cost-benefit ratios.

3.7.3 Further Evaluation of Design Alternatives With Potentially Favorable Cost-Benefit Factors

Design alternatives that are within a decade of meeting the cost-benefit criteria of $5000/person-rem were subjected to further probabilistic and deterministic considerations, including a qualitative assessment of the following:

- the impact of additional benefits that could accrue for the design alternative if it would be effective in reducing risk from certain external events, as well as internal events
- the effects of improvements already made at the plant
- any operational disadvantage associated with the potential design alternative

None of the design alternatives have a cost-benefit ratio of less than $5000/person-rem. However, the only design alternatives that are within a decade of the $5000/person-rem standard are the diverse IRWST valves at $19,800/person-rem and the self-actuating containment isolation valves at $33,700/person-rem, as described in the following sections.

3.7.3.1 Diverse IRWST Injection Valves

In the current AP600 design, a squib valve in series with a check valve isolates each of four IRWST injection paths. This design alternative would reduce the likelihood of common cause failures of IRWST injection to the reactor by utilizing diverse valves in two of the four paths. If it functioned perfectly, this design alternative could potentially reduce the CDF by about 72%. When taking into account external events, other accident sequences not yet included in the AP600 PRA, and other uncertainties, this design alternative is estimated to be highly cost-effective. In the absence of a comprehensive external events PRA for the AP600 plant, it is difficult to estimate the effectiveness of this design alternative in reducing the risk from such events. However, it appears likely that failure to inject coolant to the reactor would remain a prominent contributor to the CDF from external events, in which case, diversity in the IRWST injection valves should help to reduce the risk from both external and internal events.

Alternative vendors are available for the check valves. However, it is questionable whether check valves from different vendors would be sufficiently different to be considered diverse unless the type of check valve was changed from the current swing disk check valve to another type. The swing disk type is preferred for this application, and other types are considered less reliable.

Adding diversity to the injection line squib valves would require additional spares at the plant, and additional training for plant operations and maintenance staff, but would not appear to add significantly to the operational aspects of the AP600. However, a greater issue concerns the availability and costs of acquiring diverse valves from a second vendor. Squib valves are specialized valve designs for which there are few vendors. Westinghouse claims that a vendor may not be willing to design, qualify, and build a squib valve for this AP600 application considering that they would only supply two valves per plant. The cost estimate for this design

Table 5 Estimated Maximum Benefit from Individual SAMDAs

Design Alternative	Fractional Risk	Capital Cost	Mean Benefit from Reduced Risk[1]	Mean Benefit from Averted Onsite Costs[2]	Adjusted Capital Costs Reduced by Mean Averted Onsite Costs[3]	$/person-rem based on Mean Benefits[4]
Alternatives that Reduce Core Damage Frequency						
Upgrade Chemical and Volume Control System for Small LOCA <4"	0.075	$1,500,000	$3675	$44,850	$1,455,150	$1,979,796
Active High-Pressure Safety Injection System	0.830	$20,000,000	$40,670	$496,340	$19,503,660	$2,397,794
Steam Generator Shell-Side Heat Removal	0.070	$1,300,000	$3430	$41,860	$1,258,140	$1,834,023
Diverse IRWST Injection Valves	0.720	$570,000	$35,280	$430,560	$139,440	$19,762
Diverse Containment Recirculation Valves	0.020	$150,000	$980	$11,960	$138,040	$704,286
Increased Reliability of Diverse Actuation System	0.030	$470,000	$1470	$17,940	$452,060	$1,537,619
Alternatives that Reduce Offsite Releases but do not Impact Core Damage Frequency						
Filtered Containment Vent	0.136	$5,000,000	$6664		$5,000,000	$3,751,501
Self-Actuating Containment Isolation Valves	0.100	$33,000	$4900		$33,000	$33,673
Passive Containment Sprays	0.940	$3,900,000	$46,060		$3,900,000	$ 423,361
Direct Steam Generator Safety and Relief Valve Flow to the IRWST	0.057	$620,000	$2793		$620,000	$1,109,918
Increased Steam Generator Pressure Capability	0.057	$8,200,000	$2793		$8,200,000	$14,679,500
Secondary Containment filtered Ventilation	0.100	$2,200,000	$4900		$2,200,000	$2,244,898
Ex-Vessel Core Catcher	0.830	$1,660,000	$40,670		$1,660,000	$204,082
High-Pressure Containment Design	0.830	$50,000,000	$40,670		$50,000,000	$6,147,037

1- Benefit because of reduced offsite exposures. For design alternatives that reduce CDF, this value also includes any benefits from reduced occupational exposures from averted onsite cleanup and decontamination efforts.
2- Benefits from averted onsite costs (i.e., averted cleanup and decontamination costs and averted replacement energy costs.)
3- The benefits from averted onsite costs are used to effectively reduce the capital cost of each design alternative.
4- The cost-benefit ratio for each AP600 design alternative evaluated as "mean" estimates of benefits. Each person-rem of averted offsite exposure was assigned a value of $5000.

alternative assumes that a second squib valve vendor does exist and that the vendor would provide only the two diverse IRWST squib valves. The cost impact does not include the additional first-time engineering and qualification testing that would be incurred by the second vendor (Westinghouse estimated that those costs could be more than a million dollars). As a result, Westinghouse concluded that this design alternative would not be practicable because of the uncertainty in availability of a second squib valve design/vendor and because of the uncertainty in reliability of another type of check valve. The NRC considers the rationale set forth by Westinghouse regarding the potential reductions in reliability and high costs associated with obtaining diverse valves to be reasonable. On the bases of these arguments, the NRC concludes that this design alternative need not be pursued further.

3.7.3.2 Self-Actuating Containment Isolation Valves

This design alternative would reduce the likelihood of containment isolation failure by adding self-actuating valves or enhancing the existing containment isolation valves for automatic closure when containment conditions indicate that a severe accident has occurred. Conceptually, the design would be either an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures, thereby providing the self-actuating function to vent the air operator of a fail-closed valve. This design alternative is estimated to impact releases from containment by only 10%. It has a cost-benefit ratio of $33,000/person-rem, and achieves this ratio primarily because of its low capital costs.

This improvement to the containment isolation capability would appear to be effective in reducing offsite releases for accidents involving either external or internal events. Also, the effectiveness of this design alternative would not be affected by the design changes made as a result of the PRA. The addition of this design alternative would impose minor operational disadvantages to the plant in that the operations and maintenance staff would require additional training. In addition, these automatic features would require periodic testing to ensure that they are functioning properly.

The underlying question regarding this design alternative is whether it can realistically be implemented for only a cost of $33,000. The cost estimate does not appear to include the first-time engineering and qualification testing that would be required to demonstrate that the valve would perform its intended function in a timely and reliable manner. In addition, the costs associated with periodic testing and maintenance do not appear to have been included. The NRC staff believes that the actual costs of this design alternative would be substantially higher than Westinghouse's estimate (perhaps by a factor of 10) when all related costs are realistically considered. On the basis of the unfavorable cost-benefit ratio, and the expectation that actual costs would be even higher than estimated by Westinghouse, the NRC concludes that this design alternative is not cost-beneficial and need not be pursued further.

3.8 Conclusions

As discussed in Section 19.1 of the AP600 FSER (NUREG-1512), Westinghouse extensively used the PRA results to arrive at a final AP600 design. As a result, the estimated CDF and risk calculated for the AP600 plant are very low relative to operating plants and in absolute terms. Moreover, the low CDF and risk for the AP600 design reflect Westinghouse's efforts to systematically minimize the effect of initiators/sequences that have been important contributors to

CDF in previous PWR PRAs. Westinghouse has achieved this objective largely by incorporating a number of hardware improvements in the AP600 design. These and other AP600 design features which contribute to low CDF and risk for the AP600 design are discussed in Section 19.1 of the AP600 FSER.

Because the AP600 design already contains numerous plant features oriented toward reducing CDF and risk, the benefits and risk reduction potential of additional plant improvements are significantly reduced. This is true for both internally and externally initiated events. Moreover, with the features already incorporated in the AP600 design, the ability to estimate CDF and risk approaches the limitations of probabilistic techniques. Specifically, when CDFs are already estimated to be 1 in 100,000 or 1,000,000 years, model performance and assumptions are incomplete, and supporting data is sparse or even nonexistent; data shortfalls could actually be the more important contributors to risk. Areas not modeled or incompletely modeled include human reliability, sabotage, rare initiating events, construction or design errors, and system interactions. Although improvements in the modeling of these areas may introduce additional contributors to CDF and risk, the NRC staff does not expect that additional contributions would change its conclusions in absolute terms.

The NRC concurs with Westinghouse's conclusion that none of the potential design modifications evaluated are justified on the basis of cost-benefit considerations. The NRC further concludes that it is unlikely that any other design changes would be justified on the basis of person-rem exposure considerations, because the estimated CDFs would remain very low on an absolute scale.

The NRC finds that Westinghouse's evaluation provides a sufficient basis to conclude that there is reasonable assurance that an amendment to 10 CFR Part 52 certifying the AP600 standard plant design would not exclude a severe accident design alternative for a nuclear facility referencing the AP600 design that could have been cost-beneficial had it been considered as part of the original application for design certification.

4.0 THE ENVIRONMENTAL IMPACT OF THE PROPOSED ACTION

Issuing an amendment to 10 CFR Part 52 certifying the AP600 standard plant design would not constitute a significant environmental impact. The amendment would merely codify the NRC's approval of the AP600 design (refer to NUREG-1512). Furthermore, because the amendment is a rule, there are no resources involved that would have alternative uses.

As described in Section 3 of this EA, the NRC reviewed alternatives to the design certification rule and alternative design features related to preventing and mitigating severe accidents. Consideration of alternatives under NEPA was necessary to show that the design certification rule is the appropriate course of action, and to ensure that the design referenced in the rulemaking would not exclude any cost-beneficial design changes related to the prevention and mitigation of severe accidents. The NRC concludes that the alternatives to certification did not provide for resolution of issues as did the design certification rule.

Design certification is in keeping with the Commission's intent (refer to the Standardization and Severe Accident Policy Statements and 10 CFR Part 52) to make future plants safer than the current generation plants, to achieve early resolution of licensing issues, and to achieve the safety benefits of standardization. Through its own independent analysis, the NRC also concludes that Westinghouse adequately considered an appropriate set of SAMDAs, and none were found to be cost-beneficial. Although no design changes resulted from reviewing the SAMDAs, Westinghouse had already incorporated certain features in the AP600 design on the basis of its PRA results. Section 3.2 of this EA presents 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 AP600 design (refer to Section 19.1.6 of NUREG-1512, "Use of PRA in the Design Process").

Finally, the design certification rule by itself would not authorize the siting, construction, or operation of a nuclear power plant. The issuance of a CP, ESP, COL, or OL, which references the AP600 design, will require a prospective applicant to address the environmental impacts of construction and operation at a specific site. At that time, the NRC will evaluate the environmental impacts and issue an EIS in accordance with NEPA. However, the SAMDA analysis has been completed as part of this EA and can be incorporated by reference as part of an EIS related to siting, construction, or operation of a nuclear plant that references the AP600 design.

5.0 AGENCIES AND PERSONS CONSULTED, AND SOURCES USED

The sources for this EA include Westinghouse's "AP600 Standard Safety Analysis Report" (Revision 25, dated August 19, 1998); and the NRC's "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design" (NUREG-1512, dated September 1998).

The NRC has determined under the National Environmental Policy Act of 1969, as amended, and the NRC's regulations in 10 CFR Part 51, Subpart A, that this rule is not a major Federal action significantly affecting the quality of the human environment. Therefore, the NRC has determined that preparation of an environmental impact statement for this rulemaking is not required. The basis for this determination, as documented in this final EA, is that the amendment to 10 CFR Part 52 would not authorize the siting, construction, or operation of a facility referencing the AP600 design; it would only codify the AP600 design. The NRC's finding of no significant environmental impact was published in the Federal Register on May 20, 1999 (64 *FR* 27626), with the proposed design certification rule and draft EA for the AP600 design. This Federal Register notice was also mailed to all State liaison officers and public utility commissions. These actions provided the public, along with Federal, State, or local officials, with an opportunity to comment on the NRC's assessment of the environmental impact of the proposed rulemaking. The NRC received no comments related to this EA. The NRC will evaluate the environmental impacts and issue an EIS in accordance with NEPA as part of any application(s) for the siting, construction, or operation of a nuclear facility.

CONGRESSIONAL REVIEW ACT FORM

Attachment 3

24722

Submission of Federal Rules Under the Congressional Review Act

☐ Senate ☐ House ☐ GAO

Please fill the circles electronically or with black pen or #2 pencil.

1. Name of Department or Agency	2. Subdivision or Office
U.S. Nuclear Regulatory Commission	Office of Nuclear Reactor Regulation

3. Rule Title

Design Certification Rule for the AP600 design

4. Rule Identification Number (RIN) or Other Unique Identifier (if applicable) RIN 3150-AG23

5. Major Rule ○ Non-major Rule ●

6. Final Rule ● Other ○ ______

7. With respect to this rule, did your agency solicit public comments? Yes ● No ○ N/A ○

8. Priority of Regulation (fill in one)

● Economically Significant; or Significant; or Substantive, Nonsignificant

○ Routine and Frequent or Informational/Administrative/Other (Do not complete the other side of this form if filled in above.)

9. Effective Date (if applicable) 30 days after publication of final rule in Federal Register

10. Concise Summary of Rule (fill in one or both) attached ○ stated in rule ●

Submitted by: ______________________ (signature)

Name: Dennis K. Rathbun

Title: Director, Office of Congressional Affairs

U.S. Nuclear Regulatory Commission

For Congressional Use Only:

Date Received: ______________

Committee of Jurisdiction: ______________

10/23/98

24722

		Yes	No	N/A
A.	With respect to this rule, did your agency prepare an analysis of costs and benefits?	○	●	○
B.	With respect to this rule, at the final rulemaking stage, did your agency			
	1. certify that the rule would not have a significant economic impact on a substantial number of small entities under 5 U.S.C. § 605(b)?	●	○	○
	2. prepare a final Regulatory Flexibility Analysis under 5 U.S.C. § 604(a)?	○	○	●
C.	With respect to this rule, did your agency prepare a written statement under § 202 of the Unfunded Mandates Reform Act of 1995?	○	○	●
D.	With respect to this rule, did your agency prepare an Environmental Assessment or an Environmental Impact Statement under the National Environmental Policy Act (NEPA)?	●	○	○
E.	Does this rule contain a collection of information requiring OMB approval under the Paperwork Reduction Act of 1995?	●	○	○
F.	Did you discuss any of the following in the preamble to the rule?	●	○	○
	• E.O. 12612, Federalism	○	●	○
	• E.O. 12630, Government Actions and Interference with Constitutionally Protected Property Rights	○	●	○
	• E.O. 12866, Regulatory Planning and Review	○	●	○
	• E.O. 12875, Enhancing the Intergovernmental Partnership	○	●	○
	• E.O. 12988, Civil Justice Reform	○	●	○
	• E.O. 13045, Protection of Children from Environmental Health Risks and Safety Risks	○	●	○
	• Other statutes or executive orders discussed in the preamble concerning the rulemaking process (please specify)			

NATIONAL TECHNOLOGY TRANSFER & ADVANCEMENT ACT