TABLE B–1—SUMMARY OF FINDINGS ON NEPA ISSUES FOR LICENSE RENEWAL OF NUCLEAR POWER PLANTS [1]—Continued

Issue	Category [2]	Findings [3]
Environmental Justice		
Environmental justice [6]	[4] NA	NONE. The need for and the content of an analysis of environmental justice will be addressed in plant-specific reviews.[6]

[1] Data supporting this table are contained in NUREG–1437, ''Generic Environmental Impact Statement for License Renewal of Nuclear Plants'' (May 1996) and NUREG–1437, Vol. 1, Addendum 1, ''Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Main Report Section 6.3—'Transportation,' Table 9.1 'Summary of findings on NEPA issues for license renewal of nuclear power plants,' Final Report'' (August 1999).

[2] The numerical entries in this column are based on the following category definitions:

Category 1: For the issue, the analysis reported in the Generic Environmental Impact Statement has shown:

(1) The environmental impacts associated with the issue have been determined to apply either to all plants or, for some issues, to plants having a specific type of cooling system or other specified plant or site characteristic;

(2) A single significance level (*i.e.*, small, moderate, or large) has been assigned to the impacts (except for collective off site radiological impacts from the fuel cycle and from high level waste and spent fuel disposal); and

(3) Mitigation of adverse impacts associated with the issue has been considered in the analysis, and it has been determined that additional plant-specific mitigation measures are likely not to be sufficiently beneficial to warrant implementation.

The generic analysis of the issue may be adopted in each plant-specific review.

Category 2: For the issue, the analysis reported in the Generic Environmental Impact Statement has shown that one or more of the criteria of Category 1 cannot be met, and therefore additional plant-specific review is required.

[3] The impact findings in this column are based on the definitions of three significance levels. Unless the significance level is identified as beneficial, the impact is adverse, or in the case of ''small,'' may be negligible. The definitions of significance follow:

SMALL—For the issue, environmental effects are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource. For the purposes of assessing radiological impacts, the Commission has concluded that those impacts that do not exceed permissible levels in the Commission's regulations are considered small as the term is used in this table.

MODERATE—For the issue, environmental effects are sufficient to alter noticeably, but not to destabilize, important attributes of the resource.

LARGE—For the issue, environmental effects are clearly noticeable and are sufficient to destabilize important attributes of the resource.

For issues where probability is a key consideration (*i.e.*, accident consequences), probability was a factor in determining significance.

[4] NA (not applicable). The categorization and impact finding definitions do not apply to these issues.

[5] If, in the future, the Commission finds that, contrary to current indications, a consensus has been reached by appropriate Federal health agencies that there are adverse health effects from electromagnetic fields, the Commission will require applicants to submit plant-specific reviews of these health effects as part of their license renewal applications. Until such time, applicants for license renewal are not required to submit information on this issue.

[6] Environmental Justice was not addressed in NUREG–1437, ''Generic Environmental Impact Statement for License Renewal of Nuclear Plants,'' because guidance for implementing Executive Order 12898 issued on February 11, 1994, was not available prior to completion of NUREG–1437. This issue will be addressed in individual license renewal reviews.

[61 FR 66546, Dec. 18, 1996, as amended at 62 FR 59276, Nov. 3, 1997; 64 FR 48507, Sept. 3, 1999; 66 FR 39278, July 30, 2001]

Subpart B [Reserved]

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

GENERAL PROVISIONS

Subpart B—Standard Design Certifications

Subpart C—Combined Licenses

Subpart D [Reserved]

Subpart E—Standard Design Approvals

Subpart F—Manufacturing Licenses

Subpart G [Reserved]

Subpart H—Enforcement

AUTHORITY: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2133, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846); sec. 1704, 112 Stat. 2750 (44 U.S.C. 3504 note).

EFFECTIVE DATE NOTE: At 73 FR 63571, Oct. 24, 2008, the authority citation for part 52 was revised, effective Feb. 23, 2009. For the convenience of the user, the revised text is set forth as follows:

AUTHORITY: Sec. 161, 68 Stat. 948, as amended, sec. 274, 73 Stat. 688 (42 U.S.C. 2201, 2021); sec. 201, 88 Stat. 1242, as amended (42 U.S.C. 5841); sec. 1704, 112 Stat. 2750 (44 U.S.C. 3504 note); Energy Policy Act of 2005, Pub. L. No. 109–58, 119 Stat. 594 (2005).

Sections 150.3, 150.15, 150.15a, 150.31, 150.32 also issued under secs. 11e(2), 81, 68 Stat. 923, 935, as amended, secs. 83, 84, 92 Stat. 3033, 3039 (42 U.S.C. 2014e(2), 2111, 2113, 2114).

Section 150.14 also issued under sec. 53, 68 Stat. 930, as amended (42 U.S.C. 2073).

Section 150.15 also issued under secs. 135, 141, Pub. L. 97–425, 96 Stat. 2232, 2241 (42 U.S.C. 10155, 10161).

Section 150.17a also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152).

Section 150.30 also issued under sec. 234, 83 Stat. 444 (42 U.S.C. 2282).

SOURCE: 72 FR 49517, Aug. 28, 2007, unless otherwise noted.

GENERAL PROVISIONS

§ 52.0 Scope; applicability of 10 CFR Chapter I provisions.

(a) This part governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities licensed under Section 103 of the Atomic Energy Act of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242). This part also gives notice to all persons who knowingly provide to any holder of or applicant for an approval, certification, permit, or license, or to a contractor, subcontractor, or consultant of any of them, components, equipment, materials, or other goods or services that relate to the activities of a holder of or applicant for an approval, certification, permit, or license, subject to this part, that they may be individually subject to NRC enforcement action for violation of the provisions in 10 CFR 52.4.

(b) Unless otherwise specifically provided for in this part, the regulations in 10 CFR Chapter I apply to a holder of or applicant for an approval, certification, permit, or license. A holder of or applicant for an approval, certification, permit, or license issued under this part shall comply with all requirements in 10 CFR Chapter I that are applicable. A license, approval, certification, or permit issued under this part is subject to all requirements in 10 CFR Chapter I which, by their terms, are applicable to early site permits, design certifications, combined licenses, design approvals, or manufacturing licenses.

§ 52.1 Definitions.

(a) As used in this part—

Combined license means a combined construction permit and operating license with conditions for a nuclear power facility issued under subpart C of this part.

Decommission means to remove a facility or site safely from service and reduce residual radioactivity to a level that permits—

(i) Release of the property for unrestricted use and termination of the license; or

(ii) Release of the property under restricted conditions and termination of the license.

Design characteristics are the actual features of a reactor or reactors. Design characteristics are specified in a standard design approval, a standard design certification, a combined license application, or a manufacturing license.

Design parameters are the postulated features of a reactor or reactors that could be built at a proposed site. Design parameters are specified in an early site permit.

Early site permit means a Commission approval, issued under subpart A of this part, for a site or sites for one or more nuclear power facilities. An early site permit is a partial construction permit.

License means a license, including an early site permit, combined license or manufacturing license under this part or a renewed license issued by the Commission under this part or part 54 of this chapter.

Licensee means a person who is authorized to conduct activities under a license issued by the Commission.

Limited work authorization means the authorization provided by the Director of New Reactors or the Director of Nuclear Reactor Regulation under §50.10 of this chapter.

Major feature of the emergency plans means an aspect of those plans necessary to:

(i) Address in whole or part one or more of the 16 standards in 10 CFR 50.47(b); or

(ii) Describe the emergency planning zones as required in 10 CFR 50.33(g).

Manufacturing license means a license, issued under subpart F of this part, authorizing the manufacture of nuclear power reactors but not their construction, installation, or operation at the sites on which the reactors are to be operated.

Modular design means a nuclear power station that consists of two or more essentially identical nuclear reactors (modules) and each module is a separate nuclear reactor capable of being operated independent of the state of completion or operating condition of any other module co-located on the same site, even though the nuclear power station may have some shared or common systems.

Prototype plant means a nuclear power plant that is used to test new safety features, such as the testing required under 10 CFR 50.43(e). The prototype plant is similar to a first-of-a-kind or standard plant design in all features and size, but may include additional safety features to protect the public and the plant staff from the possible consequences of accidents during the testing period.

Site characteristics are the actual physical, environmental and demographic features of a site. Site characteristics are specified in an early site permit or in a final safety analysis report for a combined license.

Site parameters are the postulated physical, environmental and demographic features of an assumed site. Site parameters are specified in a standard design approval, standard design certification, or manufacturing license.

Standard design means a design which is sufficiently detailed and complete to support certification or approval in accordance with subpart B or E of this part, and which is usable for a multiple number of units or at a multiple number of sites without reopening or repeating the review.

Standard design approval or *design approval* means an NRC staff approval, issued under subpart E of this part, of a final standard design for a nuclear power reactor of the type described in 10 CFR 50.22. The approval may be for either the final design for the entire reactor facility or the final design of major portions thereof.

Standard design certification or *design certification* means a Commission approval, issued under subpart B of this part, of a final standard design for a nuclear power facility. This design may be referred to as a certified standard design.

(b) All other terms in this part have the meaning set out in 10 CFR 50.2, or Section 11 of the Atomic Energy Act, as applicable.

[72 FR 49517, Aug. 28, 2007, as amended at 72 FR 57446, Oct. 9, 2007]

§ 52.2 Interpretations.

Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations in this part by any officer or employee of the Commission other than a written interpretation by the General Counsel will be recognized to be binding upon the Commission.

§ 52.3 Written communications.

(a) *General requirements*. All correspondence, reports, applications, and other written communications from an applicant, licensee, or holder of a standard design approval to the Nuclear Regulatory Commission concerning the regulations in this part, individual license conditions, or the terms and conditions of an early site permit or standard design approval, must be sent either by mail addressed: ATTN: Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; by hand delivery to the NRC's offices at 11555 Rockville Pike, Rockville, Maryland, between the hours of 7:30 a.m. and 4:15 p.m. eastern time; or, where practicable, by electronic submission, for example, via Electronic Information Exchange, e-mail, or CD–ROM. Electronic submissions must be made in a manner that enables the NRC to receive, read, authenticate, distribute, and archive the submission, and process and retrieve it a single page at a time. Detailed guidance on making electronic submissions can be obtained by visiting the NRC's Web site at *http://www.nrc.gov/site-help/eie.html*, by calling (301) 415–6030, by e-mail at *EIE@nrc.gov*, or by writing the Office of Information Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. The guidance discusses, among other topics, the formats the NRC can accept, the use of electronic signatures, and the treatment of nonpublic information. If the communication is on paper, the signed original must be sent. If a submission due date falls on a Saturday, Sunday, or Federal holiday, the next Federal working day becomes the official due date.

(b) *Distribution requirements.* Copies of all correspondence, reports, and other written communications concerning the regulations in this part or individual license conditions, or the terms and conditions of an early site permit or standard design approval, must be submitted to the persons listed in paragraph (b)(1) of this section (addresses for the NRC Regional Offices are listed in appendix D to part 20 of this chapter).

(1) *Applications for amendment of permits and licenses; reports; and other communications.* All written communications (including responses to: generic letters, bulletins, information notices, regulatory information summaries, inspection reports, and miscellaneous requests for additional information) that are required of holders of early site permits, standard design approvals, combined licenses, or manufacturing licenses issued under this part must be submitted as follows, except as otherwise specified in paragraphs (b)(2) through (b)(7) of this section: to the NRC's Document Control Desk (if on paper, the signed original), with a copy to the appropriate Regional Office, and a copy to the appropriate NRC Resident Inspector, if one has been assigned to the site of the facility or the place of manufacture of a reactor licensed under subpart F of this part.

(2) *Applications and amendments to applications.* Applications for early site permits, standard design approvals, combined licenses, manufacturing licenses and amendments to any of these types of applications must be submitted to the NRC's Document Control Desk, with a copy to the appropriate Regional Office, and a copy to the appropriate NRC Resident Inspector, if one has been assigned to the site of the facility or the place of manufacture of a reactor licensed under subpart F of this part, except as otherwise specified in paragraphs (b)(3) through (b)(7) of this section. If the application or amendment is on paper, the submission to the Document Control Desk must be the signed original.

(3) *Acceptance review application.* Written communications required for an application for determination of suitability for docketing must be submitted to the NRC's Document Control Desk, with a copy to the appropriate Regional Office. If the communication is on paper, the submission to the Document Control Desk must be the signed original.

(4) *Security plan and related submissions.* Written communications, as defined in paragraphs (b)(4)(i) through (iv) of this section, must be submitted to the NRC's Document Control Desk, with a copy to the appropriate Regional Office. If the communication is on paper, the submission to the Document Control Desk must be the signed original.

(i) Physical security plan under § 52.79 of this chapter;

(ii) Safeguards contingency plan under § 52.79 of this chapter;

(iii) Change to security plan, guard training and qualification plan, or safeguards contingency plan made without prior Commission approval under § 50.54(p) of this chapter;

(iv) Application for amendment of physical security plan, guard training and qualification plan, or safeguards contingency plan under § 50.90 of this chapter.

(5) *Emergency plan and related submissions.* Written communications as defined in paragraphs (b)(5)(i) through (iii) of this section must be submitted to the NRC's Document Control Desk, with a copy to the appropriate Regional Office, and a copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility. If the communication is on

paper, the submission to the Document Control Desk must be the signed original.

(i) Emergency plan under § 52.17(b) or § 52.79(a);

(ii) Change to an emergency plan under § 50.54(q) of this chapter;

(iii) Emergency implementing procedures under appendix E, Section V of part 50 of this chapter.

(6) *Updated FSAR.* An updated final safety analysis report (FSAR) or replacement pages under § 50.71(e) of this chapter, or the regulations in this part must be submitted to the NRC's Document Control Desk, with a copy to the appropriate Regional Office, and a copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility or the place of manufacture of a reactor licensed under subpart F of this part. Paper copy submissions may be made using replacement pages; however, if a licensee chooses to use electronic submission, all subsequent updates or submissions must be performed electronically on a total replacement basis. If the communication is on paper, the submission to the Document Control Desk must be the signed original. If the communications are submitted electronically, see Guidance for Electronic Submissions to the Commission.

(7) *Quality assurance related submissions.* (i) A change to the safety analysis report quality assurance program description under § 50.54(a)(3) or § 50.55(f)(4) of this chapter, or a change to a licensee's NRC-accepted quality assurance topical report under § 50.54(a)(3) or § 50.55(f)(4) of this chapter, must be submitted to the NRC's Document Control Desk, with a copy to the appropriate Regional Office, and a copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility. If the communication is on paper, the submission to the Document Control Desk must be the signed original.

(ii) A change to an NRC-accepted quality assurance topical report from nonlicensees (*i.e.*, architect/engineers, NSSS suppliers, fuel suppliers, constructors, etc.) must be submitted to the NRC's Document Control Desk. If the communication is on paper, the signed original must be sent.

(8) *Certification of permanent cessation of operations.* The licensee's certification of permanent cessation of operations under § 52.110(a)(1), must state the date on which operations have ceased or will cease, and must be submitted to the NRC's Document Control Desk. This submission must be under oath or affirmation.

(9) *Certification of permanent fuel removal.* The licensee's certification of permanent fuel removal under § 52.110(a)(1), must state the date on which the fuel was removed from the reactor vessel and the disposition of the fuel, and must be submitted to the NRC's Document Control Desk. This submission must be under oath or affirmation.

(c) *Form of communications.* All paper copies submitted to meet the requirements set forth in paragraph (b) of this section must be typewritten, printed or otherwise reproduced in permanent form on unglazed paper. Exceptions to these requirements imposed on paper submissions may be granted for the submission of micrographic, photographic, or similar forms.

(d) *Regulation governing submission.* Applicants, licensees, and holders of standard design approvals submitting correspondence, reports, and other written communications under the regulations of this part are requested but not required to cite whenever practical, in the upper right corner of the first page of the submission, the specific regulation or other basis requiring submission.

§ 52.4 Deliberate misconduct.

(a) *Applicability.* This section applies to any:

(1) Licensee;

(2) Holder of a standard design approval;

(3) Applicant for a standard design certification;

(4) Applicant for a license or permit;

(5) Applicant for a standard design approval;

(6) Employee of a licensee;

(7) Employee of an applicant for a license, a standard design certification, or a standard design approval;

(8) Any contractor (including a supplier or consultant), subcontractor, or

employee of a contractor or subcontractor of any licensee; or

(9) Any contractor (including a supplier or consultant), subcontractor, or employee of a contractor or subcontractor of any applicant for a license, a standard design certification, or a standard design approval.

(b) *Definitions*. For purposes of this section:

Deliberate misconduct means an intentional act or omission that a person or entity knows:

(i) Would cause a licensee or an applicant for a license, standard design certification, or standard design approval to be in violation of any rule, regulation, or order; or any term, condition, or limitation, of any license, standard design certification, or standard design approval; or

(ii) Constitutes a violation of a requirement, procedure, instruction, contract, purchase order, or policy of a licensee, holder of a standard design approval, applicant for a license, standard design certification, or standard design approval, or contractor, or subcontractor.

(c) *Prohibition against deliberate misconduct*. Any person or entity subject to this section, who knowingly provides to any licensee, any applicant for a license, standard design certification or standard design approval, or a contractor, or subcontractor of a person or entity subject to this section, any components, equipment, materials, or other goods or services that relate to a licensee's or applicant's activities under this part, may not:

(1) Engage in deliberate misconduct that causes or would have caused, if not detected, a licensee, holder of a standard design approval, or applicant to be in violation of any rule, regulation, or order; or any term, condition, or limitation of any license issued by the Commission, any standard design approval, or standard design certification; or

(2) Deliberately submit to the NRC; a licensee, an applicant for a license, standard design certification or standard design approval; or a licensee's, standard design approval holder's, or applicant's contractor or subcontractor, information that the person submitting the information knows to be incomplete or inaccurate in some respect material to the NRC.

(d) A person or entity who violates paragraph (c)(1) or (c)(2) of this section may be subject to enforcement action in accordance with the procedures in 10 CFR part 2, subpart B.

§ 52.5 Employee protection.

(a) Discrimination by a Commission licensee, holder of a standard design approval, an applicant for a license, standard design certification, or standard design approval, a contractor or subcontractor of a Commission licensee, holder of a standard design approval, applicant for a license, standard design certification, or standard design approval, against an employee for engaging in certain protected activities is prohibited. Discrimination includes discharge and other actions that relate to compensation, terms, conditions, or privileges of employment. The protected activities are established in Section 211 of the Energy Reorganization Act of 1974, as amended, and in general are related to the administration or enforcement of a requirement imposed under the Atomic Energy Act or the Energy Reorganization Act.

(1) The protected activities include but are not limited to:

(i) Providing the Commission or his or her employer information about alleged violations of either of the statutes named in the introductory text of paragraph (a) of this section or possible violations of requirements imposed under either of those statutes;

(ii) Refusing to engage in any practice made unlawful under either of the statutes named in the introductory text of paragraph (a) of this section or under these requirements if the employee has identified the alleged illegality to the employer;

(iii) Requesting the Commission to institute action against his or her employer for the administration or enforcement of these requirements;

(iv) Testifying in any Commission proceeding, or before Congress, or at any Federal or State proceeding regarding any provision (or proposed provision) of either of the statutes named in the introductory text of paragraph (a) of this section; and

(v) Assisting or participating in, or is about to assist or participate in, these activities.

(2) These activities are protected even if no formal proceeding is actually initiated as a result of the employee assistance or participation.

(3) This section has no application to any employee alleging discrimination prohibited by this section who, acting without direction from his or her employer (or the employer's agent), deliberately causes a violation of any requirement of the Energy Reorganization Act of 1974, as amended, or the Atomic Energy Act of 1954, as amended.

(b) Any employee who believes that he or she has been discharged or otherwise discriminated against by any person for engaging in protected activities specified in paragraph (a)(1) of this section may seek a remedy for the discharge or discrimination through an administrative proceeding in the Department of Labor. The administrative proceeding must be initiated within 180 days after an alleged violation occurs. The employee may do this by filing a complaint alleging the violation with the Department of Labor, Employment Standards Administration, Wage and Hour Division. The Department of Labor may order reinstatement, back pay, and compensatory damages.

(c) A violation of paragraph (a), (e), or (f) of this section by a Commission licensee, a holder of a standard design approval, an applicant for a Commission license, standard design certification, or a standard design approval, or a contractor or subcontractor of a Commission licensee, holder of a standard design approval, or any applicant may be grounds for—

(1) Denial, revocation, or suspension of the license or standard design approval;

(2) Withdrawal or revocation of a proposed or final standard design certification;

(3) Imposition of a civil penalty on the licensee, holder of a standard design approval, or applicant (including an applicant for a standard design certification under this part following Commission adoption of final design certification rule) or a contractor or subcontractor of the licensee, holder of a standard design approval, or applicant.

(4) Other enforcement action.

(d) Actions taken by an employer, or others, which adversely affect an employee may be predicated upon nondiscriminatory grounds. The prohibition applies when the adverse action occurs because the employee has engaged in protected activities. An employee's engagement in protected activities does not automatically render him or her immune from discharge or discipline for legitimate reasons or from adverse action dictated by nonprohibited considerations.

(e)(1) Each licensee, each holder of a standard design approval, and each applicant for a license, standard design certification, or standard design approval, shall prominently post the revision of NRC Form 3, ''Notice to Employees,'' referenced in 10 CFR 19.11(e). This form must be posted at locations sufficient to permit employees protected by this section to observe a copy on the way to or from their place of work. Premises must be posted not later than thirty (30) days after an application is docketed and remain posted while the application is pending before the Commission, during the term of the license, standard design certification, or standard design approval under 10 CFR part 52, and for 30 days following license termination or the expiration or termination of the standard design certification or standard design approval under 10 CFR part 52.

(2) Copies of NRC Form 3 may be obtained by writing to the Regional Administrator of the appropriate U.S. Nuclear Regulatory Commission Regional Office listed in appendix D to part 20 of this chapter, by calling (301) 415–7232, via e-mail to *FORMS.Resource@nrc.gov*, or by visiting the NRC's Web site at *http://www.nrc.gov* and selecting forms from the index found on the NRC's home page.

(f) No agreement affecting the compensation, terms, conditions, or privileges of employment, including an agreement to settle a complaint filed by an employee with the Department of Labor under Section 211 of the Energy Reorganization Act of 1974, as amended, may contain any provision which would prohibit, restrict, or otherwise discourage an employee from participating in protected activity as defined in paragraph (a)(1) of this section including, but not limited to, providing information to the NRC or to his or her employer on potential violations or other matters within NRC's regulatory responsibilities.

(g) Part 19 of this chapter sets forth requirements and regulatory provisions applicable to licensees, holders of a standard design approval, applicants for a license, standard design certification, or standard design approval, and contractors or subcontractors of a Commission licensee, or holder of a standard design approval, and are in addition to the requirements in this section.

[72 FR 49517, Aug. 28, 2007, as amended at 72 FR 63974, Nov. 14, 2007; 73 FR 30458, May 28, 2008]

§ 52.6 Completeness and accuracy of information.

(a) Information provided to the Commission by a licensee (including an early site permit holder, a combined license holder, and a manufacturing license holder), a holder of a standard design approval under this part, and an applicant for a license or an applicant for a standard design certification or a standard design approval under this part, and information required by statute or by the Commission's regulations, orders, license conditions, or terms and conditions of a standard design approval to be maintained by the licensee, the holder of a standard design approval under this part, the applicant for a standard design certification under this part following Commission adoption of a final design certification rule, and an applicant for a license, a standard design certification, or a standard design approval under this part shall be complete and accurate in all material respects.

(b) Each applicant or licensee, each holder of a standard design approval under this part, and each applicant for a standard design certification under this part following Commission adoption of a final design certification regulation, shall notify the Commission of information identified by the applicant or the licensee as having for the regulated activity a significant implication for public health and safety or common defense and security. An applicant, licensee, or holder violates this paragraph only if the applicant, licensee, or holder fails to notify the Commission of information that the applicant, licensee, or holder has been identified as having a significant implication for public health and safety or common defense and security. Notification shall be provided to the Administrator of the appropriate Regional Office within 2 working days of identifying the information. This requirement is not applicable to information which is already required to be provided to the Commission by other reporting or updating requirements.

§ 52.7 Specific exemptions.

The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part. The Commission's consideration will be governed by § 50.12 of this chapter, unless other criteria are provided for in this part, in which case the Commission's consideration will be governed by the criteria in this part. Only if those criteria are not met will the Commission's consideration be governed by § 50.12 of this chapter. The Commission's consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts.

§ 52.8 Combining licenses; elimination of repetition.

(a) An applicant for a license under this part may combine in its application several applications for different kinds of licenses under the regulations of this chapter.

(b) An applicant may incorporate by reference in its application information contained in previous applications, statements or reports filed with the Commission, provided, however, that such references are clear and specific.

(c) The Commission may combine in a single license the activities of an applicant which would otherwise be licensed separately.

§ 52.9 Jurisdictional limits.

No permit, license, standard design approval, or standard design certification under this part shall be deemed to have been issued for activities which are not under or within the jurisdiction of the United States.

§ 52.10 Attacks and destructive acts.

Neither an applicant for a license to manufacture, construct, and operate a utilization facility under this part, nor for an amendment to this license, or an applicant for an early site permit, a standard design certification, or standard design approval under this part, or for an amendment to the early site permit, standard design certification, or standard design approval, is required to provide for design features or other measures for the specific purpose of protection against the effects of—

(a) Attacks and destructive acts, including sabotage, directed against the facility by an enemy of the United States, whether a foreign government or other person; or

(b) Use or deployment of weapons incident to U.S. defense activities.

§ 52.11 Information collection requirements: OMB approval.

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act (44 U.S.C. 3501 et seq.). The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number. OMB has approved the information collection requirements contained in this part under Control Number 3150–0151.

(b) The approved information collection requirements contained in this part appear in §§ 52.7, 52.15, 52.16, 52.17, 52.29, 52.35, 52.39, 52.45, 52.46, 52.47, 52.57, 52.63, 52.75, 52.77, 52.79, 52.80, 52.93, 52.99, 52.110, 52.135, 52.136, 52.137, 52.155, 52.156, 52.157, 52.158, 52.171, 52.177, and appendices A, B, C, D, and N of part 52.

Subpart A—Early Site Permits

§ 52.12 Scope of subpart.

This subpart sets out the requirements and procedures applicable to Commission issuance of an early site permit for approval of a site for one or more nuclear power facilities separate from the filing of an application for a construction permit or combined license for the facility.

§ 52.13 Relationship to other subparts.

This subpart applies when any person who may apply for a construction permit under 10 CFR part 50, or for a combined license under this part seeks an early site permit from the Commission separately from an application for a construction permit or a combined license.

§ 52.15 Filing of applications.

(a) Any person who may apply for a construction permit under 10 CFR part 50, or for a combined license under this part, may file an application for an early site permit with the Director, Office of New Reactors, or the Director, Office of Nuclear Reactor Regulation, as appropriate. An application for an early site permit may be filed notwithstanding the fact that an application for a construction permit or a combined license has not been filed in connection with the site for which a permit is sought.

(b) The application must comply with the applicable filing requirements of §§ 52.3 and 50.30 of this chapter.

(c) The fees associated with the filing and review of an application for the initial issuance or renewal of an early site permit are set forth in 10 CFR part 170.

§ 52.16 Contents of applications; general information.

The application must contain all of the information required by 10 CFR 50.33(a) through (d) and (j) of this chapter.

§ 52.17 Contents of applications; technical information.

(a) For applications submitted before September 27, 2007, the rule provisions in effect at the date of docketing apply unless otherwise requested by the applicant in writing. The application must contain:

(1) A site safety analysis report. The site safety analysis report shall include the following:

(i) The specific number, type, and thermal power level of the facilities, or range of possible facilities, for which the site may be used;

(ii) The anticipated maximum levels of radiological and thermal effluents each facility will produce;

(iii) The type of cooling systems, intakes, and outflows that may be associated with each facility;

(iv) The boundaries of the site;

(v) The proposed general location of each facility on the site;

(vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated;

(vii) The location and description of any nearby industrial, military, or transportation facilities and routes;

(viii) The existing and projected future population profile of the area surrounding the site;

(ix) A description and safety assessment of the site on which a facility is to be located. The assessment must contain an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors identified in paragraphs (a)(1)(ix)(A) and (a)(1)(ix)(B) of this section. In performing this assessment, an applicant shall assume a fission product release[1] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:

(A) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[2] total effective dose equivalent (TEDE).

(B) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE;

(x) Information demonstrating that site characteristics are such that adequate security plans and measures can be developed;

(xi) For applications submitted after September 27, 2007, a description of the quality assurance program applied to

[1] The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

[2] A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accidents.

site-related activities for the future design, fabrication, construction, and testing of the structures, systems, and components of a facility or facilities that may be constructed on the site. Appendix B to 10 CFR part 50 sets forth the requirements for quality assurance programs for nuclear power plants. The description of the quality assurance program for a nuclear power plant site shall include a discussion of how the applicable requirements of appendix B to part 50 of this chapter will be satisfied; and

(xii) An evaluation of the site against applicable sections of the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in analytical techniques and procedural measures proposed for a site and those corresponding techniques and measures given in the SRP acceptance criteria. Where such a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP is not a substitute for the regulations, and compliance is not a requirement.

(2) A complete environmental report as required by 10 CFR 51.50(b).

(b)(1) The site safety analysis report must identify physical characteristics of the proposed site, such as egress limitations from the area surrounding the site, that could pose a significant impediment to the development of emergency plans. If physical characteristics are identified that could pose a significant impediment to the development of emergency plans, the application must identify measures that would, when implemented, mitigate or eliminate the significant impediment.

(2) The site safety analysis report may also:

(i) Propose major features of the emergency plans, in accordance with the pertinent standards of 10 CFR 50.47, and the requirements of appendix E to 10 CFR part 50, such as the exact size and configuration of the emergency planning zones, for review and approval by NRC, in consultation with the Department of Homeland Security (DHS) in the absence of complete and integrated emergency plans; or

(ii) Propose complete and integrated emergency plans for review and approval by the NRC, in consultation with DHS, in accordance with the applicable standards of 10 CFR 50.47, and the requirements of appendix E to 10 CFR part 50. To the extent approval of emergency plans is sought, the application must contain the information required by §§ 50.33(g) and (j) of this chapter.

(3) Emergency plans submitted under paragraph (b)(2)(ii) of this section must include the proposed inspections, tests, and analyses that the holder of a combined license referencing the early site permit shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will be operated in conformity with the emergency plans, the provisions of the Act, and the Commission's rules and regulations. Major features of an emergency plan submitted under paragraph (b)(2)(i) of this section may include proposed inspections, tests, analyses, and acceptance criteria.

(4) Under paragraphs (b)(1) and (b)(2)(i) of this section, the site safety analysis report must include a description of contacts and arrangements made with Federal, State, and local governmental agencies with emergency planning responsibilities. The site safety analysis report must contain any certifications that have been obtained. If these certifications cannot be obtained, the site safety analysis report must contain information, including a utility plan, sufficient to show that the proposed plans provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site. Under the option set forth in paragraph (b)(2)(ii) of this section, the applicant shall make good faith efforts to obtain from the same governmental agencies certifications that:

(i) The proposed emergency plans are practicable;

(ii) These agencies are committed to participating in any further development of the plans, including any required field demonstrations, and

(iii) That these agencies are committed to executing their responsibilities under the plans in the event of an emergency.

(c) An applicant may request that a limited work authorization under 10 CFR 50.10 be issued in conjunction with the early site permit. The application must include the information otherwise required by 10 CFR 50.10(d)(3). Applications submitted before, and pending as of November 8, 2007, must include the information required by § 52.17(c) effective on the date of docketing.

[72 FR 49517, Aug. 28, 2007, as amended at 72 FR 57447, Oct. 9, 2007]

EFFECTIVE DATE NOTE: At 73 FR 63571, Oct. 24, 2008, § 52.17 was amended by adding paragraph (d), effective Feb. 23, 2009. For the convenience of the user, the added text is set forth as follows:

§ 52.17 Contents of applications; technical information.

* * * * *

(d) Each applicant for an early site permit under this part shall protect Safeguards Information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.

§ 52.18 Standards for review of applications.

Applications filed under this subpart will be reviewed according to the applicable standards set out in 10 CFR part 50 and its appendices and 10 CFR part 100. In addition, the Commission shall prepare an environmental impact statement during review of the application, in accordance with the applicable provisions of 10 CFR part 51. The Commission shall determine, after consultation with DHS, whether the information required of the applicant by § 52.17(b)(1) shows that there is no significant impediment to the development of emergency plans that cannot be mitigated or eliminated by measures proposed by the applicant, whether any major features of emergency plans submitted by the applicant under § 52.17(b)(2)(i) are acceptable in accordance with the applicable standards of 10 CFR 50.47 and the requirements of appendix E to 10 CFR part 50, and whether any emergency plans submitted by the applicant under § 52.17(b)(2)(ii) provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

§ 52.21 Administrative review of applications; hearings.

An early site permit is subject to all procedural requirements in 10 CFR part 2, including the requirements for docketing in § 2.101(a)(1) through (4) of this chapter, and the requirements for issuance of a notice of hearing in §§ 2.104(a) and (d) of this chapter, provided that the designated sections may not be construed to require that the environmental report, or draft or final environmental impact statement include an assessment of the benefits of construction and operation of the reactor or reactors, or an analysis of alternative energy sources. The presiding officer in an early site permit hearing shall not admit contentions proffered by any party concerning an assessment of the benefits of construction and operation of the reactor or reactors, or an analysis of alternative energy sources if those issues were not addressed by the applicant in the early site permit application. All hearings conducted on applications for early site permits filed under this part are governed by the procedures contained in subparts C, G, L, and N of 10 CFR part 2, as applicable.

§ 52.23 Referral to the Advisory Committee on Reactor Safeguards (ACRS).

The Commission shall refer a copy of the application for an early site permit to the ACRS. The ACRS shall report on those portions of the application which concern safety.

§ 52.24 Issuance of early site permit.

(a) After conducting a hearing under § 52.21 and receiving the report to be submitted by the ACRS under § 52.23, the Commission may issue an early site permit, in the form the Commission deems appropriate, if the Commission finds that:

(1) An application for an early site permit meets the applicable standards and requirements of the Act and the Commission's regulations;

(2) Notifications, if any, to other agencies or bodies have been duly made;

(3) There is reasonable assurance that the site is in conformity with the provisions of the Act, and the Commission's regulations;

(4) The applicant is technically qualified to engage in any activities authorized;

(5) The proposed inspections, tests, analyses and acceptance criteria, including any on emergency planning, are necessary and sufficient, within the scope of the early site permit, to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Act, and the Commission's regulations;

(6) Issuance of the permit will not be inimical to the common defense and security or to the health and safety of the public;

(7) Any significant adverse environmental impact resulting from activities requested under §52.17(c) can be redressed; and

(8) The findings required by subpart A of 10 CFR part 51 have been made.

(b) The early site permit must specify the site characteristics, design parameters, and terms and conditions of the early site permit the Commission deems appropriate. Before issuance of either a construction permit or combined license referencing an early site permit, the Commission shall find that any relevant terms and conditions of the early site permit have been met. Any terms or conditions of the early site permit that could not be met by the time of issuance of the construction permit or combined license, must be set forth as terms or conditions of the construction permit or combined license.

(c) The early site permit shall specify those 10 CFR 50.10 activities requested under §52.17(c) that the permit holder is authorized to perform.

[72 FR 49517, Aug. 28, 2007, as amended at 72 FR 57447, Oct. 9, 2007]

§52.25 Extent of activities permitted.

If the activities authorized by §52.24(c) are performed and the site is not referenced in an application for a construction permit or a combined license issued under subpart C of this part while the permit remains valid, then the early site permit remains in effect solely for the purpose of site redress, and the holder of the permit shall redress the site in accordance with the terms of the site redress plan required by §52.17(c). If, before redress is complete, a use not envisaged in the redress plan is found for the site or parts thereof, the holder of the permit shall carry out the redress plan to the greatest extent possible consistent with the alternate use.

§52.26 Duration of permit.

(a) Except as provided in paragraph (b) of this section, an early site permit issued under this subpart may be valid for not less than 10, nor more than 20 years from the date of issuance.

(b) An early site permit continues to be valid beyond the date of expiration in any proceeding on a construction permit application or a combined license application that references the early site permit and is docketed before the date of expiration of the early site permit, or, if a timely application for renewal of the permit has been docketed, before the Commission has determined whether to renew the permit.

(c) An applicant for a construction permit or combined license may, at its own risk, reference in its application a site for which an early site permit application has been docketed but not granted.

(d) Upon issuance of a construction permit or combined license, a referenced early site permit is subsumed, to the extent referenced, into the construction permit or combined license.

[72 FR 49517, Aug. 28, 2007. Redesignated at 72 FR 57447, Oct. 9, 2007]

§52.27 Limited work authorization after issuance of early site permit.

A holder of an early site permit may request a limited work authorization in accordance with § 50.10 of this chapter.

[72 FR 57447, Oct. 9, 2007]

§ 52.28 Transfer of early site permit.

An application to transfer an early site permit will be processed under 10 CFR 50.80.

§ 52.29 Application for renewal.

(a) Not less than 12, nor more than 36 months before the expiration date stated in the early site permit, or any later renewal period, the permit holder may apply for a renewal of the permit. An application for renewal must contain all information necessary to bring up to date the information and data contained in the previous application.

(b) Any person whose interests may be affected by renewal of the permit may request a hearing on the application for renewal. The request for a hearing must comply with 10 CFR 2.309. If a hearing is granted, notice of the hearing will be published in accordance with 10 CFR 2.309.

(c) An early site permit, either original or renewed, for which a timely application for renewal has been filed, remains in effect until the Commission has determined whether to renew the permit. If the permit is not renewed, it continues to be valid in certain proceedings in accordance with the provisions of § 52.27(b).

(d) The Commission shall refer a copy of the application for renewal to the ACRS. The ACRS shall report on those portions of the application which concern safety and shall apply the criteria set forth in § 52.31.

§ 52.31 Criteria for renewal.

(a) The Commission shall grant the renewal if it determines that:

(1) The site complies with the Act, the Commission's regulations, and orders applicable and in effect at the time the site permit was originally issued; and

(2) Any new requirements the Commission may wish to impose are:

(i) Necessary for adequate protection to public health and safety or common defense and security;

(ii) Necessary for compliance with the Commission's regulations, and orders applicable and in effect at the time the site permit was originally issued; or

(iii) A substantial increase in overall protection of the public health and safety or the common defense and security to be derived from the new requirements, and the direct and indirect costs of implementation of those requirements are justified in view of this increased protection.

(b) A denial of renewal for failure to comply with the provisions of § 52.31(a) does not bar the permit holder or another applicant from filing a new application for the site which proposes changes to the site or the way that it is used to correct the deficiencies cited in the denial of the renewal.

§ 52.33 Duration of renewal.

Each renewal of an early site permit may be for not less than 10, nor more than 20 years, plus any remaining years on the early site permit then in effect before renewal.

§ 52.35 Use of site for other purposes.

A site for which an early site permit has been issued under this subpart may be used for purposes other than those described in the permit, including the location of other types of energy facilities. The permit holder shall inform the Director, Office of New Reactors or Director, Office of Nuclear Reactor Regulation, as appropriate, (Director) of any significant uses for the site which have not been approved in the early site permit. The information about the activities must be given to the Director at least 30 days in advance of any actual construction or site modification for the activities. The information provided could be the basis for imposing new requirements on the permit, in accordance with the provisions of § 52.39. If the permit holder informs the Director that the holder no longer intends to use the site for a nuclear power plant, the Director may terminate the permit.

[73 FR 5724, Jan. 31, 2008]

§ 52.39 Finality of early site permit determinations.

(a) *Commission finality.* (1) Notwithstanding any provision in 10 CFR 50.109, while an early site permit is in effect under §§ 52.27 or 52.33, the Commission may not change or impose new site characteristics, design parameters, or terms and conditions, including emergency planning requirements, on the early site permit unless the Commission:

(i) Determines that a modification is necessary to bring the permit or the site into compliance with the Commission's regulations and orders applicable and in effect at the time the permit was issued;

(ii) Determines the modification is necessary to assure adequate protection of the public health and safety or the common defense and security;

(iii) Determines that a modification is necessary based on an update under paragraph (b) of this section; or

(iv) Issues a variance requested under paragraph (d) of this section.

(2) In making the findings required for issuance of a construction permit or combined license, or the findings required by § 52.103, or in any enforcement hearing other than one initiated by the Commission under paragraph (a)(1) of this section, if the application for the construction permit or combined license references an early site permit, the Commission shall treat as resolved those matters resolved in the proceeding on the application for issuance or renewal of the early site permit, except as provided for in paragraphs (b), (c), and (d) of this section.

(i) If the early site permit approved an emergency plan (or major features thereof) that is in use by a licensee of a nuclear power plant, the Commission shall treat as resolved changes to the early site permit emergency plan (or major features thereof) that are identical to changes made to the licensee's emergency plans in compliance with § 50.54(q) of this chapter occurring after issuance of the early site permit.

(ii) If the early site permit approved an emergency plan (or major features thereof) that is not in use by a licensee of a nuclear power plant, the Commission shall treat as resolved changes that are equivalent to those that could be made under § 50.54(q) of this chapter without prior NRC approval had the emergency plan been in use by a licensee.

(b) *Updating of early site permit-emergency preparedness.* An applicant for a construction permit, operating license, or combined license who has filed an application referencing an early site permit issued under this subpart shall update the emergency preparedness information that was provided under § 52.17(b), and discuss whether the updated information materially changes the bases for compliance with applicable NRC requirements.

(c) *Hearings and petitions.* (1) In any proceeding for the issuance of a construction permit, operating license, or combined license referencing an early site permit, contentions on the following matters may be litigated in the same manner as other issues material to the proceeding:

(i) The nuclear power reactor proposed to be built does not fit within one or more of the site characteristics or design parameters included in the early site permit;

(ii) One or more of the terms and conditions of the early site permit have not been met;

(iii) A variance requested under paragraph (d) of this section is unwarranted or should be modified;

(iv) New or additional information is provided in the application that substantially alters the bases for a previous NRC conclusion or constitutes a sufficient basis for the Commission to modify or impose new terms and conditions related to emergency preparedness; or

(v) Any significant environmental issue that was not resolved in the early site permit proceeding, or any issue involving the impacts of construction and operation of the facility that was resolved in the early site permit proceeding for which significant new information has been identified.

(2) Any person may file a petition requesting that the site characteristics, design parameters, or terms and conditions of the early site permit should be modified, or that the permit should be suspended or revoked. The petition will be considered in accordance with § 2.206 of this chapter. Before construction commences, the Commission shall consider the petition and determine whether any immediate action is required. If the petition is granted, an

appropriate order will be issued. Construction under the construction permit or combined license will not be affected by the granting of the petition unless the order is made immediately effective. Any change required by the Commission in response to the petition must meet the requirements of paragraph (a)(1) of this section.

(d) *Variances.* An applicant for a construction permit, operating license, or combined license referencing an early site permit may include in its application a request for a variance from one or more site characteristics, design parameters, or terms and conditions of the early site permit, or from the site safety analysis report. In determining whether to grant the variance, the Commission shall apply the same technically relevant criteria applicable to the application for the original or renewed early site permit. Once a construction permit or combined license referencing an early site permit is issued, variances from the early site permit will not be granted for that construction permit or combined license.

(e) *Early site permit amendment.* The holder of an early site permit may not make changes to the early site permit, including the site safety analysis report, without prior Commission approval. The request for a change to the early site permit must be in the form of an application for a license amendment, and must meet the requirements of 10 CFR 50.90 and 50.92.

(f) *Information requests.* Except for information requests seeking to verify compliance with the current licensing basis of the early site permit, information requests to the holder of an early site permit must be evaluated before issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information. Each evaluation performed by the NRC staff must be in accordance with 10 CFR 50.54(f), and must be approved by the Executive Director for Operations or his or her designee before issuance of the request.

Subpart B—Standard Design Certifications

§ 52.41 Scope of subpart.

(a) This subpart sets forth the requirements and procedures applicable to Commission issuance of rules granting standard design certifications for nuclear power facilities separate from the filing of an application for a construction permit or combined license for such a facility.

(b)(1) Any person may seek a standard design certification for an essentially complete nuclear power plant design which is an evolutionary change from light water reactor designs of plants which have been licensed and in commercial operation before April 18, 1989.

(2) Any person may also seek a standard design certification for a nuclear power plant design which differs significantly from the light water reactor designs described in paragraph (b)(1) of this section or uses simplified, inherent, passive, or other innovative means to accomplish its safety functions.

§ 52.43 Relationship to other subparts.

(a) This subpart applies to a person that requests a standard design certification from the NRC separately from an application for a combined license filed under subpart C of this part for a nuclear power facility. An applicant for a combined license may reference a standard design certification.

(b) Subpart E of this part governs the NRC staff review and approval of a final standard design. Subpart E may be used independently of the provisions in this subpart.

(c) Subpart F of this part governs the issuance of licenses to manufacture nuclear power reactors to be installed and operated at sites not identified in the manufacturing license application. Subpart F may be used independently of the provisions in this subpart. However, an applicant for a manufacturing license under subpart F may reference a design certification.

§ 52.45 Filing of applications.

(a) An application for design certification may be filed notwithstanding the fact that an application for a construction permit, combined license, or manufacturing license for such a facility has not been filed.

(b) The application must comply with the applicable filing requirements of §§ 52.3 and §§ 2.811 through 2.819 of this chapter.

(c) The fees associated with the review of an application for the initial issuance or renewal of a standard design certification are set forth in 10 CFR part 170.

§ 52.46 Contents of applications; general information.

The application must contain all of the information required by 10 CFR 50.33(a) through (c) and (j).

§ 52.47 Contents of applications; technical information.

The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, before design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination.

(a) The application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:

(1) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters;

(2) A description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. It is expected that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations. Such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent. The following power reactor design characteristics will be taken into consideration by the Commission:

(i) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(ii) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(iii) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials; and

(iv) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release[3] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable postulated site parameters, including site meteorology, to evaluate the offsite radiological consequences. The evaluation must determine that:

(A) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[4] total effective dose equivalent (TEDE);

(B) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE;

(3) The design of the facility including:

[3] The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

[4] A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. This dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accident.

(i) The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units;

(ii) The design bases and the relation of the design bases to the principal design criteria;

(iii) Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with an adequate margin for safety;

(4) An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of emergency core cooling system (ECCS) cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter;

(5) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20 of this chapter;

(6) The information required by § 20.1406 of this chapter;

(7) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter;

(8) The information necessary to demonstrate compliance with any technically relevant portions of the Three

Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v);

(9) For applications for light-water-cooled nuclear power plants, an evaluation of the standard plant design against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP is not a substitute for the regulations, and compliance is not a requirement.

(10) The information with respect to the design of equipment to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations described in 10 CFR 50.34a(e);

(11) Proposed technical specifications prepared in accordance with the requirements of §§ 50.36 and 50.36a of this chapter;

(12) An analysis and description of the equipment and systems for combustible gas control as required by 10 CFR 50.44;

(13) The list of electric equipment important to safety that is required by 10 CFR 50.49(d);

(14) A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in 10 CFR 50.60 and 50.61;

(15) Information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transients without scram events in § 50.62;

(16) A coping analysis, and any design features necessary to address station blackout, as required by 10 CFR 50.63;

(17) Information demonstrating how the applicant will comply with requirements for criticality accidents in § 50.68(b)(2)–(b)(4);

(18) A description and analysis of the fire protection design features for the standard plant necessary to comply with 10 CFR part 50, appendix A, GDC 3, and § 50.48 of this chapter;

(19) A description of the quality assurance program applied to the design of the structures, systems, and components of the facility. Appendix B to 10 CFR part 50, ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,'' sets forth the requirements for quality assurance programs for nuclear power plants. The description of the quality assurance program for a nuclear power plant shall include a discussion of how the applicable requirements of appendix B to 10 CFR part 50 were satisfied;

(20) The information necessary to demonstrate that the standard plant complies with the earthquake engineering criteria in 10 CFR part 50, appendix S;

(21) Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority generic safety issues which are identified in the version of NUREG–0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design;

(22) The information necessary to demonstrate how operating experience insights have been incorporated into the plant design;

(23) For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass;

(24) A representative conceptual design for those portions of the plant for which the application does not seek certification, to aid the NRC in its review of the FSAR and to permit assessment of the adequacy of the interface requirements in paragraph (a)(25) of this section;

(25) The interface requirements to be met by those portions of the plant for which the application does not seek

certification. These requirements must be sufficiently detailed to allow completion of the FSAR;

(26) Justification that compliance with the interface requirements of paragraph (a)(25) of this section is verifiable through inspections, tests, or analyses. The method to be used for verification of interface requirements must be included as part of the proposed ITAAC required by paragraph (b)(1) of this section; and

(27) A description of the design-specific probabilistic risk assessment (PRA) and its results.

(b) The application must also contain:

(1) The proposed inspections, tests, analyses, and acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Act, and the Commission's rules and regulations; and

(2) An environmental report as required by 10 CFR 51.55.

(c) This paragraph applies, according to its provisions, to particular applications:

(1) An application for certification of a nuclear power reactor design that is an evolutionary change from light-water reactor designs of plants that have been licensed and in commercial operation before April 18, 1989, must provide an essentially complete nuclear power plant design except for site-specific elements such as the service water intake structure and the ultimate heat sink;

(2) An application for certification of a nuclear power reactor design that differs significantly from the light-water reactor designs described in paragraph (c)(1) of this section or uses simplified, inherent, passive, or other innovative means to accomplish its safety functions must provide an essentially complete nuclear power reactor design except for site-specific elements such as the service water intake structure and the ultimate heat sink, and must meet the requirements of 10 CFR 50.43(e); and

(3) An application for certification of a modular nuclear power reactor design must describe and analyze the possible operating configurations of the reactor modules with common systems, interface requirements, and system interactions. The final safety analysis must also account for differences among the configurations, including any restrictions that will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating.

EFFECTIVE DATE NOTE: At 73 FR 63571, Oct. 24, 2008, § 52.47 was amended by adding paragraph (d), effective Feb. 23, 2009. For the convenience of the user, the added text is set forth as follows:

§ 52.47 Contents of applications; technical information.

* * * * *

(d) Each applicant for a standard design certification under this part shall protect Safeguards Information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.

§ 52.48 Standards for review of applications.

Applications filed under this subpart will be reviewed for compliance with the standards set out in 10 CFR parts 20, 50 and its appendices, 51, 73, and 100.

§ 52.51 Administrative review of applications.

(a) A standard design certification is a rule that will be issued in accordance with the provisions of subpart H of 10 CFR part 2, as supplemented by the provisions of this section. The Commission shall initiate the rulemaking after an application has been filed under § 52.45 and shall specify the procedures to be used for the rulemaking. The notice of proposed rulemaking published in the FEDERAL REGISTER must provide an opportunity for the submission of comments on the proposed design certification rule. If, at the time a proposed design certification rule is published in the FEDERAL REGISTER under this paragraph (a), the Commission decides that a legislative hearing should be held, the information required by 10 CFR 2.1502(c) must be included in the

FEDERAL REGISTER document for the proposed design certification.

(b) Following the submission of comments on the proposed design certification rule, the Commission may, at its discretion, hold a legislative hearing under the procedures in subpart O of part 2 of this chapter. The Commission shall publish a document in the FEDERAL REGISTER of its decision to hold a legislative hearing. The document shall contain the information specified in paragraph (c) of this section, and specify whether the Commission or a presiding officer will conduct the legislative hearing.

(c) Notwithstanding anything in 10 CFR 2.390 to the contrary, proprietary information will be protected in the same manner and to the same extent as proprietary information submitted in connection with applications for licenses, provided that the design certification shall be published in Chapter I of this title.

§ 52.53 Referral to the Advisory Committee on Reactor Safeguards (ACRS).

The Commission shall refer a copy of the application to the ACRS. The ACRS shall report on those portions of the application which concern safety.

§ 52.54 Issuance of standard design certification.

(a) After conducting a rulemaking proceeding under § 52.51 on an application for a standard design certification and receiving the report to be submitted by the Advisory Committee on Reactor Safeguards under § 52.53, the Commission may issue a standard design certification in the form of a rule for the design which is the subject of the application, if the Commission determines that:

(1) The application meets the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations;

(2) Notifications, if any, to other agencies or bodies have been duly made;

(3) There is reasonable assurance that the standard design conforms with the provisions of the Act, and the Commission's regulations;

(4) The applicant is technically qualified;

(5) The proposed inspections, tests, analyses, and acceptance criteria are necessary and sufficient, within the scope of the standard design, to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will be operated in accordance with the design certification, the provisions of the Act, and the Commission's regulations;

(6) Issuance of the standard design certification will not be inimical to the common defense and security or to the health and safety of the public;

(7) The findings required by subpart A of part 51 of this chapter have been made; and

(8) The applicant has implemented the quality assurance program described or referenced in the safety analysis report.

(b) The design certification rule must specify the site parameters, design characteristics, and any additional requirements and restrictions of the design certification rule.

(c) After the Commission has adopted a final design certification rule, the applicant shall not permit any individual to have access to or any facility to possess restricted data or classified National Security Information until the individual and/or facility has been approved for access under the provisions of 10 CFR parts 25 and/or 95, as applicable.

§ 52.55 Duration of certification.

(a) Except as provided in paragraph (b) of this section, a standard design certification issued under this subpart is valid for 15 years from the date of issuance.

(b) A standard design certification continues to be valid beyond the date of expiration in any proceeding on an application for a combined license or an operating license that references the standard design certification and is docketed either before the date of expiration of the certification, or, if a timely application for renewal of the certification has been filed, before the Commission has determined whether to renew the certification. A design certification also continues to be valid beyond the date of expiration in any hearing held under § 52.103 before operation begins under a combined license that references the design certification.

(c) An applicant for a construction permit or a combined license may, at its own risk, reference in its application a design for which a design certification application has been docketed but not granted.

§ 52.57 Application for renewal.

(a) Not less than 12 nor more than 36 months before the expiration of the initial 15-year period, or any later renewal period, any person may apply for renewal of the certification. An application for renewal must contain all information necessary to bring up to date the information and data contained in the previous application. The Commission will require, before renewal of certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if this information is necessary for the Commission to make its safety determination. Notice and comment procedures must be used for a rulemaking proceeding on the application for renewal. The Commission, in its discretion, may require the use of additional procedures in individual renewal proceedings.

(b) A design certification, either original or renewed, for which a timely application for renewal has been filed remains in effect until the Commission has determined whether to renew the certification. If the certification is not renewed, it continues to be valid in certain proceedings, in accordance with the provisions of § 52.55.

(c) The Commission shall refer a copy of the application for renewal to the Advisory Committee on Reactor Safeguards (ACRS). The ACRS shall report on those portions of the application which concern safety and shall apply the criteria set forth in § 52.59.

§ 52.59 Criteria for renewal.

(a) The Commission shall issue a rule granting the renewal if the design, either as originally certified or as modified during the rulemaking on the renewal, complies with the Atomic Energy Act and the Commission's regulations applicable and in effect at the time the certification was issued.

(b) The Commission may impose other requirements if it determines that:

(1) They are necessary for adequate protection to public health and safety or common defense and security;

(2) They are necessary for compliance with the Commission's regulations and orders applicable and in effect at the time the design certification was issued; or

(3) There is a substantial increase in overall protection of the public health and safety or the common defense and security to be derived from the new requirements, and the direct and indirect costs of implementing those requirements are justified in view of this increased protection.

(c) In addition, the applicant for renewal may request an amendment to the design certification. The Commission shall grant the amendment request if it determines that the amendment will comply with the Atomic Energy Act and the Commission's regulations in effect at the time of renewal. If the amendment request entails such an extensive change to the design certification that an essentially new standard design is being proposed, an application for a design certification must be filed in accordance with this subpart.

(d) Denial of renewal does not bar the applicant, or another applicant, from filing a new application for certification of the design, which proposes design changes that correct the deficiencies cited in the denial of the renewal.

§ 52.61 Duration of renewal.

Each renewal of certification for a standard design will be for not less than 10, nor more than 15 years.

§ 52.63 Finality of standard design certifications.

(a)(1) Notwithstanding any provision in 10 CFR 50.109, while a standard design certification rule is in effect under §§ 52.55 or 52.61, the Commission may not modify, rescind, or impose new requirements on the certification information, whether on its own motion, or in response to a petition from any person, unless the Commission determines in a rulemaking that the change:

(i) Is necessary either to bring the certification information or the referencing plants into compliance with the Commission's regulations applicable and in effect at the time the certification was issued;

(ii) Is necessary to provide adequate protection of the public health and safety or the common defense and security;

(iii) Reduces unnecessary regulatory burden and maintains protection to public health and safety and the common defense and security;

(iv) Provides the detailed design information to be verified under those inspections, tests, analyses, and acceptance criteria (ITAAC) which are directed at certification information (*i.e.*, design acceptance criteria);

(v) Is necessary to correct material errors in the certification information;

(vi) Substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; or

(vii) Contributes to increased standardization of the certification information.

(2)(i) In a rulemaking under § 52.63(a)(1), except for § 52.63(a)(1)(ii), the Commission will give consideration to whether the benefits justify the costs for plants that are already licensed or for which an application for a permit or license is under consideration.

(ii) The rulemaking procedures for changes under § 52.63(a)(1) must provide for notice and opportunity for public comment.

(3) Any modification the NRC imposes on a design certification rule under paragraph (a)(1) of this section will be applied to all plants referencing the certified design, except those to which the modification has been rendered technically irrelevant by action taken under paragraphs (a)(4) or (b)(1) of this section.

(4) The Commission may not impose new requirements by plant-specific order on any part of the design of a specific plant referencing the design certification rule if that part was approved in the design certification while a design certification rule is in effect under § 52.55 or § 52.61, unless:

(i) A modification is necessary to secure compliance with the Commission's regulations applicable and in effect at the time the certification was issued, or to assure adequate protection of the public health and safety or the common defense and security; and

(ii) Special circumstances as defined in 10 CFR 52.7 are present. In addition to the factors listed in § 52.7, the Commission shall consider whether the special circumstances which § 52.7 requires to be present outweigh any decrease in safety that may result from the reduction in standardization caused by the plant-specific order.

(5) Except as provided in 10 CFR 2.335, in making the findings required for issuance of a combined license, construction permit, operating license, or manufacturing license, or for any hearing under § 52.103, the Commission shall treat as resolved those matters resolved in connection with the issuance or renewal of a design certification rule.

(b)(1) An applicant or licensee who references a design certification rule may request an exemption from one or more elements of the certification information. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of § 52.7. In addition to the factors listed in § 52.7, the Commission shall consider whether the special circumstances that § 52.7 requires to be present outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. The granting of an exemption on request of an applicant is subject to litigation in the same manner as other issues in the operating license or combined license hearing.

(2) Subject to § 50.59 of this chapter, a licensee who references a design certification rule may make departures from the design of the nuclear power facility, without prior Commission approval, unless the proposed departure

involves a change to the design as described in the rule certifying the design. The licensee shall maintain records of all departures from the facility and these records must be maintained and available for audit until the date of termination of the license.

(c) The Commission will require, before granting a construction permit, combined license, operating license, or manufacturing license which references a design certification rule, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determinations, including the determination that the application is consistent with the certification information. This information may be acquired by appropriate arrangements with the design certification applicant.

Subpart C—Combined Licenses

§ 52.71 Scope of subpart.

This subpart sets out the requirements and procedures applicable to Commission issuance of combined licenses for nuclear power facilities.

§ 52.73 Relationship to other subparts.

(a) An application for a combined license under this subpart may, but need not, reference a standard design certification, standard design approval, or manufacturing license issued under subparts B, E, or F of this part, respectively, or an early site permit issued under subpart A of this part. In the absence of a demonstration that an entity other than the one originally sponsoring and obtaining a design certification is qualified to supply a design, the Commission will entertain an application for a combined license that references a standard design certification issued under subpart B of this part only if the entity that sponsored and obtained the certification supplies the design for the applicant's use.

(b) The Commission will require, before granting a combined license that references a standard design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determinations, including the determination that the application is consistent with the certification information.

§ 52.75 Filing of applications.

(a) Any person except one excluded by § 50.38 of this chapter may file an application for a combined license for a nuclear power facility with the Director, Office of New Reactors or Director, Office of Nuclear Reactor Regulation, as appropriate.

(b) The application must comply with the applicable filing requirements of §§ 52.3 and 50.30 of this chapter.

(c) The fees associated with the filing and review of the application are set forth in 10 CFR part 170.

[72 FR 49517, Aug. 28, 2007, as amended at 73 FR 5724, Jan. 31, 2008]

§ 52.77 Contents of applications; general information.

The application must contain all of the information required by 10 CFR 50.33.

§ 52.79 Contents of applications; technical information in final safety analysis report.

(a) The application must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components of the facility as a whole. The final safety analysis report shall include the following information, at a level of information sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved by the Commission before issuance of a combined license:

(1)(i) The boundaries of the site;

(ii) The proposed general location of each facility on the site;

(iii) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated;

(iv) The location and description of any nearby industrial, military, or transportation facilities and routes;

(v) The existing and projected future population profile of the area surrounding the site;

(vi) A description and safety assessment of the site on which the facility is to be located. The assessment must contain an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors identified in paragraphs (a)(1)(vi)(A) and (a)(1)(vi)(B) of this section. In performing this assessment, an applicant shall assume a fission product release [5] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:

(A) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem [6] total effective dose equivalent (TEDE).

(B) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE; and

(2) A description and analysis of the structures, systems, and components of the facility with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. It is expected that reactors will reflect through their design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The descriptions shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations. Items such as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent. The following power reactor design characteristics and proposed operation will be taken into consideration by the Commission:

(i) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(ii) The extent to which generally accepted engineering standards are applied to the design of the reactor;

[5] The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

[6] A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accident.

(iii) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(iv) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release[7] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated;

(3) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20 of this chapter;

(4) The design of the facility including:

(i) The principal design criteria for the facility. Appendix A to part 50 of this chapter, ''General Design Criteria for Nuclear Power Plants,'' establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units;

(ii) The design bases and the relation of the design bases to the principal design criteria;

(iii) Information relative to materials of construction, arrangement, and dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with adequate margin for safety.

(5) An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter;

(6) A description and analysis of the fire protection design features for the reactor necessary to comply with 10 CFR part 50, appendix A, GDC 3, and § 50.48 of this chapter;

(7) A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in §§ 50.60 and 50.61(b)(1) and (b)(2) of this chapter;

(8) An analysis and description of the equipment and systems for combustible gas control as required by § 50.44 of this chapter;

(9) The coping analyses, and any design features necessary to address station blackout, as described in § 50.63 of this chapter;

(10) A description of the program, and its implementation, required by § 50.49(a) of this chapter for the environmental qualification of electric equipment important to safety and the list of electric equipment important to safety that is required by 10 CFR 50.49(d);

(11) A description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants in accordance with 50.55a of this chapter;

[7] The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

(12) A description of the primary containment leakage rate testing program, and its implementation, necessary to ensure that the containment meets the requirements of appendix J to 10 CFR part 50;

(13) A description of the reactor vessel material surveillance program required by appendix H to 10 CFR part 50 and its implementation;

(14) A description of the operator training program, and its implementation, necessary to meet the requirements of 10 CFR part 55;

(15) A description of the program, and its implementation, for monitoring the effectiveness of maintenance necessary to meet the requirements of § 50.65 of this chapter;

(16)(i) The information with respect to the design of equipment to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, as described in § 50.34a(d) of this chapter;

(ii) A description of the process and effluent monitoring and sampling program required by appendix I to 10 CFR part 50 and its implementation.

(17) The information with respect to compliance with technically relevant positions of the Three Mile Island requirements in § 50.34(f) of this chapter, with the exception of §§ 50.34(f)(1)(xii), (f)(2)(ix), and (f)(3)(v);

(18) If the applicant seeks to use risk-informed treatment of SSCs in accordance with § 50.69 of this chapter, the information required by § 50.69(b)(2) of this chapter;

(19) Information necessary to demonstrate that the plant complies with the earthquake engineering criteria in 10 CFR part 50, appendix S;

(20) Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority generic safety issues which are identified in the version of NUREG–0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design;

(21) Emergency plans complying with the requirements of § 50.47 of this chapter, and 10 CFR part 50, appendix E;

(22)(i) All emergency plan certifications that have been obtained from the State and local governmental agencies with emergency planning responsibilities must state that:

(A) The proposed emergency plans are practicable;

(B) These agencies are committed to participating in any further development of the plans, including any required field demonstrations; and

(C) These agencies are committed to executing their responsibilities under the plans in the event of an emergency;

(ii) If certifications cannot be obtained after sustained, good faith efforts by the applicant, then the application must contain information, including a utility plan, sufficient to show that the proposed plans provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site.

(23) [Reserved]

(24) If the application is for a nuclear power reactor design which differs significantly from light-water reactor designs that were licensed before 1997 or use simplified, inherent, passive, or other innovative means to accomplish their safety functions, the application must describe how the design meets the requirements in § 50.43(e) of this chapter;

(25) A description of the quality assurance program, applied to the design, and to be applied to the fabrication, construction, and testing, of the structures, systems, and components of the facility. Appendix B to 10 CFR part 50 sets forth the requirements for quality assurance programs for nuclear power plants. The description of the quality assurance program for a nuclear power plant must include a discussion of how the applicable requirements of appendix B to 10 CFR part 50 have been and will be satisfied, including a discussion of how the quality assurance program will be implemented;

(26) The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements for operation;

(27) Managerial and administrative controls to be used to assure safe operation. Appendix B to 10 CFR part 50 sets forth the requirements for these controls for nuclear power plants. The information on the controls to be used

for a nuclear power plant shall include a discussion of how the applicable requirements of appendix B to 10 CFR part 50 will be satisfied;

(28) Plans for preoperational testing and initial operations;

(29)(i) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components;

(ii) Plans for coping with emergencies, other than the plans required by § 52.79(a)(21);

(30) Proposed technical specifications prepared in accordance with the requirements of §§ 50.36 and 50.36a of this chapter;

(31) For nuclear power plants to be operated on multi-unit sites, an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multi-unit sites;

(32) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter;

(33) A description of the training program required by § 50.120 of this chapter and its implementation;

(34) A description and plans for implementation of an operator requalification program. The operator requalification program must as a minimum, meet the requirements for those programs contained in § 55.59 of this chapter;

(35)(i) A physical security plan, describing how the applicant will meet the requirements of 10 CFR part 73 (and 10 CFR part 11, if applicable, including the identification and description of jobs as required by § 11.11(a) of this chapter, at the proposed facility). The plan must list tests, inspections, audits, and other means to be used to demonstrate compliance with the requirements of 10 CFR parts 11 and 73, if applicable;

(ii) A description of the implementation of the physical security plan;

(36)(i) A safeguards contingency plan in accordance with the criteria set forth in appendix C to 10 CFR part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for this type of license shall include the information contained in the applicant's safeguards contingency plan.[8] (Implementing procedures required for this plan need not be submitted for approval.)

(ii) A training and qualification plan in accordance with the criteria set forth in appendix B to 10 CFR part 73.

(iii) A description of the implementation of the safeguards contingency plan and the training and qualification plan;

(iv) Each applicant who prepares a physical security plan, a safeguards contingency plan, or a guard qualification and training plan, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter, as appropriate.

(37) The information necessary to demonstrate how operating experience insights have been incorporated into the plant design;

(38) For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass;

(39) A description of the radiation protection program required by § 20.1101 of this chapter and its implementation.

(40) A description of the fire protection program required by § 50.48 of this chapter and its implementation.

(41) For applications for light-water-cooled nuclear power plant combined licenses, an evaluation of the facility against the Standard Review Plan

[8] A physical security plan that contains all the information required in both § 73.55 of this chapter and appendix C to 10 CFR part 73 satisfies the requirement for a contingency plan.

(SRP) revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP is not a substitute for the regulations, and compliance is not a requirement;

(42) Information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transients without scram (ATWS) events in § 50.62 of this chapter;

(43) Information demonstrating how the applicant will comply with requirements for criticality accidents in § 50.68 of this chapter;

(44) A description of the fitness-for-duty program required by 10 CFR part 26 and its implementation.

(45) The information required by § 20.1406 of this chapter.

(46) A description of the plant-specific probabilistic risk assessment (PRA) and its results.

(b) If the combined license application references an early site permit, then the following requirements apply:

(1) The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the early site permit, *provided, however*, that the final safety analysis report must either include or incorporate by reference the early site permit site safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the design of the facility falls within the site characteristics and design parameters specified in the early site permit.

(2) If the final safety analysis report does not demonstrate that design of the facility falls within the site characteristics and design parameters, the application shall include a request for a variance that complies with the requirements of §§ 52.39 and 52.93.

(3) The final safety analysis report must demonstrate that all terms and conditions that have been included in the early site permit, other than those imposed under § 50.36b, will be satisfied by the date of issuance of the combined license. Any terms or conditions of the early site permit that could not be met by the time of issuance of the combined license, must be set forth as terms or conditions of the combined license.

(4) If the early site permit approves complete and integrated emergency plans, or major features of emergency plans, then the final safety analysis report must include any new or additional information that updates and corrects the information that was provided under § 52.17(b), and discuss whether the new or additional information materially changes the bases for compliance with the applicable requirements. The application must identify changes to the emergency plans or major features of emergency plans that have been incorporated into the proposed facility emergency plans and that constitute or would constitute a decrease in effectiveness under § 50.54(q) of this chapter.

(5) If complete and integrated emergency plans are approved as part of the early site permit, new certifications meeting the requirements of paragraph (a)(22) of this section are not required.

(c) If the combined license application references a standard design approval, then the following requirements apply:

(1) The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the design approval, *provided, however*, that the final safety analysis report must either include or incorporate by reference the standard design approval final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the characteristics of the site fall within the site parameters specified in the design approval. In addition, the plant-specific PRA information must use the PRA information for

the design approval and must be updated to account for site-specific design information and any design changes or departures.

(2) The final safety analysis report must demonstrate that all terms and conditions that have been included in the final design approval will be satisfied by the date of issuance of the combined license.

(d) If the combined license application references a standard design certification, then the following requirements apply:

(1) The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the design certification, *provided, however*, that the final safety analysis report must either include or incorporate by reference the standard design certification final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the site characteristics fall within the site parameters specified in the design certification. In addition, the plant-specific PRA information must use the PRA information for the design certification and must be updated to account for site-specific design information and any design changes or departures.

(2) The final safety analysis report must demonstrate that the interface requirements established for the design under § 52.47 have been met.

(3) The final safety analysis report must demonstrate that all requirements and restrictions set forth in the referenced design certification rule, other than those imposed under § 50.36b, must be satisfied by the date of issuance of the combined license. Any requirements and restrictions set forth in the referenced design certification rule that could not be satisfied by the time of issuance of the combined license, must be set forth as terms or conditions of the combined license.

(e) If the combined license application references the use of one or more manufactured nuclear power reactors licensed under subpart F of this part, then the following requirements apply:

(1) The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the manufacturing license, *provided, however*, that the final safety analysis report must either include or incorporate by reference the manufacturing license final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the site characteristics fall within the site parameters specified in the manufacturing license. In addition, the plant-specific PRA information must use the PRA information for the manufactured reactor and must be updated to account for site-specific design information and any design changes or departures.

(2) The final safety analysis report must demonstrate that the interface requirements established for the design have been met.

(3) The final safety analysis report must demonstrate that all terms and conditions that have been included in the manufacturing license, other than those imposed under § 50.36b, will be satisfied by the date of issuance of the combined license. Any terms or conditions of the manufacturing license that could not be met by the time of issuance of the combined license, must be set forth as terms or conditions of the combined license.

EFFECTIVE DATE NOTE: At 73 FR 63571, Oct. 24, 2008, § 52.79 was amended by adding paragraph (f), effective Feb. 23, 2009. For the convenience of the user, the added text is set forth as follows:

§ 52.79 Contents of application; technical information in final safety analysis report.

* * * * *

(f) Each applicant for a combined license under this subpart shall protect Safeguards Information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.

§ 52.80 Contents of applications; additional technical information.

The application must contain:

(a) The proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will be operated in conformity with the combined license, the provisions of the Act, and the Commission's rules and regulations.

(1) If the application references an early site permit with ITAAC, the early site permit ITAAC must apply to those aspects of the combined license which are approved in the early site permit.

(2) If the application references a standard design certification, the ITAAC contained in the certified design must apply to those portions of the facility design which are approved in the design certification.

(3) If the application references an early site permit with ITAAC or a standard design certification or both, the application may include a notification that a required inspection, test, or analysis in the ITAAC has been successfully completed and that the corresponding acceptance criterion has been met. The FEDERAL REGISTER notification required by § 52.85 must indicate that the application includes this notification.

(b) An environmental report, either in accordance with 10 CFR 51.50(c) if a limited work authorization under 10 CFR 50.10 is not requested in conjunction with the combined license application, or in accordance with §§ 51.49 and 51.50(c) of this chapter if a limited work authorization is requested in conjunction with the combined license application.

(c) If the applicant wishes to request that a limited work authorization under 10 CFR 50.10 be issued before issuance of the combined license, the application must include the information otherwise required by 10 CFR 50.10, in accordance with either 10 CFR 2.101(a)(1) through (a)(4), or 10 CFR 2.101(a)(9).

[72 FR 49517, Aug. 28, 2007, as amended at 72 FR 57447, Oct. 9, 2007]

§ 52.81 Standards for review of applications.

Applications filed under this subpart will be reviewed according to the standards set out in 10 CFR parts 20, 50, 51, 54, 55, 73, 100, and 140.

§ 52.83 Finality of referenced NRC approvals; partial initial decision on site suitability.

(a) If the application for a combined license under this subpart references an early site permit, design certification rule, standard design approval, or manufacturing license, the scope and nature of matters resolved for the application and any combined license issued are governed by the relevant provisions addressing finality, including §§ 52.39, 52.63, 52.98, 52.145, and 52.171.

(b) While a partial decision on site suitability is in effect under 10 CFR 2.617(b)(2), the scope and nature of matters resolved in the proceeding are governed by the finality provisions in 10 CFR 2.629.

§ 52.85 Administrative review of applications; hearings.

A proceeding on a combined license is subject to all applicable procedural requirements contained in 10 CFR part 2, including the requirements for docketing (§ 2.101 of this chapter) and issuance of a notice of hearing (§ 2.104 of this chapter). If an applicant requests a Commission finding on certain ITAAC with the issuance of the combined license, then those ITAAC will be identified in the notice of hearing. All hearings on combined licenses are governed by the procedures contained in 10 CFR part 2.

§ 52.87 Referral to the Advisory Committee on Reactor Safeguards (ACRS).

The Commission shall refer a copy of the application to the ACRS. The ACRS shall report on those portions of the application that concern safety and shall apply the standards referenced in § 52.81, in accordance with the finality provisions in § 52.83.

§ 52.89 [Reserved]

§ 52.91 Authorization to conduct limited work authorization activities.

(a) If the application does not reference an early site permit which authorizes the holder to perform the activities under 10 CFR 50.10(d), the applicant may not perform those activities without obtaining the separate authorization required by 10 CFR 50.10(d). Authorization may be granted only after the presiding officer in the proceeding on the application has made the findings and determination required by 10 CFR 50.10(e), and the Director of New Reactors or the Director of Nuclear Reactor Regulation makes the determination required by 10 CFR 50.10(e).

(b) If, after an applicant has performed the activities permitted by paragraph (a) of this section, the application for the combined license is withdrawn or denied, then the applicant shall implement the approved site redress plan.

[72 FR 57447, Oct. 9, 2007]

§ 52.93 Exemptions and variances.

(a) Applicants for a combined license under this subpart, or any amendment to a combined license, may include in the application a request for an exemption from one or more of the Commission's regulations.

(1) If the request is for an exemption from any part of a referenced design certification rule, the Commission may grant the request if it determines that the exemption complies with any exemption provisions of the referenced design certification rule, or with § 52.63 if there are no applicable exemption provisions in the referenced design certification rule.

(2) For all other requests for exemptions, the Commission may grant a request if it determines that the exemption complies with § 52.7.

(b) An applicant for a combined license who has filed an application referencing an early site permit issued under subpart A of this part may include in the application a request for a variance from one or more site characteristics, design parameters, or terms and conditions of the permit, or from the site safety analysis report. In determining whether to grant the variance, the Commission shall apply the same technically relevant criteria as were applicable to the application for the original or renewed site permit. Once a construction permit or combined license referencing an early site permit is issued, variances from the early site permit will not be granted for that construction permit or combined license.

(c) An applicant for a combined license who has filed an application referencing a nuclear power reactor manufactured under a manufacturing license issued under subpart F of this part may include in the application a request for a departure from one or more design characteristics, site parameters, terms and conditions, or approved design of the manufactured reactor. The Commission may grant a request only if it determines that the departure will comply with the requirements of 10 CFR 52.7, and that the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the departure.

(d) Issuance of a variance under paragraph (b) or a departure under paragraph (c) of this section is subject to litigation during the combined license proceeding in the same manner as other issues material to that proceeding.

§ 52.97 Issuance of combined licenses.

(a)(1) After conducting a hearing in accordance with § 52.85 and receiving the report submitted by the ACRS, the Commission may issue a combined license if the Commission finds that:

(i) The applicable standards and requirements of the Act and the Commission's regulations have been met;

(ii) Any required notifications to other agencies or bodies have been duly made;

(iii) There is reasonable assurance that the facility will be constructed and will operate in conformity with the license, the provisions of the Act, and the Commission's regulations.

(iv) The applicant is technically and financially qualified to engage in the activities authorized; and

(v) Issuance of the license will not be inimical to the common defense and security or to the health and safety of the public; and

(vi) The findings required by subpart A of part 51 of this chapter have been made.

(2) The Commission may also find, at the time it issues the combined license, that certain acceptance criteria in one or more of the inspections, tests, analyses, and acceptance criteria (ITAAC) in a referenced early site permit or standard design certification have been met. This finding will finally resolve that those acceptance criteria have been met, those acceptance criteria will be deemed to be excluded from the combined license, and findings under § 52.103(g) with respect to those acceptance criteria are unnecessary.

(b) The Commission shall identify within the combined license the inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that, if met, are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Act, and the Commission's rules and regulations.

(c) A combined license shall contain the terms and conditions, including technical specifications, as the Commission deems necessary and appropriate.

§ 52.98 Finality of combined licenses; information requests.

(a) After issuance of a combined license, the Commission may not modify, add, or delete any term or condition of the combined license, the design of the facility, the inspections, tests, analyses, and acceptance criteria contained in the license which are not derived from a referenced standard design certification or manufacturing license, except in accordance with the provisions of § 52.103 or § 50.109 of this chapter, as applicable.

(b) If the combined license does not reference a design certification or a reactor manufactured under a subpart F of this part manufacturing license, then a licensee may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described in the final safety analysis report (as updated), and conduct tests or experiments not described in the final safety analysis report (as updated) under the applicable change processes in 10 CFR part 50 (e.g., §§ 50.54, 50.59, or 50.90 of this chapter).

(c) If the combined license references a certified design, then—

(1) Changes to or departures from information within the scope of the referenced design certification rule are subject to the applicable change processes in that rule; and

(2) Changes that are not within the scope of the referenced design certification rule are subject to the applicable change processes in 10 CFR part 50, unless they also involve changes to or noncompliance with information within the scope of the referenced design certification rule. In these cases, the applicable provisions of this section and the design certification rule apply.

(d) If the combined license references a reactor manufactured under a subpart F of this part manufacturing license, then—

(1) Changes to or departures from information within the scope of the manufactured reactor's design are subject to the change processes in § 52.171; and

(2) Changes that are not within the scope of the manufactured reactor's design are subject to the applicable change processes in 10 CFR part 50.

(e) The Commission may issue and make immediately effective any amendment to a combined license upon a determination by the Commission that the amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person. The amendment may be issued and made immediately effective in advance of the holding and completion of any required hearing. The amendment will be processed in accordance with the procedures specified in 10 CFR 50.91.

(f) Any modification to, addition to, or deletion from the terms and conditions of a combined license, including any modification to, addition to, or deletion from the inspections, tests, analyses, or related acceptance criteria contained in the license is a proposed amendment to the license. There must be an opportunity for a hearing on the amendment.

(g) Except for information sought to verify licensee compliance with the current licensing basis for that facility, information requests to the holder of a combined license must be evaluated before issuance to ensure that the burden to be imposed on the licensee is justified in view of the potential safety significance of the issue to be addressed in the requested information. Each evaluation performed by the NRC staff must be in accordance with 10 CFR 50.54(f) and must be approved by the Executive Director for Operations or his or her designee before issuance of the request.

§ 52.99 Inspection during construction.

(a) The licensee shall submit to the NRC, no later that 1 year after issuance of the combined license or at the start of construction as defined in 10 CFR 50.10(a), whichever is later, its schedule for completing the inspections, tests, or analyses in the ITAAC. The licensee shall submit updates to the ITAAC schedules every 6 months thereafter and, within 1 year of its scheduled date for initial loading of fuel, the licensee shall submit updates to the ITAAC schedule every 30 days until the final notification is provided to the NRC under paragraph (c)(1) of this section.

(b) With respect to activities subject to an ITAAC, an applicant for a combined 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 pre-operational activities, even though the NRC may not have found that any one of the prescribed acceptance criteria have been met.

(c)(1) The licensee shall notify the NRC that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met. The notification must contain sufficient information to demonstrate that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met.

(2) If the licensee has not provided, by the date 225 days before the scheduled date for initial loading of fuel, the notification required by paragraph (c)(1) of this section for all ITAAC, then the licensee shall notify the NRC that the prescribed inspections, tests, or analyses for all uncompleted ITAAC will be performed and that the prescribed acceptance criteria will be met prior to operation. The notification must be provided no later than the date 225 days before the scheduled date for initial loading of fuel, and must provide sufficient information to demonstrate that the prescribed inspections, tests, or analyses will be performed and the prescribed acceptance criteria for the uncompleted ITAAC will be met, including, but not limited to, a description of the specific procedures and analytical methods to be used for performing the prescribed inspections, tests, and analyses and determining that the prescribed acceptance criteria have been met.

(d)(1) In the event that an activity is subject to an ITAAC derived from a referenced standard design certification and the licensee has not demonstrated that the ITAAC has been met, the licensee may take corrective actions to successfully complete that ITAAC or request an exemption from the standard design certification ITAAC, as applicable. A request for an exemption must also be accompanied by a request for a license amendment under § 52.98(f).

(2) In the event that an activity is subject to an ITAAC not derived from a referenced standard design certification and the licensee has not demonstrated that the ITAAC has been met, the licensee may take corrective actions to successfully complete that ITAAC or request a license amendment under § 52.98(f).

(e) The NRC shall ensure that the prescribed inspections, tests, and analyses in the ITAAC are performed.

(1) At appropriate intervals until the last date for submission of requests for hearing under § 52.103(a), the NRC shall

publish notices in the FEDERAL REGISTER of the NRC staff's determination of the successful completion of inspections, tests, and analyses.

(2) The NRC shall make publicly available the licensee notifications under paragraph (c)(1), and, no later than the date of publication of the notice of intended operation required by § 52.103(a), make available all licensee notifications under paragraphs (c)(1) and (c)(2) of this section.

[72 FR 49517, Aug. 28, 2007, as amended at 72 FR 57447, Oct. 9, 2007]

§ 52.103 Operation under a combined license.

(a) The licensee shall notify the NRC of its scheduled date for initial loading of fuel no later than 270 days before the scheduled date and shall notify the NRC of updates to its schedule every 30 days thereafter. Not less than 180 days before the date scheduled for initial loading of fuel into a plant by a licensee that has been issued a combined license under this part, the Commission shall publish notice of intended operation in the FEDERAL REGISTER. The notice must provide that any person whose interest may be affected by operation of the plant may, within 60 days, request that the Commission hold a hearing on whether the facility as constructed complies, or on completion will comply, with the acceptance criteria in the combined license, except that a hearing shall not be granted for those ITAAC which the Commission found were met under § 52.97(a)(2).

(b) A request for hearing under paragraph (a) of this section must show, prima facie, that—

(1) One or more of the acceptance criteria of the ITAAC in the combined license have not been, or will not be, met; and

(2) The specific operational consequences of nonconformance that would be contrary to providing reasonable assurance of adequate protection of the public health and safety.

(c) The Commission, acting as the presiding officer, shall determine whether to grant or deny the request for hearing in accordance with the applicable requirements of 10 CFR 2.309. If the Commission grants the request, the Commission, acting as the presiding officer, shall determine whether during a period of interim operation there will be reasonable assurance of adequate protection to the public health and safety. The Commission's determination must consider the petitioner's *prima facie* showing and any answers thereto. If the Commission determines there is such reasonable assurance, it shall allow operation during an interim period under the combined license.

(d) The Commission, in its discretion, shall determine appropriate hearing procedures, whether informal or formal adjudicatory, for any hearing under paragraph (a) of this section, and shall state its reasons therefore.

(e) The Commission shall, to the maximum possible extent, render a decision on issues raised by the hearing request within 180 days of the publication of the notice provided by paragraph (a) of this section or by the anticipated date for initial loading of fuel into the reactor, whichever is later.

(f) A petition to modify the terms and conditions of the combined license will be processed as a request for action in accordance with 10 CFR 2.206. The petitioner shall file the petition with the Secretary of the Commission. Before the licensed activity allegedly affected by the petition (fuel loading, low power testing, etc.) commences, the Commission shall determine whether any immediate action is required. If the petition is granted, then an appropriate order will be issued. Fuel loading and operation under the combined license will not be affected by the granting of the petition unless the order is made immediately effective.

(g) The licensee shall not operate the facility until the Commission makes a finding that the acceptance criteria in the combined license are met, except for those acceptance criteria that the Commission found were met under § 52.97(a)(2). If the combined license is for a modular design, each reactor module may require a separate finding as construction proceeds.

(h) After the Commission has made the finding in paragraph (g) of this section, the ITAAC do not, by virtue of their inclusion in the combined license, constitute regulatory requirements either for licensees or for renewal of the license; except for the specific ITAAC for which the Commission has granted a hearing under paragraph (a) of this section, all ITAAC expire upon final Commission action in the proceeding. However, subsequent changes to the facility or procedures described in the final safety analysis report (as updated) must comply with the requirements in §§52.98(e) or (f), as applicable.

§52.104 Duration of combined license.

A combined license is issued for a specified period not to exceed 40 years from the date on which the Commission makes a finding that acceptance criteria are met under §52.103(g) or allowing operation during an interim period under the combined license under §52.103(c).

§52.105 Transfer of combined license.

A combined license may be transferred in accordance with §50.80 of this chapter.

§52.107 Application for renewal.

The filing of an application for a renewed license must be in accordance with 10 CFR part 54.

§52.109 Continuation of combined license.

Each combined license for a facility that has permanently ceased operations, continues in effect beyond the expiration date to authorize ownership and possession of the production or utilization facility, until the Commission notifies the licensee in writing that the license is terminated. During this period of continued effectiveness the licensee shall—

(1) Take actions necessary to decommission and decontaminate the facility and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

(2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC's regulations and the provisions of the combined license for the facility.

§52.110 Termination of license.

(a)(1) When a licensee has determined to permanently cease operations the licensee shall, within 30 days, submit a written certification to the NRC, consistent with the requirements of §52.3(b)(8);

(2) Once fuel has been permanently removed from the reactor vessel, the licensee shall submit a written certification to the NRC that meets the requirements of §52.3(b)(9); and

(3) For licensees whose licenses have been permanently modified to allow possession but not operation of the facility, before September 27, 2007, the certification required in paragraph (a)(1) of this section shall be deemed to have been submitted.

(b) Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR part 52 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel.

(c) Decommissioning will be completed within 60 years of permanent cessation of operations. Completion of decommissioning beyond 60 years will be approved by the Commission only when necessary to protect public health and safety. Factors that will be considered by the Commission in evaluating an alternative that provides for completion of decommissioning beyond 60 years of permanent cessation of operations include unavailability of waste disposal capacity and other site-specific factors affecting the licensee's capability to carry out decommissioning, including presence of other nuclear facilities at the site.

(d)(1) Before or within 2 years following permanent cessation of operations, the licensee shall submit a post-shutdown decommissioning activities report (PSDAR) to the NRC, and a copy to the affected State(s). The report must include a description of the planned decommissioning activities along with a schedule for their accomplishment, an estimate of expected costs, and a discussion that provides the reasons for concluding that the environmental impacts associated with site-specific decommissioning activities will be bounded by appropriate previously issued environmental impact statements.

(2) The NRC shall notice receipt of the PSDAR and make the PSDAR available for public comment. The NRC shall also schedule a public meeting in the vicinity of the licensee's facility upon receipt of the PSDAR. The NRC shall publish a document in the FEDERAL REGISTER and in a forum, such as local newspapers, that is readily accessible to individuals in the vicinity of the site, announcing the date, time and location of the meeting, along with a brief description of the purpose of the meeting.

(e) Licensees shall not perform any major decommissioning activities, as defined in § 50.2 of this chapter, until 90 days after the NRC has received the licensee's PSDAR submittal and until certifications of permanent cessation of operations and permanent removal of fuel from the reactor vessel, as required under § 52.110(a)(1), have been submitted.

(f) Licensees shall not perform any decommissioning activities, as defined in § 52.1, that—

(1) Foreclose release of the site for possible unrestricted use;

(2) Result in significant environmental impacts not previously reviewed; or

(3) Result in there no longer being reasonable assurance that adequate funds will be available for decommissioning.

(g) In taking actions permitted under § 50.59 of this chapter following submittal of the PSDAR, the licensee shall notify the NRC in writing and send a copy to the affected State(s), before performing any decommissioning activity inconsistent with, or making any significant schedule change from, those actions and schedules described in the PSDAR, including changes that significantly increase the decommissioning cost.

(h)(1) Decommissioning trust funds may be used by licensees if—

(i) The withdrawals are for expenses for legitimate decommissioning activities consistent with the definition of decommissioning in § 52.1;

(ii) The expenditure would not reduce the value of the decommissioning trust below an amount necessary to place and maintain the reactor in a safe storage condition if unforeseen conditions or expenses arise and;

(iii) The withdrawals would not inhibit the ability of the licensee to complete funding of any shortfalls in the decommissioning trust needed to ensure the availability of funds to ultimately release the site and terminate the license.

(2) Initially, 3 percent of the generic amount specified in § 50.75 of this chapter may be used for decommissioning planning. For licensees that have submitted the certifications required under § 52.110(a) and commencing 90 days after the NRC has received the PSDAR, an additional 20 percent may be used. A site-specific decommissioning cost estimate must be submitted to the NRC before the licensee may use any funding in excess of these amounts.

(3) Within 2 years following permanent cessation of operations, if not already submitted, the licensee shall submit a site-specific decommissioning cost estimate.

(4) For decommissioning activities that delay completion of decommissioning by including a period of storage or surveillance, the licensee shall provide a means of adjusting cost estimates and associated funding levels over the storage or surveillance period.

(i) All power reactor licensees must submit an application for termination of license. The application for termination of license must be accompanied or preceded by a license termination plan to be submitted for NRC approval.

(1) The license termination plan must be a supplement to the FSAR or equivalent and must be submitted at least 2 years before termination of the license date.

(2) The license termination plan must include—

(i) A site characterization;

(ii) Identification of remaining dismantlement activities;

(iii) Plans for site remediation;

(iv) Detailed plans for the final radiation survey;

(v) A description of the end use of the site, if restricted;

(vi) An updated site-specific estimate of remaining decommissioning costs;

(vii) A supplement to the environmental report, under § 51.53 of this chapter, describing any new information or significant environmental change associated with the licensee's proposed termination activities; and

(viii) Identification of parts, if any, of the facility or site that were released for use before approval of the license termination plan.

(3) The NRC shall notice receipt of the license termination plan and make the license termination plan available for public comment. The NRC shall also schedule a public meeting in the vicinity of the licensee's facility upon receipt of the license termination plan. The NRC shall publish a document in the FEDERAL REGISTER and in a forum, such as local newspapers, which is readily accessible to individuals in the vicinity of the site, announcing the date, time and location of the meeting, along with a brief description of the purpose of the meeting.

(j) If the license termination plan demonstrates that the remainder of decommissioning activities will be performed in accordance with the regulations in this chapter, will not be inimical to the common defense and security or to the health and safety of the public, and will not have a significant effect on the quality of the environment and after notice to interested persons, the Commission shall approve the plan, by license amendment, subject to terms and conditions as it deems appropriate and necessary and authorize implementation of the license termination plan.

(k) The Commission shall terminate the license if it determines that—

(1) The remaining dismantlement has been performed in accordance with the approved license termination plan; and

(2) The final radiation survey and associated documentation, including an assessment of dose contributions associated with parts released for use before approval of the license termination plan, demonstrate that the facility and site have met the criteria for decommissioning in subpart E to 10 CFR part 20.

(l) For a facility that has permanently ceased operation before the expiration of its license, the collection period for any shortfall of funds will be determined, upon application by the licensee, on a case-by-case basis taking into account the specific financial situation of each licensee.

Subpart D [Reserved]

Subpart E—Standard Design Approvals

§ 52.131 Scope of subpart.

This subpart sets out procedures for the filing, NRC staff review, and referral to the Advisory Committee on Reactor Safeguards of standard designs for a nuclear power reactor of the type described in § 50.22 of this chapter or major portions thereof.

§ 52.133 Relationship to other subparts.

(a) This subpart applies to a person that requests a standard design approval from the NRC staff separately from an application for a construction permit filed under 10 CFR part 50 or a combined license filed under subpart C of this part. An applicant for a construction permit or combined license may reference a standard design approval.

(b) Subpart B of this part governs the certification by rulemaking of the design of a nuclear power plant. Subpart B may be used independently of the provisions in this subpart.

(c) Subpart F of this part governs the issuance of licenses to manufacture nuclear power reactors to be installed and operated at sites not identified in the manufacturing license application. Subpart F of this part may be used independently of the provisions in this subpart.

§ 52.135 Filing of applications.

(a) Any person may submit a proposed standard design for a nuclear power reactor of the type described in 10 CFR 50.22 to the NRC staff for its review. The submittal may consist of either the final design for the entire facility or the final design of major portions thereof.

(b) The submittal for review of the proposed standard design must be made in the same manner and in the same number of copies as provided in 10 CFR 50.30 and 52.3 for license applications.

(c) The fees associated with the filing and review of the application are set forth in 10 CFR part 170.

§ 52.136 Contents of applications; general information.

The application must contain all of the information required by 10 CFR 50.33(a) through (d) and (j).

§ 52.137 Contents of applications; technical information.

If the applicant seeks review of a major portion of a standard design, the application need only contain the information required by this section to the extent the requirements are applicable to the major portion of the standard design for which NRC staff approval is sought.

(a) The application must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility, or major portion thereof, and must include the following information:

(1) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters;

(2) A description and analysis of the SSCs of the facility, with emphasis upon performance requirements, the bases, with technical justification, upon which the requirements have been established, and the evaluations required to show that safety functions will be accomplished. It is expected that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations. Items such as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent. The following power reactor design characteristics will be taken into consideration by the Commission:

(i) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(ii) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(iii) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials; and

(iv) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release[9] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable postulated site parameters, including site meteorology, to evaluate the offsite radiological consequences. The evaluation must determine that:

(A) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission

[9] The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

product release, would not receive a radiation dose in excess of 25 rem[10] total effective dose equivalent (TEDE); and

(B) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE;

(3) The design of the facility including:

(i) The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units;

(ii) The design bases and the relation of the design bases to the principal design criteria; and

(iii) Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with adequate margin for safety;

(4) An analysis and evaluation of the design and performance of SSC with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of 10 CFR 50.46 and 50.46a;

(5) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20 of this chapter;

(6) The information required by §20.1406 of this chapter;

(7) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter;

(8) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v) of 10 CFR 50.34(f);

(9) For applications for light-water-cooled nuclear power plants, an evaluation of the standard plant design against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP is not a substitute for the regulations, and compliance is not a requirement;

(10) The information with respect to the design of equipment to maintain control over radioactive materials in gaseous and liquid effluents produced

[10] A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accident.

during normal reactor operations described in 10 CFR 50.34a(e);

(11) The information pertaining to design features that affect plans for coping with emergencies in the operation of the reactor facility or a major portion thereof;

(12) An analysis and description of the equipment and systems for combustible gas control as required by § 50.44 of this chapter;

(13) The list of electric equipment important to safety that is required by 10 CFR 50.49(d);

(14) A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in 10 CFR 50.60 and 50.61;

(15) Information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transients without scram (ATWS) events in § 50.62;

(16) The coping analysis, and any design features necessary to address station blackout, as described in § 50.63 of this chapter;

(17) Information demonstrating how the applicant will comply with requirements for criticality accidents in § 50.68(b)(2)–(b)(4);

(18) A description and analysis of the fire protection design features for the standard plant necessary to comply with part 50, appendix A, GDC 3, and § 50.48 of this chapter;

(19) A description of the quality assurance program applied to the design of the SSCs of the facility. Appendix B to 10 CFR part 50, ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,'' sets forth the requirements for quality assurance programs for nuclear power plants. The description of the quality assurance program for a nuclear power plant shall include a discussion of how the applicable requirements of appendix B to 10 CFR part 50 were satisfied;

(20) The information necessary to demonstrate that the standard plant complies with the earthquake engineering criteria in 10 CFR part 50, appendix S;

(21) Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority generic safety issues which are identified in the version of NUREG–0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design;

(22) The information necessary to demonstrate how operating experience insights have been incorporated into the plant design;

(23) For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass;

(24) A description, analysis, and evaluation of the interfaces between the standard design and the balance of the nuclear power plant; and

(25) A description of the design-specific probabilistic risk assessment and its results.

(b) An application for approval of a standard design, which differs significantly from the light-water reactor designs of plants that have been licensed and in commercial operation before April 18, 1989, or uses simplified, inherent, passive, or other innovative means to accomplish its safety functions, must meet the requirements of 10 CFR 50.43(e).

§ 52.139 Standards for review of applications.

Applications filed under this subpart will be reviewed for compliance with the standards set out in 10 CFR parts 20, 50 and its appendices, and 10 CFR parts 73 and 100.

§ 52.141 Referral to the Advisory Committee on Reactor Safeguards (ACRS).

The Commission shall refer a copy of the application to the ACRS. The ACRS shall report on those portions of the application which concern safety.

§ 52.143 Staff approval of design.

Upon completion of its review of a submittal under this subpart and receipt of a report by the Advisory Committee on Reactor Safeguards under § 52.141 of this subpart, the NRC staff shall publish a determination in the FEDERAL REGISTER as to whether or not the design is acceptable, subject to appropriate terms and conditions, and make an analysis of the design in the form of a report available at the NRC Web site, *http://www.nrc.gov*.

§ 52.145 Finality of standard design approvals; information requests.

(a) An approved design must be used by and relied upon by the NRC staff and the ACRS in their review of any individual facility license application that incorporates by reference a standard design approved in accordance with this paragraph unless there exists significant new information that substantially affects the earlier determination or other good cause.

(b) The determination and report by the NRC staff do not constitute a commitment to issue a permit or license, or in any way affect the authority of the Commission, Atomic Safety and Licensing Board Panel, or presiding officers in any proceeding under part 2 of this chapter.

(c) Except for information requests seeking to verify compliance with the current licensing basis of the standard design approval, information requests to the holder of a standard design approval must be evaluated before issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information. Each evaluation performed by the NRC staff must be in accordance with 10 CFR 50.54(f) and must be approved by the Executive Director for Operations or his or her designee before issuance of the request.

§ 52.147 Duration of design approval.

A standard design approval issued under this subpart is valid for 15 years from the date of issuance and may not be renewed. A design approval continues to be valid beyond the date of expiration in any proceeding on an application for a construction permit or an operating license under part 50 or a combined license or manufacturing license under part 52 that references the final design approval and is docketed before the date of expiration of the design approval.

Subpart F—Manufacturing Licenses

§ 52.151 Scope of subpart.

This subpart sets out the requirements and procedures applicable to Commission issuance of a license authorizing manufacture of nuclear power reactors to be installed at sites not identified in the manufacturing license application.

§ 52.153 Relationship to other subparts.

(a) A nuclear power reactor manufactured under a manufacturing license issued under this subpart may only be transported to and installed at a site for which either a construction permit under part 50 of this chapter or a combined license under subpart C of this part has been issued.

(b) Subpart B of this part governs the certification by rulemaking of the design of standard nuclear power facilities. Subpart E of this part governs the NRC staff review and approval of standard designs for a nuclear power facility. A manufacturing license applicant may reference a standard design certification or a standard design approval in its application. These subparts may also be used independently of the provisions in this subpart.

§ 52.155 Filing of applications.

(a) Any person, except one excluded by 10 CFR 50.38, may file an application for a manufacturing license under this subpart with the Director of New Reactors or the Director of Nuclear Reactor Regulation, as appropriate.

(b) The application must comply with the applicable filing requirements of §§ 52.3 and 50.30 of this chapter.

(c) The fees associated with the filing and review of the application are set forth in 10 CFR part 170.

§ 52.156 Contents of applications; general information.

The application must contain all of the information required by 10 CFR 50.33(a) through (d), and (j).

§ 52.157 Contents of applications; technical information in final safety analysis report.

The application must contain a final safety analysis report containing the information set forth below, with a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that the manufacturing conforms to the design and to reach a final conclusion on all safety questions associated with the design, permit the preparation of construction and installation specifications by an applicant who seeks to use the manufactured reactor, and permit the preparation of acceptance and inspection requirements by the NRC:

(a) The principal design criteria for the reactor to be manufactured. Appendix A of 10 CFR part 50, ''General Design Criteria for Nuclear Power Plants,'' establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units;

(b) The design bases and the relation of the design bases to the principal design criteria;

(c) A description and analysis of the structures, systems, and components of the reactor to be manufactured, with emphasis upon the materials of manufacture, performance requirements, the bases, with technical justification therefor, upon which the performance requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations. Items such as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent. The following power reactor design characteristics will be taken into consideration by the Commission:

(1) Intended use of the manufactured reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(2) The extent to which generally accepted engineering standards are applied to the design of the reactor; and

(3) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(d) The safety features that are engineered into the reactor and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to reactor design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release[11] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable postulated site parameters, including site meteorology, to evaluate the offsite radiological consequences. The evaluation must determine that:

(1) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation

[11] The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

dose in excess of 25 rem[12] total effective dose equivalent (TEDE);

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE; and

(e) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20 of this chapter.

(f) Information necessary to establish that the design of the reactor to be manufactured complies with the technical requirements in 10 CFR Chapter I, including:

(1) An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter;

(2) A description and analysis of the fire protection design features for the reactor necessary to comply with 10 CFR part 50, appendix A, GDC 3 and § 50.48 of this chapter;

(3) A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in §§ 50.60 and 50.61 of this chapter;

(4) An analysis and description of the equipment and systems for combustible gas control as required by § 50.44 of this chapter;

(5) The coping analysis, and any design features necessary to address station blackout, as described in § 50.63 of this chapter;

(6) The list of electric equipment important to safety that is required by 10 CFR 50.49(d);

(7) Information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transients without scram (ATWS) events in § 50.62;

(8) Information demonstrating how the applicant will comply with requirements for criticality accidents in § 50.68(b)(2)–(b)(4);

(9) The information required by § 20.1406 of this chapter;

(10) [Reserved]

(11) The information with respect to the design of equipment to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, as described in § 50.34a(e) of this chapter;

(12) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in § 50.34(f) of this chapter, except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v);

(13) If the applicant seeks to use risk-informed treatment of SSCs in accordance with § 50.69 of this chapter, the information required by § 50.69(b)(2) of this chapter;

(14) The information necessary to demonstrate that the manufactured reactor complies with the earthquake engineering criteria in appendix S to 10 CFR part 50;

[12] A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accident.

(15) Information sufficient to demonstrate compliance with the applicable requirements regarding testing, analysis, and prototypes as set forth in § 50.43(e) of this chapter;

(16) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter;

(17) A description of the quality assurance program applied to the design, and to be applied to the manufacture of, the structures, systems, and components of the reactor. Appendix B to 10 CFR part 50, ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,'' sets forth the requirements for quality assurance programs for nuclear power plants. The description of the quality assurance program must include a discussion of how the applicable requirements of appendix B to 10 CFR part 50 have been and will be satisfied; and

(18) Proposed technical specifications applicable to the reactor being manufactured, prepared in accordance with the requirements of §§ 50.36 and 50.36a of this chapter;

(19) The site parameters postulated for the design, and an analysis and evaluation of the reactor design in terms of those site parameters;

(20) The interface requirements between the manufactured reactor and the remaining portions of the nuclear power plant. These requirements must be sufficiently detailed to allow for completion of the final safety analysis;

(21) Justification that compliance with the interface requirements of paragraph (f)(20) of this section is verifiable through inspections, testing, or analysis. The method to be used for verification of interface requirements must be included as part of the proposed ITAAC required by § 52.158(a);

(22) A representative conceptual design for a nuclear power facility using the manufactured reactor, to aid the NRC in its review of the final safety analysis required by this section and to permit assessment of the adequacy of the interface requirements in paragraph (f)(20) of this section;

(23) For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass;

(24) [Reserved]

(25) If the reactor is to be used in modular plant design, a description of the possible operating configurations of the reactor modules with common systems, interface requirements, and system interactions. The final safety analysis must also account for differences among the configurations, including any restrictions that will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating;

(26) A description of the management plan for design and manufacturing activities, including:

(i) The organizational and management structure singularly responsible for direction of design and manufacture of the reactor;

(ii) Technical resources directed by the applicant, and the qualifications requirements;

(iii) Details of the interaction of design and manufacture within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor, as applicable;

(iv) Proposed procedures governing the preparation of the manufactured reactor for shipping to the site where it is to be operated, the conduct of shipping, and verifying the condition of the manufactured reactor upon receipt at the site; and

(v) The degree of top level management oversight and technical control to be exercised by the applicant during design and manufacture, including the preparation and implementation of procedures necessary to guide the effort;

(27) Necessary parameters to be used in developing plans for preoperational testing and initial operation;

(28) Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority generic safety issues which are identified in the version of NUREG–0933 current on the date up to 6 months before the

docket date of the application and which are technically relevant to the design;

(29) The information necessary to demonstrate how operating experience insights have been incorporated into the manufactured reactor design;

(30) For applications for light-water-cooled nuclear power plants, an evaluation of the design to be manufactured against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP is not a substitute for the regulations, and compliance is not a requirement; and

(31) A description of the design-specific probabilistic risk assessment and its results.

§ 52.158 Contents of application; additional technical information.

The application must contain:

(a)(1) *Inspections, tests, analyses, and acceptance criteria (ITAAC).* The proposed inspections, tests, and analyses that the licensee who will be operating the reactor shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met:

(i) The reactor has been manufactured in conformity with the manufacturing license; the provisions of the Act, and the Commission's rules and regulations; and

(ii) The manufactured reactor will be operated in conformity with the approved design and any license authorizing operation of the manufactured reactor.

(2) If the application references a standard design certification, the ITAAC contained in the certified design must apply to those portions of the facility design which are covered by the design certification.

(3) If the application references a standard design certification, the application may include a notification that a required inspection, test, or analysis in the design certification ITAAC has been successfully completed and that the corresponding acceptance criterion has been met. The FEDERAL REGISTER notification required by § 52.163 must indicate that the application includes this notification.

(b)(1) An environmental report as required by 10 CFR 51.54.

(2) If the manufacturing license application references a standard design certification, the environmental report need not contain a discussion of severe accident mitigation design alternatives for the reactor.

§ 52.159 Standards for review of application.

Applications filed under this subpart will be reviewed according to the applicable standards set out in 10 CFR parts 20, 50 and its appendices, 51, 73, and 100 and its appendices.

§ 52.161 [Reserved]

§ 52.163 Administrative review of applications; hearings.

A proceeding on a manufacturing license is subject to all applicable procedural requirements contained in 10 CFR part 2, including the requirements for docketing in § 2.101(a)(1) through (4) of this chapter, and the requirements for issuance of a notice of proposed action in § 2.105 of this chapter, *provided, however*, that the designated sections may not be construed to require that the environmental report or draft or final environmental impact statement include an assessment of the benefits of constructing and/or operating the manufactured reactor or an evaluation of alternative energy sources. All hearings on manufacturing licenses are governed by the hearing procedures contained in 10 CFR part 2, subparts C, G, L, and N.

§ 52.165 Referral to the Advisory Committee on Reactor Safeguards (ACRS).

The Commission shall refer a copy of the application to the ACRS. The ACRS shall report on those portions of the application which concern safety.

§ 52.167 Issuance of manufacturing license.

(a) After completing any hearing under § 52.163, and receiving the report submitted by the ACRS, the Commission may issue a manufacturing license if the Commission finds that:

(1) Applicable standards and requirements of the Act and the Commission's regulations have been met;

(2) There is reasonable assurance that the reactor(s) will be manufactured, and can be transported, incorporated into a nuclear power plant, and operated in conformity with the manufacturing license, the provision of the Act, and the Commission's regulations;

(3) The proposed reactor(s) can be incorporated into a nuclear power plant and operated at sites having characteristics that fall within the site parameters postulated for the design of the manufactured reactor(s) without undue risk to the health and safety of the public;

(4) The applicant is technically qualified to design and manufacture the proposed nuclear power reactor(s);

(5) The proposed inspections, tests, analyses and acceptance criteria are necessary and sufficient, within the scope of the manufacturing license, to provide reasonable assurance that the manufactured reactor has been manufactured and will be operated in conformity with the license, the provisions of the Act, and the Commission's regulations;

(6) The issuance of a license to the applicant will not be inimical to the common defense and security or to the health and safety of the public; and

(7) The findings required by subpart A of part 51 of this chapter have been made.

(b) Each manufacturing license issued under this subpart shall specify:

(1) Terms and conditions as the Commission deems necessary and appropriate;

(2) Technical specifications for operation of the manufactured reactor, as the Commission deems necessary and appropriate;

(3) Site parameters and design characteristics for the manufactured reactor; and

(4) The interface requirements to be met by the site-specific elements of the facility, such as the service water intake structure and the ultimate heat sink, not within the scope of the manufactured reactor.

(c)(1) A holder of a manufacturing license may not transport or allow to be removed from the place of manufacture the manufactured reactor except to the site of a licensee with either a construction permit under part 50 of this chapter or a combined license under subpart C of this part. The construction permit or combined license must authorize the construction of a nuclear power facility using the manufactured reactor(s).

(2) A holder of a manufacturing license shall include, in any contract governing the transport of a manufactured reactor from the place of manufacture to any other location, a provision requiring that the person or entity transporting the manufactured reactor to comply with all NRC-approved shipping requirements in the manufacturing license.

§ 52.169 [Reserved]

§ 52.171 Finality of manufacturing licenses; information requests.

(a)(1) Notwithstanding any provision in 10 CFR 50.109, during the term of a manufacturing license the Commission may not modify, rescind, or impose new requirements on the design of the nuclear power reactor being manufactured, or the requirements for the manufacture of the nuclear power reactor, unless the Commission determines that a modification is necessary to bring the design of the reactor or its manufacture into compliance with the Commission's requirements applicable and in effect at the time the manufacturing license was issued, or to provide reasonable assurance of adequate protection to public health and safety or common defense and security.

(2) Any modification to the design of a manufactured nuclear power reactor which is imposed by the Commission under paragraph (a)(1) of this section will be applied to all reactors manufactured under the license, including those that have already been transported and sited, except those reactors to which the modification has been rendered technically irrelevant by action taken under paragraph (b) of this section.

(3) In making the findings required for issuance of a construction permit, operating license, combined license, in any hearing under § 52.103, or in any enforcement hearing other than one initiated by the Commission under paragraph (a)(1) of this section, for which a nuclear power reactor manufactured under this subpart is referenced or used, the Commission shall treat as resolved those matters resolved in the proceeding on the application for issuance or renewal of the manufacturing license, including the adequacy of design of the manufactured reactor, the costs and benefits of severe accident mitigation design alternatives, and the bases for not incorporating severe accident mitigation design alternatives into the design of the reactor to be manufactured.

(b)(1) The holder of a manufacturing license may not make changes to the design of the nuclear power reactor authorized to be manufactured without prior Commission approval. The request for a change to the design must be in the form of an application for a license amendment, and must meet the requirements of 10 CFR 50.90 and 50.92.

(2) An applicant or licensee who references or uses a nuclear power reactor manufactured under a manufacturing license under this subpart may request a departure from the design characteristics, site parameters, terms and conditions, or approved design of the manufactured reactor. The Commission may grant a request only if it determines that the departure will comply with the requirements of 10 CFR 52.7, and that the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the departure. The granting of a departure on request of an applicant is subject to litigation in the same manner as other issues in the construction permit or combined license hearing.

(c) Except for information requests seeking to verify compliance with the current licensing basis of either the manufacturing license or the manufactured reactor, information requests to the holder of a manufacturing license or an applicant or licensee using a manufactured reactor must be evaluated before issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information. Each evaluation performed by the NRC staff must be in accordance with 10 CFR 50.54(f) and must be approved by the Executive Director for Operations or his or her designee before issuance of the request.

§ 52.173 Duration of manufacturing license.

A manufacturing license issued under this subpart may be valid for not less than 5, nor more than 15 years from the date of issuance. A holder of a manufacturing license may not initiate the manufacture of a reactor less than 3 years before the expiration of the license even though a timely application for renewal has been docketed with the NRC. Upon expiration of the manufacturing license, the manufacture of any uncompleted reactors must cease unless a timely application for renewal has been docketed with the NRC.

§ 52.175 Transfer of manufacturing license.

A manufacturing license may be transferred in accordance with § 50.80 of this chapter.

§ 52.177 Application for renewal.

(a) Not less than 12 months, nor more than 5 years before the expiration of the manufacturing license, or any later renewal period, the holder of the manufacturing license may apply for a renewal of the license. An application for renewal must contain all information necessary to bring up to date the information and data contained in the previous application.

(b) The filing of an application for a renewed license must be in accordance with subpart A of 10 CFR part 2 and 10 CFR 52.3 and 50.30.

(c) A manufacturing license, either original or renewed, for which a timely application for renewal has been filed, remains in effect until the Commission has made a final determination on the renewal application, *provided, however*, that in accordance with § 52.173, the holder of a manufacturing license may not begin manufacture of a reactor less than 3 years before the expiration of the license.

(d) Any person whose interest may be affected by renewal of the permit may request a hearing on the application for renewal. The request for a hearing must comply with 10 CFR 2.309. If a hearing is granted, notice of the hearing will be published in accordance with 10 CFR 2.104.

(e) The Commission shall refer a copy of the application for renewal to the Advisory Committee on Reactor Safeguards (ACRS). The ACRS shall report on those portions of the application which concern safety and shall apply the criteria set forth in § 52.159.

§ 52.179 Criteria for renewal.

The Commission may grant the renewal if the Commission determines:

(a) The manufacturing license complies with the Atomic Energy Act and the Commission's regulations and orders applicable and in effect at the time the manufacturing license was originally issued; and

(b) Any new requirements the Commission may wish to impose are:

(1) Necessary for adequate protection to public health and safety or common defense and security;

(2) Necessary for compliance with the Commission's regulations and orders applicable and in effect at the time the manufacturing license was originally issued; or

(3) A substantial increase in overall protection of the public health and safety or the common defense and security to be derived from the new requirements, and the direct and indirect costs of implementation of those requirements are justified in view of this increased protection.

§ 52.181 Duration of renewal.

A renewed manufacturing license may be issued for a term of not less than 5, nor more than 15 years, plus any remaining years on the manufacturing license then in effect before renewal. The renewed license shall be subject to the requirements of §§ 52.171 and 52.175.

Subpart G [Reserved]

Subpart H—Enforcement

§ 52.301 Violations.

(a) The Commission may obtain an injunction or other court order to prevent a violation of the provisions of—

(1) The Atomic Energy Act of 1954, as amended;

(2) Title II of the Energy Reorganization Act of 1974, as amended; or

(3) A regulation or order issued under those Acts.

(b) The Commission may obtain a court order for the payment of a civil penalty imposed under Section 234 of the Atomic Energy Act:

(1) For violations of—

(i) Sections 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, or 109 of the Atomic Energy Act of 1954, as amended;

(ii) Section 206 of the Energy Reorganization Act;

(iii) Any regulation, or order issued under the sections specified in paragraph (b)(1)(i) of this section;

(iv) Any term, condition, or limitation of any license issued under the sections specified in paragraph (b)(1)(i) of this section.

(2) For any violation for which a license may be revoked under Section 186 of the Atomic Energy Act of 1954, as amended.

§ 52.303 Criminal penalties.

(a) Section 223 of the Atomic Energy Act of 1954, as amended, provides for criminal sanctions for willful violation of, attempted violation of, or conspiracy to violate, any regulation issued under Sections 161b, 161i, or 161o of the Act. For purposes of Section 223, all the regulations in part 52 are issued under one or more of Sections 161b, 161i, or 160o, except for the sections listed in paragraph (b) of this section.

(b) The regulations in part 52 that are not issued under Sections 161b, 161i, or 161o for the purposes of Section 223 are as follows: §§52.0, 52.1, 52.2, 52.3, 52.7, 52.8, 52.9, 52.10, 52.11, 52.12, 52.13, 52.15, 52.16, 52.17, 52.18, 52.21, 52.23, 52.24, 52.27, 52.28, 52.29, 52.31, 52.33, 52.39, 52.41, 52.43, 52.45, 52.46, 52.47, 52.48, 52.51, 52.53, 52.54, 52.55, 52.57, 52.59, 52.61, 52.63, 52.71, 52.73, 52.75, 52.77, 52.79, 52.80, 52.81, 52.83, 52.85, 52.87, 52.93, 52.97, 52.98, 52.103, 52.104, 52.105, 52.107, 52.109, 52.131, 52.133, 52.135, 52.136, 52.137, 52.139, 52.141, 52.143, 52.145, 52.147, 52.151, 52.153, 52.155, 52.156, 52.157, 52.158, 52.159, 52.161, 52.163, 52.165, 52.167, 52.171, 52.173, 52.175, 52.177, 52.179, 52.181, 52.301, and 52.303.

Appendix A to Part 52—Design Certification Rule for the U.S. Advanced Boiling Water Reactor

I. Introduction

Appendix A constitutes the standard design certification for the U.S. Advanced Boiling Water Reactor (ABWR) design, in accordance with 10 CFR part 52, subpart B. The applicant for certification of the U.S. ABWR design was GE Nuclear Energy.

II. Definitions

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

B. Generic technical specifications 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 (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 of this appendix to reference Tier 2 when referencing Tier 1. Tier 2 information includes:

1. Information required by §§52.47(a) and 52.47(c), with the exception of generic technical specifications and conceptual design information;
2. Supporting information on the inspections, tests, and analyses that will be performed to demonstrate that the acceptance criteria in the ITAAC have been met; and
3. Combined license (COL) action items (COL license information), which identify certain matters that must 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 are not requirements for the licensee unless such items are restated in the FSAR.

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 Section VIII.B.6 of this appendix. This designation expires for some Tier 2* information under Section VIII.B.6.

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

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

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

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

III. Scope and Contents

A. Tier 1, Tier 2, and the generic technical specifications in the U.S. ABWR Design Control Document, GE Nuclear Energy, Revision 4 dated March 1997, are approved for incorporation by reference by the Director of the Office of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR part 51. Copies of the generic DCD may be obtained from the National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22161. A copy is available for examination and copying at the NRC Public Document Room located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Copies are also available for examination at the NRC Library located at Two White Flint North, 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, and the generic technical specifications except as otherwise provided in this appendix. Conceptual design information, as set forth in the generic DCD, and the ''Technical Support Document for the ABWR'' are not part of this appendix. Tier 2 references to the probabilistic risk assessment (PRA) in the ABWR standard safety analysis report do not incorporate the PRA into Tier 2.

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 U.S. ABWR design or NUREG–1503, ''Final Safety Evaluation Report related to the Certification of the Advanced Boiling Water Reactor Design'' (FSER), and Supplement No. 1, then the generic DCD controls.

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

IV. Additional Requirements and Restrictions

A. An applicant for a combined license that wishes to reference this appendix shall, in addition to complying with the requirements of 10 CFR 52.77, 52.79, and 52.80, comply with the following requirements:

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

2. Include, as part of its application:

a. A plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for the U.S. ABWR design, as modified and supplemented by the applicant's exemptions and departures;

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

c. Plant-specific 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 that is not within the scope of this appendix.

3. Include, in the plant-specific DCD, the proprietary information and safeguards information referenced in the U.S. ABWR 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 10 CFR part 50.

V. Applicable Regulations

A. Except as indicated in paragraph B of this section, the regulations that apply to the U.S. ABWR design are in 10 CFR parts 20, 50, 73, and 100, codified as of May 2, 1997, that are applicable and technically relevant, as described in the FSER (NUREG–1503) and Supplement No. 1.

B. The U.S. ABWR design is exempt from portions of the following regulations:

1. Paragraph (f)(2)(iv) of 10 CFR 50.34—Separate Plant Safety Parameter Display Console;

2. Paragraph (f)(2)(viii) of 10 CFR 50.34—Post-Accident Sampling for Boron, Chloride, and Dissolved Gases; and

3. Paragraph (f)(3)(iv) of 10 CFR 50.34—Dedicated Containment Penetration.

VI. Issue Resolution

A. The Commission has determined that the structures, systems, components, and design features of the U.S. ABWR 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 U.S. ABWR design.

B. The Commission considers the following matters resolved within the meaning of 10 CFR 52.63(a)(5) in subsequent proceedings for issuance of a combined license, amendment of a combined license, or renewal of a combined license, proceedings held under 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 and Supplement No. 1, Tier 1, Tier 2 (including referenced information which the context indicates is intended as requirements), and the rulemaking record for certification of the U.S. ABWR 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 U.S. ABWR design;

3. All generic changes to the DCD under 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 under and in compliance with the change processes in Sections VIII.A.4 and VIII.B.4 of this appendix, but only for that plant;

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

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

7. All environmental issues concerning severe accident mitigation design alternatives associated with the information in the NRC's final environmental assessment for the U.S. ABWR design and Revision 1 of the technical support document for the U.S. ABWR, dated December 1994, for plants referencing this appendix whose site parameters are within those specified in the technical support document.

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)(5). The Commission reserves the right to require operational requirements for an applicant or licensee who references this appendix by rule, regulation, order, or license condition.

D. Except 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 DCD for the U.S. ABWR design, 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 GE Nuclear Energy. 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 at the NRC Web site, *http://www.nrc.gov*, and/or at the NRC 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 GE Nuclear Energy declines to provide the information sought, GE Nuclear Energy shall send a written response within 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 GE Nuclear Energy), and GE Nuclear Energy's response. The Commission and presiding officer may order GE Nuclear Energy 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 June 11, 1997, 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)(4).

4. Exemptions from Tier 1 information are governed by the requirements in 10 CFR 52.63(b)(1) and 52.98(f). The Commission will deny a request for an exemption from Tier 1, if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design.

B. Tier 2 information.

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

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

3. The Commission may not require new requirements on Tier 2 information by plant-specific order while this appendix is in effect under §§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 52.7 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 52.7. 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 requires a license amendment under paragraphs B.5.b or B.5.c of this section. When evaluating the proposed departure, an applicant or licensee shall consider all matters described in the plant-specific DCD.

b. A proposed departure from Tier 2, other than one affecting resolution of a severe accident issue identified in the plant-specific DCD, requires a license amendment if it would:

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

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

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

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

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

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

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

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

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

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

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

d. If a departure requires a license amendment pursuant to paragraphs B.5.b or B.5.c 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 paragraph VIII.B.5 of this appendix when departing from Tier 2 information, may petition the NRC to admit into the proceeding such a contention. In addition to compliance with the general requirements of 10 CFR 2.309, the petition must demonstrate that the departure does not comply with paragraph VIII.B.5 of this appendix. Further, the petition must demonstrate that the change bears on an asserted noncompliance with an ITAAC acceptance criterion in the case of a 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 material fact regarding compliance with paragraph VIII.B.5 of this appendix.

6.a. An applicant who references this appendix may not depart from Tier 2* information, which is designated with italicized text or brackets and an asterisk in the generic DCD, without NRC approval. The departure will not be considered a resolved issue, within the meaning of Section VI of this appendix and 10 CFR 52.63(a)(5).

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

(1) Fuel burnup limit (4.2).

(2) Fuel design evaluation (4.2.3).

(3) Fuel licensing acceptance criteria (appendix 4B).

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) ASME Boiler & Pressure Vessel Code, Section III.

(2) ACI 349 and ANSI/AISC–690.

(3) Motor-operated valves.

(4) Equipment seismic qualification methods.

(5) Piping design acceptance criteria.

(6) Fuel system and assembly design (4.2), except burnup limit.

(7) Nuclear design (4.3).

(8) Equilibrium cycle and control rod patterns (App. 4A).

(9) Control rod licensing acceptance criteria (App. 4C).

(10) Instrument setpoint methodology.

(11) EMS performance specifications and architecture.

(12) SSLC hardware and software qualification.

(13) Self-test system design testing features and commitments.

(14) Human factors engineering design and implementation process.

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

3. The Commission may require plant-specific departures on generic 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 2.335 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 52.7. The grant of an exemption must be subject to litigation in the same manner as other issues material to the license hearing.

5. A party to an adjudicatory proceeding for 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 2.309 and must demonstrate why

special circumstances as defined in 10 CFR 2.335 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 met.

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 met, 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 52.103(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 §52.103(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 52.98 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, Tier 2, and the generic TS and other operational requirements. 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 under Section VIII of this appendix throughout the period of application and for the term of the license (including any period of renewal).

3. An applicant or licensee who references this appendix shall prepare and maintain written evaluations 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 plant-specific departures from the DCD, including a summary of the evaluation of each. This report must be filed in accordance with the filing requirements applicable to reports in 10 CFR 52.3.

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

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

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

b. During the interval from the date of application for a license to the date the Commission makes the finding required by 10 CFR 52.103(g), the report must be submitted semiannually. Updates to the plant-specific DCD must be submitted annually and may be submitted along with amendments to the application.

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

APPENDIX B TO PART 52—DESIGN CERTIFICATION RULE FOR THE SYSTEM 80+ DESIGN

I. INTRODUCTION

Appendix B constitutes design certification for the System 80+[1] standard plant design, in accordance with 10 CFR part 52, subpart B. The applicant for certification of the System 80+ design was Combustion Engineering, Inc. (ABB–CE), which is now 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 (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 of this appendix to reference Tier 2 when referencing Tier 1. Tier 2 information includes:

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

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

3. Combined license (COL) action items (COL license information), which identify certain matters that must 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 are not requirements for the licensee unless such items are restated in the FSAR.

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 Section VIII.B.6 of this appendix. This designation expires for some Tier 2* information under Section VIII.B.6 of this appendix.

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

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

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

H. All other terms in this appendix have the meaning set out in 10 CFR 50.2 or 52.1, or

[1] "System 80+" is a trademark of Westinghouse Electric Company LLC.

Section 11 of the Atomic Energy Act of 1954, as amended, as applicable.

III. SCOPE AND CONTENTS

A. Tier 1, Tier 2, and the generic technical specifications in the System 80+ Design Control Document, ABB–CE, with revisions dated January 1997, are approved for incorporation by reference by the Director of the Office of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR part 51. Copies of the generic DCD may be obtained from the National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22161. A copy is available for examination and copying at the NRC Public Document Room located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Copies are also available for examination at the NRC Library located at Two White Flint North, 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, and the generic technical specifications except as otherwise provided in this appendix. Conceptual design information, as set forth in the generic DCD, and the Technical Support Document for the System 80+ design 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 System 80+ design or NUREG–1462, ''Final Safety Evaluation Report Related to the Certification of the System 80+ Design,'' (FSER) and Supplement No. 1, then the generic DCD controls.

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

IV. ADDITIONAL REQUIREMENTS AND RESTRICTIONS

A. An applicant for a combined license that wishes to reference this appendix shall, in addition to complying with the requirements of 10 CFR 52.77, 52.79, and 52.80, comply with the following requirements:

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

2. Include, as part of its application:

a. A plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for the System 80+ design, as modified and supplemented by the applicant's exemptions and departures;

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

c. Plant-specific 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 that is not within the scope of this appendix.

3. Include, in the plant-specific DCD, the proprietary information referenced in the System 80+ 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 10 CFR part 50.

V. APPLICABLE REGULATIONS

A. Except as indicated in paragraph B of this section, the regulations that apply to the System 80+ design are in 10 CFR parts 20, 50, 73, and 100, codified as of May 9, 1997, that are applicable and technically relevant, as described in the FSER (NUREG–1462) and Supplement No. 1.

B. The System 80+ design is exempt from portions of the following regulations:

1. Paragraph (f)(2)(iv) of 10 CFR 50.34—Separate Plant Safety Parameter Display Console;

2. Paragraphs (f)(2) (vii), (viii), (xxvi), and (xxviii) of 10 CFR 50.34—Accident Source Terms;

3. Paragraph (f)(2)(viii) of 10 CFR 50.34—Post-Accident Sampling for Hydrogen, Boron, Chloride, and Dissolved Gases;

4. Paragraph (f)(3)(iv) of 10 CFR 50.34—Dedicated Containment Penetration; and

5. Paragraphs III.A.1(a) and III.C.3(b) of Appendix J to 10 CFR 50—Containment Leakage Testing.

VI. ISSUE RESOLUTION

A. The Commission has determined that the structures, systems, components, and design features of the System 80+ 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 System 80+ design.

B. The Commission considers the following matters resolved within the meaning of 10 CFR 52.63(a)(5) in subsequent proceedings for issuance of a combined license, amendment of a combined license, or renewal of a combined license, proceedings held under 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 and Supplement No. 1, Tier 1, Tier 2 (including referenced information which the context indicates is intended as requirements), and the rulemaking record for certification of the System 80+ 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 System 80+ design;

3. All generic changes to the DCD under 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 under and in compliance with the change processes in Sections VIII.A.4 and VIII.B.4 of this appendix, but only for that plant;

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

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

7. All environmental issues concerning severe accident mitigation design alternatives associated with the information in the NRC's final environmental assessment for the System 80+ design and the technical support document for the System 80+ design, dated January 1995, for plants referencing this appendix whose site parameters are within those specified in the technical support document.

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)(5). The Commission reserves the right to require operational requirements for an applicant or licensee who references this appendix by rule, regulation, order, or license condition.

D. Except 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 information or other secondary references in the DCD for the System 80+ design, 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 at the NRC Web site, *http://www.nrc.gov*, and/or at the NRC 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 June 20, 1997, 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)(4).

4. Exemptions from Tier 1 information are governed by the requirements in 10 CFR 52.63(b)(1) and 52.98(f). The Commission will deny a request for an exemption from Tier 1, if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design.

B. TIER 2 INFORMATION

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

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

3. The Commission may not require new requirements on Tier 2 information by plant-specific order while this appendix is in effect under §§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 52.7 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 52.7. 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 requires a license amendment under paragraphs B.5.b or B.5.c of this section. When evaluating the proposed departure, an applicant or licensee shall consider all matters described in the plant-specific DCD.

b. A proposed departure from Tier 2, other than one affecting resolution of a severe accident issue identified in the plant-specific DCD, requires a license amendment if it would—

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

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

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

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

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

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

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

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

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

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

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

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

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 paragraph VIII.B.5 of this appendix when departing from Tier 2 information, may petition the NRC to admit into the proceeding such a contention. In addition to compliance with the general requirements of 10 CFR 2.309, the petition must demonstrate that the departure does not comply with paragraph VIII.B.5 of this appendix. Further, the petition must demonstrate that the change bears on an asserted noncompliance with an ITAAC acceptance criterion in the case of a 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 material fact regarding compliance with paragraph VIII.B.5 of this appendix.

6.a. An applicant who references this appendix may not depart from Tier 2* information, which is designated with italicized text or brackets and an asterisk in the generic DCD, without NRC approval. The departure will not be considered a resolved issue, within the meaning of Section VI of this appendix and 10 CFR 52.63(a)(5).

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

(1) Maximum fuel rod average burnup.

(2) Control room 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) ASME Boiler & Pressure Vessel Code, Section III.

(2) ACI 349 and ANSI/AISC–690.

(3) Motor-operated valves.

(4) Equipment seismic qualification methods.

(5) Piping design acceptance criteria.

(6) Fuel and control rod design, except burnup limit.

(7) Instrumentation and controls setpoint methodology.

(8) Instrumentation and controls hardware and software changes.

(9) Instrumentation and controls environmental qualification.

(10) Seismic design criteria for non-seismic Category I structures.

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

3. The Commission may require plant-specific departures on generic 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 2.335 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 52.7. The grant of an exemption must be subject to litigation in the same manner as other issues material to the license hearing.

5. A party to an adjudicatory proceeding for 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 a petition must comply with the general requirements of 10 CFR 2.309 and must demonstrate why special circumstances as defined in 10 CFR 2.335 are present, or 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 met.

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 met, 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 Section 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 52.103(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 §52.103(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 52.98 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, Tier 2, and the generic TS and other operational requirements. 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 under Section VIII of this appendix throughout the period of application and for the term of the license (including any period of renewal).

3. An applicant or licensee who references this appendix shall prepare and maintain written evaluations 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 plant-specific departures from the DCD, including a summary of the evaluation of each. This report must be filed in accordance with the filing requirements applicable to reports in 10 CFR 52.3.

2. An applicant or licensee who references this appendix shall submit updates to its DCD, which reflect the generic changes to

and plant-specific departures from the generic DCD made under Section VIII of this appendix. These updates must be filed under the filing requirements applicable to final safety analysis report updates in 10 CFR 52.3 and 50.71(e).

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

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

b. During the interval from the date of application for a license to the date the Commission makes the finding required by 10 CFR 52.103(g), the report must be submitted semi-annually. Updates to the plant-specific DCD must be submitted annually and may be submitted along with amendments to the application.

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

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 (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 of this appendix to reference Tier 2 when referencing Tier 1. Tier 2 information includes:

1. Information required by §§52.47(a) and 52.47(c), with the exception of generic technical specifications and conceptual design information;
2. Supporting information on the inspections, tests, and analyses that will be performed to demonstrate that the acceptance criteria in the ITAAC have been met; and
3. Combined license (COL) action items (COL license information), which identify certain matters that must 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 are not requirements for the licensee unless such items are restated in the FSAR.
4. 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 Section VIII.B.6 of this appendix. This designation expires for some Tier 2* information under Section VIII.B.6.

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

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

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

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

H. All other terms in this appendix have the meaning set out in 10 CFR 50.2 or 52.1, 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 (12/99 revision) are approved for incorporation by reference by the Director of the Office of the Federal Register on January 24, 2000, in accordance with 5 U.S.C. 552(a) and 1 CFR part 51. Copies of the generic DCD may be obtained from Ronald P. Vijuk, Manager, Passive Plant Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230–0355. A copy of the generic DCD is available for examination and copying at the NRC Public Document Room located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Copies are also available for examination at the NRC Library located at Two White Flint North, 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 characteristics, provided the design activities do not affect the DCD or conflict with the interface requirements.

IV. ADDITIONAL REQUIREMENTS AND RESTRICTIONS

A. An applicant for a combined license that wishes to reference this appendix shall, in addition to complying with the requirements of 10 CFR 52.77, 52.79, and 52.80, comply with the following requirements:

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

2. Include, as part of its application:

a. A plant-specific DCD containing the same type of information and utilizing the same organization and numbering as the generic DCD for the AP600 design, as modified and supplemented by the applicant's exemptions and departures;

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

c. Plant-specific 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 that is not within the scope of this appendix.

3. Include, in the plant-specific DCD, the proprietary information 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 10 CFR 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 December 16, 1999, that are applicable and technically relevant, as described in the FSER (NUREG–1512) and the supplementary information for this section.

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 (a)(2) of 10 CFR 50.55a—ASME Boiler and Pressure Vessel Code;

5. Paragraph (c)(1) of 10 CFR 50.62—Auxiliary (or emergency) feedwater system;

6. Appendix A to 10 CFR part 50, GDC 17—Offsite Power Sources; and

7. 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)(5) in subsequent proceedings for issuance of a combined license, amendment of a combined license, or renewal of a combined license, proceedings held under 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 and Supplement No. 1, 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 under 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 under and in compliance with the change processes in Sections VIII.A.4 and VIII.B.4 of this appendix, but only for that plant;

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

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

7. All environmental issues concerning severe accident mitigation design alternatives associated with the information in the NRC's environmental assessment for the AP600 design and appendix 1B of the generic DCD, for plants referencing this appendix whose site parameters are within those specified in the severe accident mitigation design alternatives 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)(5). The Commission reserves the right to require operational requirements for an applicant or licensee who references this appendix by rule, regulation, order, or license condition.

D. Except 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 at the NRC Web site, *http://www.nrc.gov*, and/or at the NRC 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 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 January 24, 2000, 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)(4).

4. Exemptions from Tier 1 information are governed by the requirements in 10 CFR 52.63(b)(1) and 52.98(f). The Commission will deny a request for an exemption from Tier 1, if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design.

B. Tier 2 information.

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

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

3. The Commission may not require new requirements on Tier 2 information by plant-specific order while this appendix is in effect under §§ 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 52.7 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 52.7. 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 requires a license amendment under paragraphs B.5.b or B.5.c of this section. When evaluating the proposed departure, an applicant or licensee shall consider all matters described in the plant-specific DCD.

b. A proposed departure from Tier 2, other than one affecting resolution of a severe accident issue identified in the plant-specific DCD, requires a license amendment if it would:

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

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

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

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

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

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

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

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

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

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

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

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

d. If a departure requires a license amendment under paragraphs B.5.b or B.5.c 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 paragraph VIII.B.5 of this appendix when departing from Tier 2 information, may petition the NRC to admit into the proceeding such a contention. In addition to compliance with the general requirements of 10 CFR 2.309, the petition must demonstrate that the departure does not comply with paragraph VIII.B.5 of this appendix. Further, the petition must demonstrate that the change bears on an asserted noncompliance with an ITAAC acceptance criterion in the case of a 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 material fact regarding compliance with paragraph VIII.B.5 of this appendix.

6a. 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)(5).

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

(1) 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 and Pressure Vessel Code, Section III, and Code Case—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 and 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 TS and other operational requirements are applicable to all applicants who reference this appendix, except those for which the change has been rendered technically irrelevant by action taken under paragraphs C.3 or C.4 of this section.

3. The Commission may require plant-specific departures on generic 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 2.335 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 52.7. The grant of an exemption must be subject to litigation in the same manner as other issues material to the license hearing.

5. A party to an adjudicatory proceeding for 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 2.309 and must demonstrate why special circumstances as defined in 10 CFR 2.335 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 met.

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 met, 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 52.103(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 §52.103(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 52.98 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, Tier 2, and the generic TS and other operational requirements. 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 under Section VIII of this appendix throughout the period of application and for the term of the license (including any period of renewal).

3. An applicant or licensee who references this appendix shall prepare and maintain written evaluations 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 plant-specific departures from the DCD, including a summary of the evaluation of each. This report must be filed in accordance with the filing requirements applicable to reports in 10 CFR 52.3.

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

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

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

b. During the interval from the date of application for a license to the date the Commission makes the finding required by 10 CFR 52.103(g), the report must be submitted semi-annually. Updates to the plant-specific DCD must be submitted annually and may be submitted along with amendments to the application.

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

Appendix D to Part 52—Design Certification Rule for the AP1000 Design

I. Introduction

Appendix D constitutes the standard design certification for the AP1000[1] design, in accordance with 10 CFR part 52, subpart B. The applicant for certification of the AP1000 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 (Tier 1 information). The design descriptions, interface requirements, and site parameters are derived from Tier 2 information. Tier 1 information includes:

1. Definitions and general provisions;
2. Design descriptions;
3. Inspections, tests, analyses, and acceptance criteria (ITAAC);
4. Significant site parameters; and
5. Significant interface requirements.

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

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

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

3. Combined license (COL) action items (COL license information), which identify certain matters that must 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 are not requirements for the licensee unless such items are restated in the FSAR.

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

[1] AP1000 is a trademark of Westinghouse Electric Company LLC.

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 Section VIII.B.6 of this appendix. This designation expires for some Tier 2* information under paragraph VIII.B.6.

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

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

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

H. All other terms in this appendix have the meaning set out in 10 CFR 50.2, or 52.1, 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 TS in the AP1000 DCD (Revision 15, dated December 8, 2005) are approved for incorporation by reference by the Director of the Office of the Federal Register on February 27, 2006, under 5 U.S.C. 552(a) and 1 CFR part 51. Copies of the generic DCD may be obtained from Ronald P. Vijuk, Manager, Passive Plant Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230–0355. A copy of the generic DCD is also available for examination and copying at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852. Copies are available for examination at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, telephone (301) 415–5610, e-mail *LIBRARY@NRC.GOV* or at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call (202) 741–6030 or go to *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html.*

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 of the DCD), and the generic TS except as otherwise provided in this appendix. Conceptual design information in the generic DCD and the evaluation of severe accident mitigation design alternatives in 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 AP1000 design or NUREG–1793, ''Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,'' (FSER) and Supplement No. 1, then the generic DCD controls.

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

IV. Additional Requirements and Restrictions

A. An applicant for a combined license that wishes to reference this appendix shall, in addition to complying with the requirements of 10 CFR 52.77, 52.79, and 52.80, comply with the following requirements:

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

2. Include, as part of its application:

a. A plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for the AP1000 design, as modified and supplemented by the applicant's exemptions and departures;

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

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

d. Information demonstrating 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. Include, in the plant-specific DCD, the proprietary information and safeguards information referenced in the AP1000 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 10 CFR part 50.

V. Applicable Regulations

A. Except as indicated in paragraph B of this section, the regulations that apply to the AP1000 design are in 10 CFR parts 20, 50, 73, and 100, codified as of January 23, 2006, that are applicable and technically relevant, as described in the FSER (NUREG–1793) and Supplement No. 1.

B. The AP1000 design is exempt from portions of the following regulations:

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

2. Paragraph (c)(1) of 10 CFR 50.62—Auxiliary (or emergency) feedwater system; and

3. Appendix A to 10 CFR part 50, GDC 17—Second offsite power supply circuit.

VI. ISSUE RESOLUTION

A. The Commission has determined that the structures, systems, components, and design features of the AP1000 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 AP1000 design.

B. The Commission considers the following matters resolved within the meaning of 10 CFR 52.63(a)(5) in subsequent proceedings for issuance of a COL, amendment of a COL, or renewal of a COL, proceedings held under 10 CFR 52.103, and enforcement proceedings involving plants referencing this appendix:

1. All nuclear safety issues, except for the generic TS and other operational requirements, associated with the information in the FSER and Supplement No. 1, Tier 1, Tier 2 (including referenced information, which the context indicates is intended as requirements, and the investment protection short-term availability controls in Section 16.3 of the DCD), and the rulemaking record for certification of the AP1000 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 AP1000 design;

3. All generic changes to the DCD under 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 under and in compliance with the change processes in Sections VIII.A.4 and VIII.B.4 of this appendix, but only for that plant;

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

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

7. All environmental issues concerning severe accident mitigation design alternatives associated with the information in the NRC's EA for the AP1000 design and Appendix 1B of the generic DCD, for plants referencing this appendix whose site parameters are within those specified in the severe accident mitigation design alternatives 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)(5). The Commission reserves the right to require operational requirements for an applicant or licensee who references this appendix by rule, regulation, order, or license condition.

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

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

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

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

E.1. Persons who wish to review proprietary and safeguards information or other secondary references in the AP1000 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 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 February 27, 2006, 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)(4).

4. Exemptions from Tier 1 information are governed by the requirements in 10 CFR 52.63(b)(1) and 52.98(f). The Commission will deny a request for an exemption from Tier 1, if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design.

B. Tier 2 information.

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

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

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

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

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

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

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

b. A proposed departure from Tier 2, other than one affecting resolution of a severe accident issue identified in the plant-specific DCD, requires a license amendment if it would:

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

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

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

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

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

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

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

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

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

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

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

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

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 paragraph VIII.B.5 of this appendix when departing from Tier 2 information, may petition to admit into the proceeding such a contention. In addition to compliance with the general requirements of 10 CFR 2.309, the petition must demonstrate that the departure does not comply with paragraph VIII.B.5 of this appendix. Further, the petition must demonstrate that the change bears on an asserted noncompliance with an ITAAC acceptance criterion in the case of a 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 material fact regarding compliance with paragraph VIII.B.5 of this appendix.

6.a. An applicant who references this appendix may not depart from Tier 2* information, which is designated with italicized text or brackets and an asterisk in the generic DCD, without NRC approval. The departure will not be considered a resolved issue, within the meaning of Section VI of this appendix and 10 CFR 52.63(a)(5).

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

(1) Maximum fuel rod average burn-up.

(2) Fuel principal design requirements.

(3) Fuel criteria evaluation process.

(4) Fire areas.

(5) Human factors engineering.

(6) Small-break loss-of-coolant accident (LOCA) analysis methodology.

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 under paragraph B.6.b of this section. After the plant first achieves full power, the following Tier 2* matters revert to Tier 2 status and are subject to the departure provisions in paragraph B.5 of this section.

(1) Nuclear Island structural dimensions.

(2) American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code), Section III, and Code Case–284.

(3) Design Summary of Critical Sections.

(4) American Concrete Institute (ACI) 318, ACI 349, American National Standards Institute/American Institute of Steel Construction (ANSI/AISC)–690, and American Iron and Steel Institute (AISI), ''Specification for the Design of Cold Formed Steel Structural Members, Part 1 and 2,'' 1996 Edition and 2000 Supplement.

(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 and control system design processes, methods, and standards.

(10) Passive residual heat removal (PRHR) natural circulation test (first plant only).

(11) Automatic depressurization system (ADS) and core make-up tank (CMT) verification tests (first three plants only).

(12) Polar crane parked orientation.

(13) Piping design acceptance criteria.

(14) Containment vessel design parameters.

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

C. Operational requirements.

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

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

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

4. An applicant who references this appendix may request an exemption from the generic 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 52.7. The grant of an exemption must be subject to litigation in the same manner as other issues material to the license hearing.

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

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

IX. 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. A licensee may also 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 met.

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. If an activity is subject to an ITAAC and the applicant or licensee who references this appendix has not demonstrated that the ITAAC has been met, the applicant or licensee may either take corrective actions to successfully complete that ITAAC, request an exemption from the ITAAC under 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 that 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 52.103(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 §52.103(a) hearing, their expiration will occur upon final Commission action in such a 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 52.98 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, Tier 2, and the generic TS and other operational requirements. 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 under Section VIII of this appendix throughout the period of application and for the term of the license (including any period of renewal).

3. An applicant or licensee who references this appendix shall prepare and maintain written evaluations 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 plant-specific departures from the DCD, including a summary of the evaluation of each. This report must be filed in accordance with the filing requirements applicable to reports in 10 CFR 52.3.

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

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

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

b. During the interval from the date of application for a license to the date the Commission makes its findings required by 10 CFR 52.103(g), the report must be submitted semi-annually. Updates to the plant-specific DCD must be submitted annually and may be submitted along with amendments to the application.

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

APPENDIXES E–M TO PART 52 [RESERVED]

APPENDIX N TO PART 52—STANDARDIZATION OF NUCLEAR POWER PLANT DESIGNS: COMBINED LICENSES TO CONSTRUCT AND OPERATE NUCLEAR POWER REACTORS OF IDENTICAL DESIGN AT MULTIPLE SITES

The Commission's regulations in part 2 of this chapter specifically provide for the holding of hearings on particular issues separately from other issues involved in hearings in licensing proceedings, and for the consolidation of adjudicatory proceedings and of the presentations of parties in adjudicatory proceedings such as licensing proceedings (§§2.316 and 2.317 of this chapter).

This appendix sets out the particular requirements and provisions applicable to situations in which applications for combined licenses under subpart C of this part are filed by one or more applicants for licenses to construct and operate nuclear power reactors of identical design (''common design'') to be located at multiple sites.[1]

1. Except as otherwise specified in this appendix or as the context otherwise indicates, the provisions of subpart C of this part and subpart D of part 2 of this chapter apply to combined license applications subject to this appendix.

2. Each combined license application submitted pursuant to this appendix must be submitted as specified in §52.75 and 10 CFR 2.101. Each application must state that the applicant wishes to have the application considered under 10 CFR part 52, appendix N, and must list each of the applications to be treated together under this appendix.

3. Each application must include the information required by §§52.77, 52.79, and 52.80(a), *provided however*, that the application must identify the common design, and, if applicable, reference a standard design certification under subpart B of this part, or the use of a reactor manufactured under subpart F of this part. The final safety analysis report for each application must either incorporate by reference or include the final safety analysis of the common design, including, if applicable, the final safety analysis report for the referenced design certification or the manufactured reactor.[2]

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[1] If the design for the power reactor(s) proposed in a particular application is not identical to the others, that application may not be processed under this appendix and subpart D of part 2 of this chapter.

[2] As used in this appendix, the design of a nuclear power reactor included in a single referenced safety analysis report means the design of those structures, systems, and

Continued

4. Each combined license application submitted pursuant to this appendix must contain an environmental report as required by §52.80(b), and which complies with the applicable provisions of 10 CFR part 51, *provided, however*, that the application may incorporate by reference a single environmental report on the environmental impacts of the common design.

5. Upon a determination that each application is acceptable for docketing under 10 CFR 2.101, each application will be docketed and a notice of docketing for each application will be published in the FEDERAL REGISTER, in accordance with 10 CFR 2.104, *provided, however*, that the notice must state that the application will be processed under the provisions of 10 CFR part 52, appendix N, and subpart D of part 2 of this chapter. As the discretion of the Commission, a single notice of docketing for multiple applications may be published in the FEDERAL REGISTER.

6. The NRC staff shall prepare draft and final environmental impact statements for each of the applications under part 51 of this chapter. Scoping under 10 CFR 51.28 and 51.29 for each of the combined license applications may be conducted simultaneously and joint scoping may be conducted with respect to the environmental issues relevant to the common design.

If the applications reference a standard design certification, then the environmental impact statement for each of the applications must incorporate by reference the design certification environmental assessment. If the applications do not reference a standard design certification, then the NRC staff shall prepare draft and final supplemental environmental impact statements which address severe accident mitigation design alternatives for the common design, which must be incorporated by reference into the environmental impact statement prepared for each application. Scoping under 10 CFR 51.28 and 51.29 for the supplemental environmental impact statement may be conducted simultaneously, and may be part of the scoping for each of the combined license applications.

7. The ACRS shall report on each of the applications as required by §52.87. Each report must be limited to those safety matters for each application which are not relevant to the common design. In addition, the ACRS shall separately report on the safety of the common design, *provided, however*, that the report need not address the safety of a referenced standard design certification or reactor manufactured under subpart F of this part.

components important to radiological health and safety and the common defense and security.

8. The Commission shall designate a presiding officer to conduct the proceeding with respect to the health and safety, common defense and security, and environmental matters relating to the common design. The hearing will be governed by the applicable provisions of subparts A, C, G, L, N, and O of part 2 of this chapter relating to applications for combined licenses. The presiding officer shall issue a partial initial decision on the common design.

PART 53 [RESERVED]

PART 54—REQUIREMENTS FOR RENEWAL OF OPERATING LICENSES FOR NUCLEAR POWER PLANTS

GENERAL PROVISIONS

AUTHORITY: Secs. 102, 103, 104, 161, 181, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs 201, 202, 206, 88 Stat. 1242, 1244, as amended (42 U.S.C. 5841, 5842), E.O. 12829, 3 CFR, 1993 Comp., p. 570;